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RECIP.NAME RECIPIENT AFFILIATION
DENTON,H.R. Office of Nuclear Reactor Regulation

2-13-79

SUBJECT: Forwards request for changes to App A of Tech Spec for License DPR-74. Changes cover: fire detection instrumentation emergency core cooling sys, & turbine overspeed protection.

(see reports)

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February 13, 1979
AEP:NRC:00111

Donald C. Cook Nuclear Plant Unit No. 2
Docket No. 50-316
License DPR No. 74

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Denton:

This letter serves to request changes in several areas of the Donald C. Cook Nuclear Plant Unit No. 2 Appendix 'A' Technical Specifications. Attachment 'A' to this letter contains the description and review of each change. Attachment 'B' contains a copy of the corresponding revised pages. We would like to point out that a number of these changes are editorial in nature and noted as such. We request that the NRC file and process this technical specification change package as a single amendment.

All of the proposed technical specification changes contained herein have been reviewed and approved by the Plant Nuclear Safety Review Committee (PNSRC) and the AEPSC Nuclear Safety & Design Review Committee (NSDRC). The result of these reviews indicates that in no instance will the proposed technical specification change adversely affect the health and safety of the public.

This application for technical specification revision is considered to be a Class II License Amendment as per the provisions of 10 CFR 170.22. As required by Part 170 Subsection 22 a check for $1,200.00 accompanies this submittal.

Very truly yours,

G. P. Maloney
Vice President

GPM:em

Sworn and subscribed to before me
this 13TH day of February, 1979 in
New York County, New York

Gregory M. G[illegible]
Notary Public

GREGORY M. GURICAN
Notary Public, State of New York
No. 31-4643431
Qualified In New York County
Commission Expires March 30, 1979

7902230104

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ATTACHMENT 'A' TO AEP:NRC:00111

PROPOSED REVISIONS TO THE
DONALD C. COOK NUCLEAR PLANT UNIT NO. 2
APPENDIX 'A' TECHNICAL SPECIFICATIONS

50-316
7902230104
2/13/79

CHANGE NO. 1

Revision to Table 3.3-11; "Fire Detection Instrumentation"

This change involves a revision to Table 3.3-11 entitled, "Fire Detection Instrumentation" on page 3/4 3-51. The minimum number of thermistor detectors specified for the Containment Quadrants 1, 2, 3 and 4 does not agree with the as built installation of the thermistor detection system for the containment cable trays. This change has been discussed with members of the NRC staff and is consistent with the requirements of the fire protection program for the Donald C. Cook Nuclear Plant. This change will not adversely affect the health and safety of the public.

CHANGE NO. 2

This change involves a revision to surveillance requirement 4.5.2.h.. We are requesting that the flow rates listed for the Boron Injection System (single pump), and Safety Injection System (single pump) be revised to assure consistency between the pump design capacities, the plant safety analysis and the technical specifications. These changes will not adversely affect the health and safety of the public.

CHANGE NO. 3 (EDITORIAL)

This change involves a revision to the Bases Section page B 3/4 4-4. The reason for this change is to provide a clarifying statement as to how the 52 gpm controlled leakage limitation was accounted for in the accident analysis for the Donald C. Cook Nuclear Plant.

CHANGE NO. 4

This change involves revising Surveillance Requirement 4.6.2.2.d of the Spray Additive System Technical Specification Page 3/4 6-12 Unit 2.

The current surveillance requirement is unworkable as written and this revision will provide better consistency between the intent of the surveillance requirement and the design capability of the Spray Additive System. In addition, the revised flow rates included in the attached page 3/4 6-12 will provide consistency with the flow and pH requirements used in the safety analysis and also assure that the contents of the Spray Additive Tank are added to the system at the proper rate. Analyses have been performed to show that with a flow rate from the spray additive tank of 20 to 50 gpm, the pH of the spray solution will be in accordance with the requirements for the accident analysis in the FSAR. This change is consistent with the functional requirements of the spray additive system included in the safety analysis and will not adversely affect the health and safety of the public.

CHANGE NO. 5 (EDITORIAL)

This change involves a revision to Table 3.3-4 on page 3/4 3-25a. The trip setpoint and allowable value for Item 6a and 7a for Steam Generator Water Level Low-Low must be consistent with Item 13 of Table 2.2-1. The Reactor Trip System Instrumentation Trip Setpoints from Table 2.2-1 are being used for Item 6a and 7a of Table 3.3.-4 so this change is only editorial in nature. These changes are shown on the attached revised page 3/4 3-25a.

CHANGE NO. 6

This change involves a revision of the Applicability of Technical Specification 3.9.9. We are requesting that the Applicability be changed from "Mode 6" to "During Core Alterations or movement of Irradiated Fuel within the Containment." The reason for this change is for consistency with Specification 3.9.4 in that 3.9.4 allows certain building penetrations (air locks) to be open while not moving irradiated fuel during Mode 6. Further, since it is not possible to establish containment integrity with the air locks open, both Specifications 3.9.4 and 3.9.9 should be consistent with regard to their Applicability. This change is consistent with the intent of the Technical Specifications and will not have any adverse affect on the health and safety of the public.

CHANGE NO. 7

This change involves a revision to Table 3.3-5 on page 3/4 3-27. The response times for Items 4a and 4b must be revised as follows:

Item 4a - Change 13.0 to 12.0
Change 23.0 to 24.0

Item 4b - Change 3.0 to 2.0

The revised response times for this mitigating signal and function are those that were assumed in the various safety analyses. The reason for this change is to make the Technical Specification requirements consistent with the assumptions of the safety analysis for Unit 2 of the Donald C. Cook Nuclear Plant, and hence will have no adverse affect on the health and safety of the public.

CHANGE NO. 8

This change involves a revision to the definitions section on page 1-5. Definition 1.22 measures the Reactor Trip System Response Time by using the loss of stationary gripper coil voltage. However, this loss of voltage is a result of the reactor trip breakers opening. We are requesting measurement of the time interval by using the opening of the

CHANGE NO. 8 (Cont'd.)

reactor trip breakers as shown in the attached revised page 1-5. This change provides better consistency between the Technical Specifications and how the response time interval is actually measured. When the reactor trip breakers open, we get a status light indication of no voltage at the stationary gripper coil. This change has no adverse affect on the health and safety of the public.

CHANGE NO. 9

This change involves a revision to Surveillance Requirement 4.3.3.7.1. We are presently required to update the incore flux map every 31 days. However, since the flux is burnup dependent, we request that this be changed to 31 EFPD as shown in the attached revised page 3/4 3-48. The reason for this change is that a flux map taken every 31 EFPD will be more meaningful in terms of the dependence on accumulated core burnup and the requirements for taking a meaningful flux map. This change will not adversely affect the health and safety of the public.

CHANGE NO. 10

This change involves a revision to Table 3.6-1. We have installed an automatic trip (isolation) valve on the return line to the containment from the Containment Air Particulate/Radio Gas Monitors (R-11 & R-12). This valve is a Phase "B" Containment Isolation Valve and should be included in the Technical Specifications as shown in the attached revised page 3/4 6-21. Also note that in order to have the proper numbering sequences, we have revised the valve numbering on page 3/4 6-21 thru 6-24. This change will not adversely affect the health and safety of the public. See Notes (1) and (2) on Page 6.

CHANGE NO. 11 (EDITORIAL)

This change involves a revision to Figure 3.4-2 on page 3/4 4-25. The curve shown for the Reactor Coolant System limiting heatup rate is for a maximum rate of 100°F/Hr. and the caption at the bottom of the page indicates 60°F/Hr. This change is editorial in nature and is shown in the attached revised page 3/4 4-25. This change will not adversely affect the health and safety of the public.

CHANGE NO. 12

This change revises Technical Specifications 6.5.2.2, 6.5.2.6, 6.5.2.9, 6.5.2.10, and 6.6.1 (pages 6-9, 6-11 and 6-12). These specifications will be amended to indicated the revised NSDRC membership, the number of members/alternates required to constitute a quorum of the NSDRC, and to clear up minor (editorial) inaccuracies with respect to AEPSC management titles. The above changes will not adversely affect the health and safety of the public.

CHANGE NO. 13 (EDITORIAL)

This change involves a revision to Specification 3/4.3.4 in ACTION statement b on page 3/4 3-53. The word "overspeed" was repeated twice in succession and this is an editorial error. The attached revised page 3/4 3-53 has this editorial error corrected. This change will not adversely affect the health and safety of the public.

CHANGE NO. 14 (EDITORIAL)

This change involves a revision to the footnote on Table 2.2-1, page 2-9. The words "excluding transmitter" have been added to the footnote. This change will not adversely affect the health and safety of the public.

CHANGE NO. 15 (EDITORIAL)

This change involves a revision to Surveillance Requirement 4.1.1.1.1.d on page 3/4 1-2. The reference to Specification 3.1.3.5 is not correct and is an editorial error. The correct reference should be 3.1.3.6 as indicated on the attached revised page 3/4 1-2. This change will not adversely affect the health and safety of the public.

CHANGE NO. 16 (EDITORIAL)

This change involves a revision to Table 3.3-4 on page 3/4 3-25. The trip setpoint and allowable value for functional unit 5.a indicates ≥ 67% and ≥ 68% respectively and this is an editorial error. The inequality signs should be switched around to indicate ≤ 67% and ≤ 68% as shown in the attached revised page 3/4 3-25. This change will not adversely affect the health and safety of the public.

CHANGE NO. 17

Containment Air Recirculation Systems

This change revised Technical Specifications 4.6.5.6(a) and (d) on page 3/4 6-44. The delay times for the containment air recirculation fan auto-start and the suction line valve opening time (on auto-start signal) will be changed to 9 ± 1 minutes. We have been informed by Westinghouse that a value of seven minutes was used in the safety analysis for fan - auto start delay time. The present hydrogen analysis for Unit 2 (FSAR Section 14.3.6 -Unit 2 Yellow Pages) assumes a maximum auto-start delay time of ten minutes. Therefore, the above indicated changes will provide additional margin, in the conservative direction, between the values assumed in the safety analyses and the Technical Specification values. The above changes will not adversely effect the health and safety of the public.

CHANGE NO. 18

This change deletes specification 6.10.2.C on page 6-19. Specification 6.10.2.C requires that facility radiation and contamination survey records be retained for the duration of the Facility Operating License. Deletion of the specification will bring the Cook Plant Technical Specifications in line with the present standardized technical specification (STS) format. This change will not adversely affect the health and safety of the public.

CHANGE NO. 19

This change revised the footnote to Table 3.2.1. This change will allow for rapid power decreases without violating the pressurizer pressure 'limit' of the table 3.2-1 (an event which requires an LER be submitted to the NRC). This change will not adversely affect the health and safety of the public.

CHANGE NO. 20 (EDITORIAL)

This change involves a revision to Technical Specification 3/4.4.1 on page 3/4 4-1, bases section 3/4.4.1 on page B 3/4 4-1 and Table 3.3-1 on page 3/4 3-8. The value used for Rated Thermal Power for P-8 in all N-1 loop analyses is 31% steady state initial power level. Although 3 loop operation is not permitted, this change is for consistency with the safety analysis for Unit 2 of the Donald C. Cook Nuclear Plant and is editorial in nature. This change is shown in the attached revised pages 3/4 4-1, B 3/4 4-1 and 3/4 3-8. This change will not adversely affect the health and safety of the public.

CHANGE NO. 21 (EDITORIAL)

This change corrects typographical error on page 3/4 2-19. Technical Specification 4.2.6.1 incorrectly refers to Specification 3.3.3.6. The correct reference, as indicated on the attached page 3/4 2-19, is to Specification 3.3.3.7. This change is editorial in nature and will not adversely effect the health and safety of the public.

CHANGE NO. 22 (EDITORIAL)

This change involves a revision to Table 3.3-1 on page 3/4 3-3. The words "Same Loop" should be added to Item 15, under total no. of channels as shown on the attached revised page 3/4 3-3. This change is editorial in nature. This change will not adversely affect the health and safety of the public.

CHANGE NO. 23 (EDITORIAL)

This change involves a revision to bases section 3/4 2.2 on page B 3/4 2-4. In paragraph 'a' the word "Rod" should be changed to "Rods" as shown in the attached revised page B 3/4 2-4. This change will not adversely affect the health and safety of the public.

CHANGE NO. 24 (EDITORIAL)

This change involves a revision to Figure 6.2-2 "Facility Organization" (Page 6-3) and Table 6.2-1 (Page 6-4). We have had a system wide change in the titles of the staff members of our operating plant. Certain "Foremen" are now "Supervisors" and "Supervisors" are now "Superintendents." The attached figure 6.2-2 has been revised accordingly.

Note (1): By letter dated 03 February 1978 AEPSC requested a Technical Specification revision to change the closure times of the "containment purge and exhaust valves" to $\leq$ 5 sec. (Item 'C' of Table 3.6.1). This change request has not received NRC approval as of this writing.

Note (2): Typographical errors on page 3/4 6-28 have also been corrected.

ATTACHMENT 'B' TO AEP:NRC:00111

PROPOSED REVISIONS TO THE
DONALD C. COOK NUCLEAR PLANT UNIT NO. 2
APPENDIX 'A' TECHNICAL SPECIFICATIONS

AEP:NRC:00111

CHANGE NO. /

TABLE 3.3-11

FIRE DETECTION INSTRUMENTATION

INSTRUMENT LOCATION	MINIMUM INSTRUMENTS OPERABLE	
	SMOKE (IONIZATION)	HEAT (THERMISTOR)
1. Containment		
Zone 1, Quadrant 1 Cable Tunnel	3	
Zone 2, Quadrant 2 Cable Tunnel	5	
Zone 3, Quadrant 3N Cable Tunnel	3	
Zone 4, Quadrant 3M Cable Tunnel	3	
Zone 5, Quadrant 3S Cable Tunnel	2	
Zone 6, Quadrant 4 Cable Tunnel	5	
Quadrant 1		12
Quadrant 2		5
Quadrant 3		23
Quadrant 4		11
2-HV-CFT-1 Charcoal Filters		1
2-HV-CFT-2 Charcoal Filters		1
2. Control Room		
Zone 16, Control Room	8	
3. Cable Spreading Room		
Zone 10, Switchgear Cable Vault	10	
Zone 11, Auxiliary Cable Vault	5	
Zone 12, Control Room Cable Vault	24	
Zone 13, Control Room Cable Vault	25	
4. Diesel Generator		
Diesel Generator Room 2AB		1
Diesel Generator Room 2CD		1
5. Diesel Fuel Oil Room		1

TABLE 3.3-11 (Cont'd)

FIRE DETECTION INSTRUMENTATION

INSTRUMENT LOCATION	MINIMUM INSTRUMENTS OPERABLE	
	SMOKE (IONIZATION)	HEAT (THERMISTOR)
6. Auxiliary Building		
Elevation 573 ft. *	5	
Elevation 587 ft. *	20	
Elevation 609 ft. *	20	
Elevation 633 ft. *	20	
Elevation 650 ft. *	26	
Zone 7, 4 Kv Switchgear	3	
Zone 8, Engineered Safety Switchgear	7	
Zone 9, CRD Switchgear	6	
2-HV-AES-1 Charcoal Filters		1
2-HV-AES-2 Charcoal Filters		1
12-HV-AFX Charcoal Filters		1
2-HV-CPR Charcoal Filters		1
2-HV-CIPX Charcoal Filters		1
2-HV-ACRF Charcoal Filters		1
7. Fuel Storage		
New Fuel Storage Room *	3	

*Shared system with D. C. COOK - UNIT 1.

AEP:NRC:00111

CHANGE NO. 2

SURVEILLANCE REQUIREMENTS (Continued)

d. At least once per 18 months by:

   1. Verifying automatic isolation and interlock action of the RHR system from the Reactor Coolant System when the Reactor Coolant System pressure is above 600 psig.

   2. A visual inspection of the containment sump and verifying that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or corrosion.

e. At least once per 18 months, during shutdown, by:

   1. Verifying that each automatic valve in the flow path actuates to its correct position on a Safety Injection test signal.

   2. Verifying that each of the following pumps start automatically upon receipt of a safety injection test signal:

      a) Centrifugal charging pump

      b) Safety injection pump

      c) Residual heat removal pump

f. By verifying that each of the following pumps develops the indicated discharge pressure on recirculation flow when tested pursuant to Specification 4.0.5:

   1. Centrifugal charging pump $\geq$ 2405 psig

   2. Safety Injection pump $\geq$ 1445 psig

   3. Residual heat removal pump $\geq$ 195 psig

g. By verifying the correct position of each mechanical stop for the the following Emergency Core Cooling System throttle valves:

   1. Within 4 hours following completion of each valve stroking operation or maintenance on the valve when the ECCS subsystems are required to be OPERABLE.

SURVEILLANCE REQUIREMENTS (Continued)

2. At least once per 18 months.

Boron Injection Throttle Valves	Safety Injection Throttle Valves
Valve Number	Valve Number
1. 2-SI-141 L1	1. 2-SI-121 N
2. 2-SI-141 L2	2. 2-SI-121 S
3. 2-SI-141 L3	
4. 2-SI-141 L4	

h. By performing a flow balance test during shutdown following completion of modifications to the ECCS subsystem that alter the subsystem flow characteristics and verifying the following flow rates:

Boron Injection System Single Pump *	Safety Injection System Single Pump **
Loop 1 Boron Injection Flow 117.5 gpm	Loop 1 and 4 Cold Leg Flow ≥ 300 gpm
Loop 2 Boron Injection Flow 117.5 gpm	Loop 2 and 3 Cold Leg Flow ≥ 300 gpm.
Loop Boron Injection Flow 117.5 gpm	** Total SIS (single pump) flow, including mini-flow, shall not exceed 650 gpm.
Loop 4 Boron Injection Flow 117.5 gpm	

*The flow rate in each Boron Injection (BI) line should be adjusted to provide 117.5 gpm (nominal) flow into each loop. Under these conditions there is zero mini-flow and 80 gpm simulated RCP seal injection line flow. The actual flow in each BI line may deviate from the nominal so long as the difference between the highest and lowest flow is 10 gpm or less and the total flow to the four branch lines does not exceed 470 gpm. Minimum flow (total flow) required is 345.8 gpm to the three most conservative (lowest flow) branch lines.

AEP:NRC:00111

CHANGE NO. 3

be detected by radiation monitors of steam generator blowdown. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged.

Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging will be required for all tubes with imperfections exceeding the plugging limit of 40% of the tube nominal wall thickness. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be promptly reported to the Commission pursuant to Specification 6.9.1 prior to resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary.

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 LEAKAGE DETECTION SYSTEMS

The RCS leakage detection systems required by this specification are provided to monitor and detect leakage from the Reactor Coolant Pressure Boundary. These detection systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973.

3/4.4.6.2 OPERATIONAL LEAKAGE

Industry experience has shown that while a limited amount of leakage is expected from the RCS, the unidentified portion of this leakage can be reduced to a threshold value of less than 1 GPM. This threshold value is sufficiently low to ensure early detection of additional leakage.

The 10 GPM IDENTIFIED LEAKAGE limitations provides allowance for a limited amount of leakage from known sources whose presence will not interfere with the detection of UNIDENTIFIED LEAKAGE by the leakage detection systems.

BASES

The CONTROLLED LEAKAGE limitation restricts operation when the total flow supplied to the reactor coolant pump seals exceeds 52 GPM with the modulating valve in the supply line fully open at a nominal RCS pressure of 2235 psig. This limitation is based on the maximum seal injection flow capability of the Reactor Coolant Pumps and ensures a maximum safety injection flow assumed in the accident analysis.

The total steam generator tube leakage limit of 1 GPM for all steam generators ensures that the dosage contribution from the tube leakage will be limited to a small fraction of Part 100 limits in the event of either a steam generator tube rupture or steam line break. The 1 GPM limit is consistent with the assumptions used in the analysis of these accidents. The 500 gpd leakage limit per steam generator ensures that steam generator tube integrity is maintained in the event of a main steam line rupture or under LOCA conditions.

PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Therefore, the presence of any PRESSURE BOUNDARY LEAKAGE requires the unit to be promptly placed in COLD SHUTDOWN.

3/4.4.7 CHEMISTRY

The limitations on Reactor Coolant System chemistry ensure that corrosion of the Reactor Coolant System is minimized and reduces the potential for Reactor Coolant System leakage or failure due to stress corrosion. Maintaining the chemistry within the Steady State Limits provides adequate corrosion protection to ensure the structural integrity of the Reactor Coolant System over the life of the plant. The associated effects of exceeding the oxygen, chloride and fluoride limits are time and temperature dependent. Corrosion studies show that operation may be continued with contaminant concentration levels in excess of the Steady State Limits, up to the Transient Limits, for the specified limited time intervals without having a significant effect on the structural integrity of teh Reactor Coolant System. The time interval permitting continued operation within the restrictions of the Transient Limits provides time for taking corrective actions to restore the contaminant concentrations to within the Steady State Limits.

The surveillance requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

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CHANGE NO. 4

SPRAY ADDITIVE SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.2 The spray additive system shall be OPERABLE with:

a. A spray additive tank containing a volume of between 4000 and 4600 gallons of between 30 and 34 percent by weight NaOH solution, and

b. Two spray additive eductors each capable of adding NaOH solution from the chemical additive tank to a containment spray system pump flow.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the spray additive system inoperable, restore the system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours; restore the spray additive system to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.2 The spray additive system shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

b. At least once per 6 months by:

1. Verifying the contained solution volume in the tank, and

2. Verifying the concentration of the NaOH solution by chemical analysis.

SURVEILLANCE REQUIREMENTS (Continued)

c. At least once per 18 months during shutdown, by verifying that each automatic valve in the flow path actuates to its correct position on a Containment Pressure--High-High test signal.

d. At least once per 5 years by verifying a water flow rate of at least 20gpm ($\geq$ 20 gpm) but not to exceed 50gpm ($\leq$ 50gpm) from the spray additive tank to each containment spray system with the spray pump operating on recirculation with a pump discharge pressure $\geq$ 225 psig.

AEP:NRC:00111

CHANGE NO. 5

TABLE 3.3-4 (Continued)

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	$\geq$ 15% of narrow range instrument span each steam generator	$\geq$ 14% of narrow range instrument span each steam generator
b. 4 kv Bus Loss of Voltage	2400 volts	2400 volts
7. TURBINE DRIVEN AUXILIARY FEEDWATER PUMPS		
a. Steam Generator Water Level -- Low-Low	$\geq$ 15% of narrow range instrument span each steam generator	$\geq$ 14% of narrow range instrument span each steam generator
8. LOSS OF POWER		
a. 4 kv Bus Loss of Voltage	2400 volts	2400 volts
b. Grid Degraded Voltage	32.4 kvolts with a 2.0 + 0.1/-2.0 second time delay	32.4 kvolts with a 2.0 + 0.1/-2 second time delay

TABLE 3.3-5

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
1. Manual	
a. Safety Injection (ECCS)	Not Applicable
Feedwater Isolation	Not Applicable
Reactor Trip (SI)	Not Applicable
Containment Isolation-Phase "A"	Not Applicable
Containment Purge and Exhaust Isolation	Not Applicable
Auxiliary Feedwater Pumps	Not Applicable
Essential Service Water System	Not Applicable
Containment Air Recirculation Fan	Not Applicable
b. Containment Spray	Not Applicable
Containment Isolation-Phase "B"	Not Applicable
Containment Purge and Exhaust Isolation	Not Applicable
c. Containment Isolation-Phase "A"	Not Applicable
Containment Purge and Exhaust Isolation	Not Applicable
d. Steam Line Isolation	Not Applicable
2. Containment Pressure-High	
a. Safety Injection (ECCS)	Not Applicable
b. Reactor Trip (from SI)	Not Applicable
c. Feedwater Isolation	Not Applicable
d. Containment Isolation-Phase "A"	Not Applicable
e. Containment Purge and Exhaust Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	Not Applicable
g. Essential Service Water System	Not Applicable

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CHANGE NO. 6

CONTAINMENT PURGE AND EXHAUST ISOLATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.9 The Containment Purge and Exhaust isolation system shall be OPERABLE.

APPLICABILITY: During CORE ALTERATIONS or movement of irradiated fuel within the containment.

ACTION:

With the Containment Purge and Exhaust isolation system inoperable, close each of the Purge and Exhaust penetrations providing direct access from the containment atmosphere to the outside atmosphere. The provision of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.9 The Containment Purge and Exhaust isolation system shall be demonstrated OPERABLE within 100 hours prior to the start of and at least once per 7 days during CORE ALTERATIONS by verifying that containment Purge and Exhaust isolation occurs on manual initiation and on a high radiation test signal from each of the containment radiation monitoring instrumentation channels.

WATER LEVEL - REACTOR VESSEL

LIMITING CONDITION FOR OPERATION

3.9.10 At least, 23 feet of water shall be maintained over the top of irradiated fuel assemblies seated within the reactor pressure vessel.

APPLICABILITY: During movement of fuel assemblies or control rods within the reactor pressure vessel while in MODE 6.

ACTION:

With the requirements of the above specification not satisfied, suspend all operations involving movement of fuel assemblies or control rods within the pressure vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.10 The water level shall be determined to be at least its minimum required depth within 2 hours prior to the start of and at least once per 24 hours thereafter during movement of fuel assemblies or control rods.

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CHANGE NO. 7

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
3. Pressurizer Pressure-Low with Pressurizer Level-Low	
a. Safety Injection (ECCS)	≤ 24.0*/12.0#
b. Reactor Trip (from SI)	≤ 2.0
c. Feedwater Isolation	≤ 8.0
d. Containment Isolation-Phase "A"	≤ 18.0#
e. Containment Purge and Exhaust Isolation	Not Applicable
f. Motor Driven Auxiliary Feedwater Pumps	≤ 60.0
g. Essential Service Water System	≤ 48.0*/13.0#
4. Differential Pressure Between Steam Lines-High	
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	≤ 13.0#/48.0##
5. Steam Flow in Two Steam Lines - High Coincident with T_{avg}--Low-Low	
a. Safety Injection (ECCS)	Not Applicable
b. Reactor Trip (from SI)	Not Applicable
c. Feedwater Isolation	Not Applicable
d. Containment Isolation-Phase "A"	Not Applicable
e. Containment Purge and Exhaust Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	Not Applicable
g. Essential Service Water System	Not Applicable
h. Steam Line Isolation	Not Applicable

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	1
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	

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CHANGE NO. 8

STAGGERED TEST BASIS

1.20 A STAGGERED TEST BASIS shall consist of:

a. A test schedule for n systems, subsystems, trains or other designated components obtained by dividing the specified test interval into n equal subintervals,

b. The testing of one system, subsystem, train or other designated component at the begining of each subinterval.

FREQUENCY NOTATION

1.21 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.2.

REACTOR TRIP SYSTEM RESPONSE TIME

1.22 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until the reactor trip breakers open.

ENGINEERED SAFETY FEATURE RESPONSE TIME

1.23 The ENGINEERED SAFETY FEATURE RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays where applicable.

AXIAL FLUX DIFFERENCE

1.24 AXIAL FLUX DIFFERENCE shall be the difference in normalized flux signals between the top and bottom halves of a two section excore neutron detector.

PHYSICS TESTS

1.25 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and 1) described in Chapter 13.0 of the FSAR, 2) authorized under the provisions of 10 CFR 50.59, or 3) otherwise approved by the Commission.

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

1.26 $\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disnitegration (in MeV) for isotopes, other than iodines, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

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CHANGE NO. 9

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. Steam Generator Water Level - Wide Range	M	R
9. RWST Water Level	M	R
10. Boric Acid Tank Solution Level	M	R

AXIAL POWER DISTRIBUTION MONITORING SYSTEM

LIMITING CONDITION FOR OPERATION

3.3.3.7 The axial power distribution monitoring system (APDMS) shall be OPERABLE with:

a. At least two detector thimbles available for which $\overline{R}$ has been determined from full incore flux maps. These two thimbles shall be those having the lowest uncertainty, σ, covering the full configuration of permissible rod patterns permitted at RATED THERMAL POWER.

b. At least two movable detectors, with associated devices and readout equipment, available for mapping $F_j(Z)$ in the above required thimbles.

APPLICABILITY: When the APDMS is used for monitoring the axial power distribution*#.

ACTION: With the APDMS inoperable, do not use the system for determining the Axial Power Distribution. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.7.1 The full incore flux maps used to determine $\overline{R}$ and for monitoring $F_j(Z)$ shall be updated at least once per 31 EFPD. The continued accuracy and representativeness of the selected thimbles shall be verified by using their latest flux maps to update the $\overline{R}$ for each representative thimble. The original uncertainty, σ, shall not be updated, except as follows:

*Except as provided in Specification 4.2.6.1.b.
#The APDMS may be out of service when surveillance for determining power distribution maps is being performed.

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CHANGE NO. 10

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	ISOLATION TIME IN SECONDS
A. PHASE "A" ISOLATION (Continued)		
73. XCR-102	Control Air to Containment Isolation	$\leq$ 10
74. XCR-103	Control Air to Containment	$\leq$ 10
B. Phase "B" ISOLATION		
1. CCM-451	CCW from RCP Oil Coolers	$\leq$ 60
2. CCM-452	CCW from RCP Oil Coolers	$\leq$ 60
3. CCM-453	CCW from RCP Thermal Barrier	$\leq$ 30
4. CCM-454	CCW from RCP Thermal Barrier	$\leq$ 30
5. CCM-458	CCW to RCP Oil Coolers & Thermal Barrier	$\leq$ 60
6. CCM-459	CCW to RCP Oil Coolers & Thermal Barrier	$\leq$ 60
7. ECR-31	Containment Air Particle Radio Gas Detector	$\leq$ 10
8. ECR-32	Containment Air Particle Radio Gas Detector	$\leq$ 10
9. ECR-33	Containment Air Particle Radio Gas Detector	$\leq$ 10

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER		FUNCTION	ISOLATION TIME IN SECONDS
B. PHASE "B" ISOLATION (Continued)			
10.	WCR-901	NESW to Low. Containment Vent #1	≤ 10
11.	WCR-903	NESW from Low. Containment Vent #1	≤ 10
12.	WCR-905	NESW to Low. Containment Vent #2	≤ 10
13.	WCR-907	NESW from Low. Containment Vent #2	≤ 10
14.	WCR-909	NESW to Low. Containment Vent #3	≤ 10
15.	WCR-911	NESW from Low. Containment Vent #3	≤ 10
16.	WCR-913	NESW to Low. Containment Vent #4	≤ 10
17.	WCR-915	NESW from Low. Containment Vent #4	≤ 10
18.	WCR-921	NESW to Up. Containment Vent #1	≤ 10
19.	WCR-923	NESW from Up. Containment Vent #1	≤ 10
20.	WCR-925	NESW to Up. to Containment Vent #2	≤ 10
21.	WCR-927	NESW from Up. Containment Vent #2	≤ 10

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER		FUNCTION	ISOLATION TIME IN SECONDS
B.	PHASE "B" ISOLATION (Continued)		
22.	WCR-929	NESW to Up. Containment Vent #3	≤ 10
23.	WCR-931	NESW from Up. Containment Vent #3	≤ 10
24.	WCR-933	NESW to Up. Containment Vent #4	≤ 10
25.	WCR-935	NESW from Up. Containment Vent #4	≤ 10
26.	WCR-945	NESW from RCP Motor Air Cooler	≤ 10
27.	WCR-946	NESW from RCP Motor Air Cooler	≤ 10
28.	WCR-947	NESW from RCP Motor Air Cooler	≤ 10
29.	WCR-948	NESW from RCP Motor Air Cooler	≤ 10
30.	WCR-951	NESW to RCP Motor Air Cooler Vent #1	≤ 10
31.	WCR-952	NESW to RCP Motor Air Cooler Vent #2	≤ 10
32.	WCR-953	NESW to RCP Motor Air Cooler Vent #3	≤ 10
33.	WCR-954	NESW to RCP Motor Air Cooler Vent #4	≤ 10

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	ISOLATION TIME IN SECONDS
B. PHASE "B" ISOLATION (Continued)		
34. WCR-955	NESW from RCP Motor Air Cooler Vent #1	≤ 10
35. WCR-956	NESW from RCP Motor Air Cooler Vent #2	≤ 10
36. WCR-957	NESW from RCP Motor Air Cooler Vent #3	≤ 10
37. WCR-958	NESW from RCP Motor Air Cooler Vent #4	≤ 10
38. WCR-961	NESW to Instr. Rm. East Vent	≤ 10
39. WCR-963	NESW from Instr. Rm. West Vent	≤ 10
40. WCR-965	NESW to Instr. Rm. East Vent	≤ 10
41. WCR-967	NESW from Instr. Rm. West Vent	≤ 10
42. WCR-902	NESW from Lower Containment Vent #1	≤ 10
43. WCR-906	NESW from Lower Containment Vent #2	≤ 10
44. WCR-910	NESW from Lower Containment Vent #3	≤ 10
45. WCR-914	NESW from Lower Containment Vent #3	≤ 10

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER		FUNCTION	ISOLATION TIME IN SECONDS
B.	PHASE "B" ISOLATION (Continued)		
46.	WCR-922	NESW from Upper Containment Vent #1	$\leq$ 10
47.	WCR-926	NESW from Upper Containment Vent #2	$\leq$ 10
48.	WCR-930	NESW from Upper Containment Vent #3	$\leq$ 10
49.	WCR-934	NESW from Upper Containment Vent #4	$\leq$ 10
50.	WCR-962	NESW from Instrument Room East Vent	$\leq$ 10
51.	WCR-966	NESW from Instrument Room West Vent	$\leq$ 10
C.	CONTAINMENT PURGE AND EXHAUST		
1.	VCR-101	Instr. Room Purge Air Inlet	$\leq$ 10
2.	VCR-102	Instr. Room Purge Air Outlet	$\leq$ 10
3.	VCR-103	Lower Comp. Purge Air Inlet	$\leq$ 10
4.	VCR-104	Lower Comp. Purge Air Outlet	$\leq$ 10
5.	VCR-105	Upper Comp. Purge Air Inlet	$\leq$ 10

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	ISOLATION TIME IN SECONDS
C. CONTAINMENT PURGE AND EXHAUST (Continued)		
6. VCR-106	Upper Comp. Purge Air Outlet	$\leq$ 10
7. VCR-107*	Cont. Press. Relief Fan Isolation	$\leq$ 10
8. VCR-201	Instr. Room Purge Air Inlet	$\leq$ 10
9. VCR-202	Instr. Room Purge Air Outlet	$\leq$ 10
10. VCR-203	Lower Comp. Purge Air Inlet	$\leq$ 10
11. VCR-204	Lower Comp. Purge Air Outlet	$\leq$ 10
12. VCR-205	Upper Comp. Purge Air Outlet	$\leq$ 10
13. VCR-206	Upper Comp. Purge Air Outlet	$\leq$ 10
14. VCR-207*	Cont. Press Relief Fan Isolation	$\leq$ 10
D. MANUAL ISOLATION VALVES (1)		
1. 1CM-111#	RHR to RC Cold Legs	NA
2. 1CM-129	RHR Inlet to Pumps .	NA

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER		FUNCTION	ISOLATION TIME IN SECONDS
D. MANUAL ISOLATION VALVES[1] (Continued)			
3.	1CM-250	Boron Injection Inlet	NA
4.	1CM-251	Boron Injection Inlet	NA
5.	1CM-260	Safety Injection Inlet	NA
6.	1CM-265	Safety Injection Inlet	NA
7.	1CM-305	RHR Suction From Sump	NA
8.	1CM-306	RHR Suction From Sump	NA
9.	1CM-311#	RHR to RC Hot Legs	NA
10.	1CM-321#	RHR to RC Hot Legs	NA
E. OTHER			
1.	CS-442-1	Seal Wtr. to RCP #1	NA
2.	CS-442-2	Seal Wtr. to RCP #2	NA
3.	CS-442-3	Seal Wtr. to RCP #3	NA
4.	CS-442-4	Seal Wtr. to RCP #4	NA

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER		FUNCTION	ISOLATION TIME IN SECONDS
E.	OTHER (Continued)		
5.	S1-189	R. C. Relief Valve Vent Hole	NA
6.	NSW-415-1	NESW to Lower Cont. Vent #1	NA
7.	NSW-415-2	NESW to Lower Cont. Vent #2	NA
8.	NSW-415-3	NESW to Lower Cont. Vent #3	NA
9.	NSW-415-4	NESW to Lower Cont. Vent #4	NA
10.	NSW-419-1	NESW to Upper Cont. Vent #1	NA
11.	NSW-419-2	NESW to Upper Cont. Vent #2	NA
12.	NSW-419-3	NESW to Upper Cont. Vent #3	NA
13.	NSW-419-4	NESW to Upper Cont. Vent #4	NA
14.	NSW-244-1	NESW to RCP #1 Motor Air Cooler	NA
15.	NSW-244-2	NESW to RCP #2 Motor Air Cooler	NA
16.	NSW-244-3	NESW to RCP #3 Motor Air Cooler	NA
17.	NSW-244-4	NESW to RCP #4 Motor Air Cooler	NA

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CHANGE NO. 11

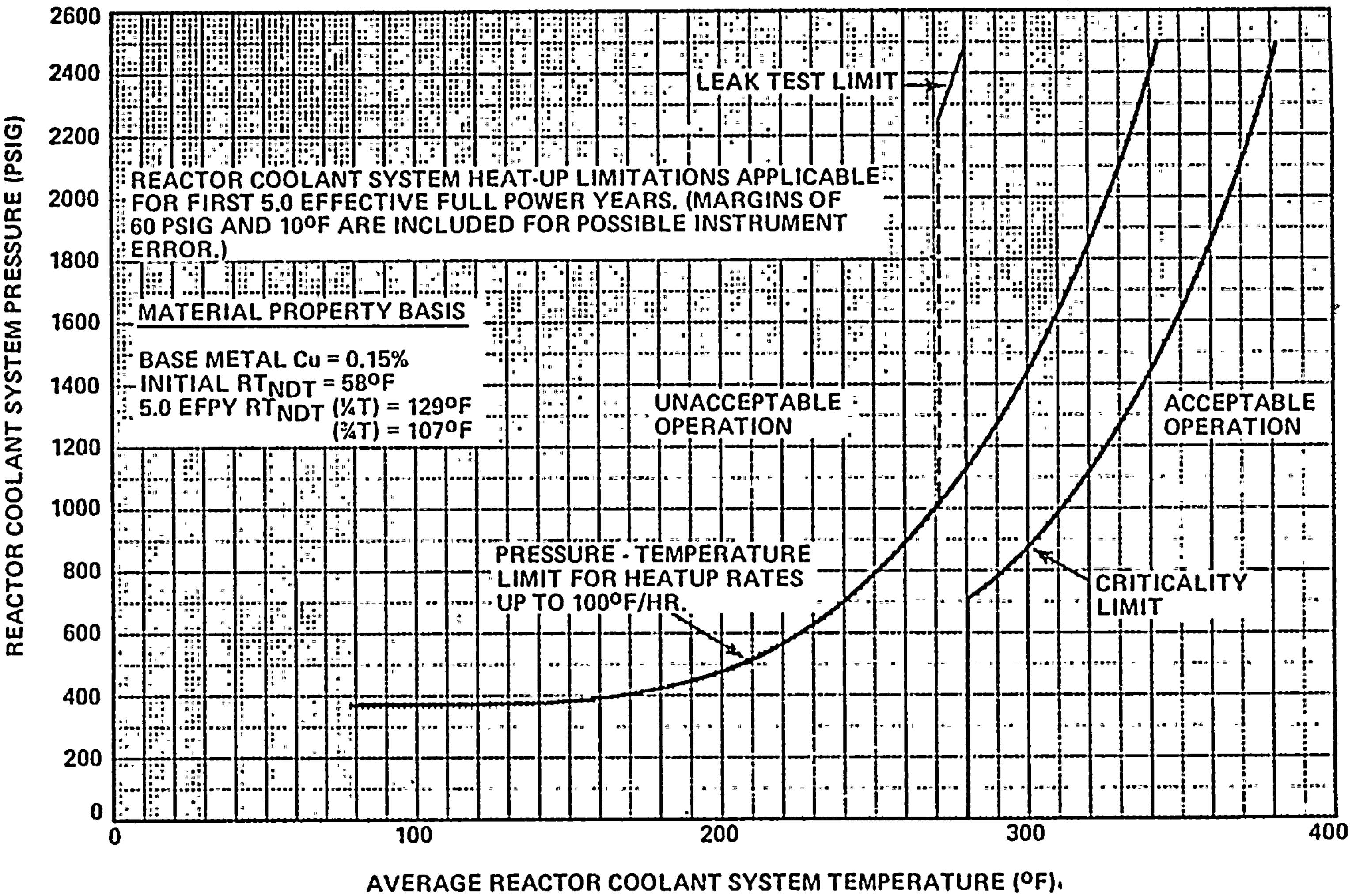

FIGURE 3.4-2

REACTOR COOLANT SYSTEM PRESSURE — TEMPERATURE LIMITS VERSUS 100°F/Hour Rate-
CRITICALITY LIMIT AND HYDROSTATIC TEST LIMIT

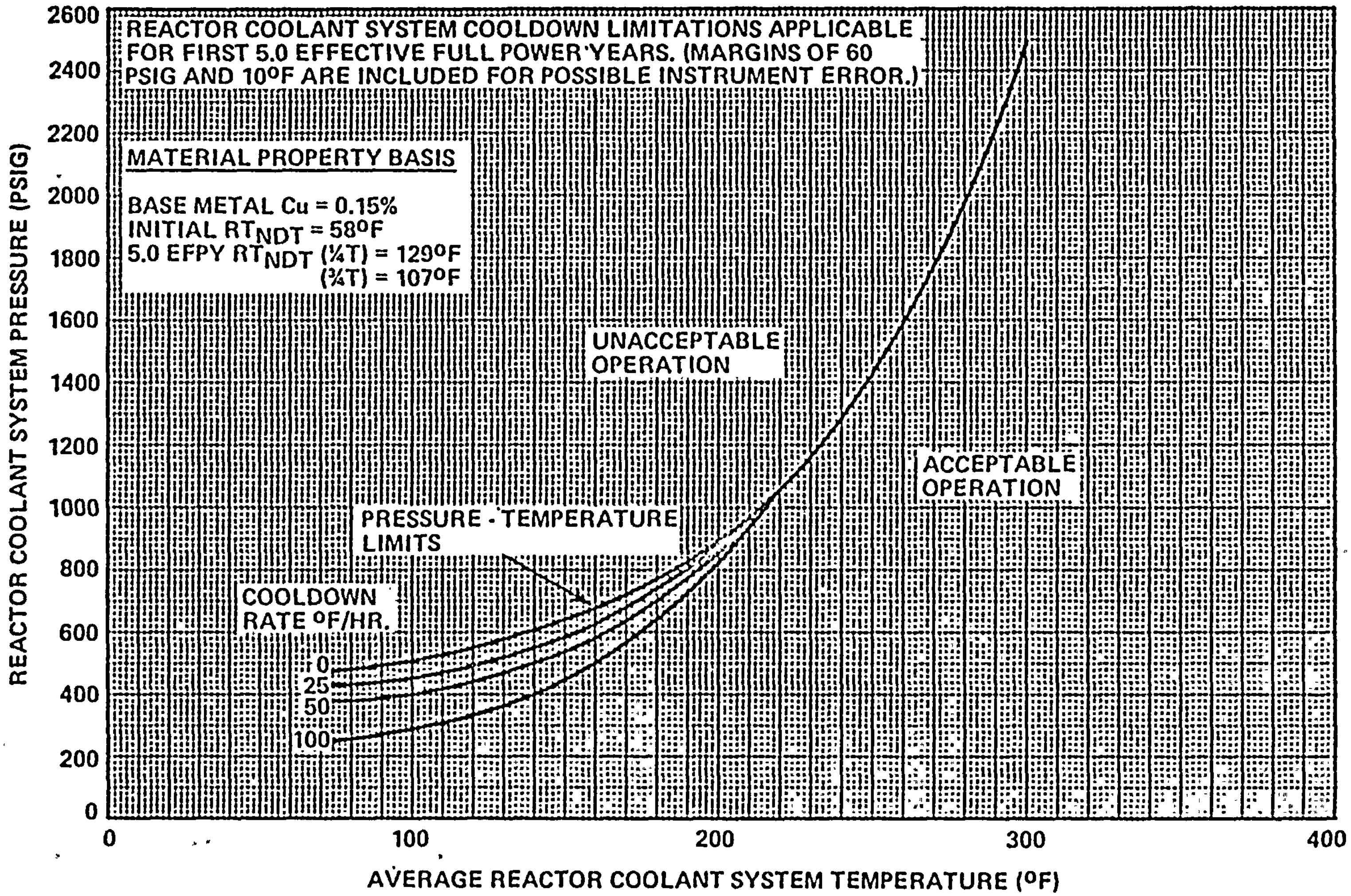

FIGURE 3.4-3

REACTOR COOLANT SYSTEM PRESSURE — TEMPERATURE LIMITS VERSUS COOLDOWN RATES

AEP:NRC:00111

CHANGE NO. 12

COMPOSITION

6.5.2.2 The NSDRC shall be composed of the:

Vice Chairman Engineering and Construction
Senior Executive Vice President Engineering
Senior Vice President Construction
Executive Vice President Indiana & Michigan Electric Company
Vice President Electrical Engineering
Vice President Mechanical Engineering
Assistant Vice President and Chief Civil Engineer
Chief Nuclear Engineer (Chairman)
Chief Design Engineer
Plant Manager, Donald C. Cook Plant
Head Environmental Engineering Division
Head, Nuclear Safety & Licensing Section (Secretary)
Alternate: Executive Assistant to the Vice Chairman Engineering & Construction
Alternate: Assistant Division Head, Project Control and Support Division
Alternate: Executive Assistant to the Executive Vice President I & M
Alternate: Assistant Chief Mechanical Engineer
Alternate: Assistant Chief Civil Engineer
Alternate: Assistant Division Head, Nuclear Engineering Division
Alternate: Head, Electrical Plant Design Section
Alternate: Assistant Plant Manager, Donald C. Cook Plant
Alternate: Senior Staff Engineer, Environmental Engineering Division
Alternate: Engineer, Nuclear Safety & Licensing Section
Alternate: AEPSC Manager of Quality Assurance
Alternate: Assistant Chief Electrical Engineer

ALTERNATES

6.5.2.3 All alternate members shall be appointed in writing by the NSDRC Chairman to serve on a temporary basis; however, no more than two alternates shall participate as voting members in NSDRC activities at any one time.

CONSULTANTS

6.5.2.4 Consultants shall be utilized as determined by the NSDRC Chairman to provide expert advice to the NSDRC.

MEETING FREQUENCY

6.5.2.5 The NSDRC shall meet at least once per calendar quarter during the initial year of facility operation following fuel loading and at least once per six months thereafter.

QUORUM

6.5.2.6 A quorum of NSDRC shall consist of the Chairman or his designated alternate and at least 4 NSDRC members including alternates. No more than a minority of the quorum shall have line responsibility for operation of the facility.

REVIEW

6.5.2.7 The NSDRC shall review:

a. The safety evaluations for 1) changes to procedures, equipment or systems and 2) tests or experiments completed under the provision of Section 50.59, 10 CFR, to verify that such actions did not constitute an unreviewed safety question.

b. Proposed changes to procedures, equipment or systems which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

c. Proposed tests or experiments which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

d. Proposed changes to Technical Specifications or this Operating License.

e. Violations of codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance.

f. Significant operating abnormalities or deviations from normal and expected performance of plant equipment that affect nuclear safety.

g. Events requiring 24 hour written notification to the Commission.

h. All recognized indications of an unanticipated deficiency in some aspect of design or operation of safety related structures, systems, or components.

i. Reports and meetings minutes of the PNSRC.

AUDITS

6.5.2.8 Audits of facility activities shall be performed under the cognizance of the NSDRC. These audits shall encompass:

a. The conformance of facility operation to provisions contained within the Technical Specifications and applicable license conditions at least once per 12 months.

b. The performance, training and qualifications of the entire facility staff at least once per 12 months.

c. The results of actions taken to correct deficiencies occurring in facility equipment, structures, systems or method of operation that affect nuclear safety at least once per 6 months.

d. The performance of activities required by the Operational Quality Assurance Program to meet the criteria of Appendix "B", 10 CFR 50, at least once per 24 months.

e. The Facility Emergency Plan and implementing procedures at least once per 24 months.

f. The Facility Security Plan and implementing procedures at least once per 24 months.

g. Any other area of facility operation considered appropriate by the NSDRC.

h. The Facility Fire Protection Program and implementing procedures at least once per 24 months.

i. An independent fire protection and loss prevention program inspection and audit shall be performed at least once per 12 months utilizing either qualified offsite licensee personnel or an outside fire protection firm.

j. An inspection and audit of the fire protection and loss prevention program shall be performed by a qualified outside fire consultant at least once per 36 months.

AUTHORITY

6.5.2.9 The NSDRC shall report to and advise the Vice Chairman, Engineering and Construction, AEPSC, on those areas of responsibility specified in Sections 6.5.2.7 and 6.5.2.8.

RECORDS

6.5.2.10 Records of NSDRC activities shall be prepared, approved and distributed as indicated below:

a. Minutes of each NSDRC meeting shall be prepared, approved and forwarded to the Vice Chairman, Engineering and Construction, AEPSC, within 14 days following each meeting.

b. Reports of reviews encompassed by Section 6.5.2.7 above, shall be prepared, approved and forwarded to the Vice Chairman, Engineering and Construction ,AEPSC within 14 days following completion of the review.

c. Audit reports encompassed by Section 6.5.2.8 above, shall be forwarded to the Vice Chairman, Engineering and Construction, AEPSC, and to the management positions responsible for the areas audited within 30 days after completion of the audit.

6.6 REPORTABLE OCCURRENCE ACTION

6.6.1 The following actions shall be taken for REPORTABLE OCCURRENCES:

a. The Commission shall be notified and/or a report submitted pursuant to the requirements of Specification 6.9.

b. Each REPORTABLE OCCURRENCE requiring 24 hour notification to the Commission shall be reviewed by the PNSRC and submitted to the NSDRC and the Chief Nuclear Engineer.

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CHANGE NO. 13

3/4.3.4 TURBINE OVERSPEED PROTECTION

LIMITING CONDITION FOR OPERATION

3.3.4.1 At least one turbine overspeed protection system shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one stop valve or one control valve per high pressure turbine steam lead inoperable or with one reheat stop valve or one reheat intercept valve per low pressure turbine steam lead inoperable, operation may continue for up to 72 hours provided the inoperable valve(s) is restored to OPERABLE status or at least one valve in the affected steam lead is closed; otherwise, isolate the turbine from the steam supply within the next 6 hours.

b. With the above required turbine overspeed protection system otherwise inoperable, within 6 hours either restore the system to OPERABLE status or isolate the turbine from the steam supply.

SURVEILLANCE REQUIREMENTS

4.3.4.1.1 The provisions of Specification 4.0.4 are not applicable.

4.3.4.1.2 The above required turbine overspeed protection system shall be demonstrated OPERABLE:

a. At least once per 7 days by cycling each of the following valves through at least one complete cycle from the running position.

   1. Four high pressure turbine stop valves.

   2. Four high pressure turbine control valves.

   3. Six low pressure turbine reheat stop valves.

   4. Six low pressure turbine reheat intercept valves.

LIMITING CONDITION FOR OPERATION

b. At least once per 31 days by direct observation of the movement of each of the above valves through one complete cycle from the running position.

c. At least once per 18 months by performance of a CHANNEL CALIBRATION on the turbine overspeed protection systems.

d. At least once per 40 months by disassembling at least one of each of the above valves and performing a visual and surface inspection of valve seats, disks and stems and verifying no unacceptable flaws or corrosion.

AEP:NRC:00111

CHANGE NO. 14

D.C. COOK - UNIT 2 2-9

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

Note 2: Overpower $\Delta T \leq \Delta T_o[K_4 - K_5 \left(\frac{\tau_3 S}{1+\tau_3 S}\right) T - K_6 (T - T") - f_2(\Delta I)]$

where:

ΔT_o	=	Indicated ΔT at rated power
T	=	Average temperature, °F
$T"$	=	Indicated T_{avg} at RATED THERMAL POWER $\leq$ 572.2°F
K_4	=	1.078
K_5	=	0.02/°F for increasing average temperature and 0 for decreasing average temperature
K_6	=	0.00197 for $T > T"$; $K_6 = 0$ for $T \leq T"$
$\frac{\tau_3 S}{1+\tau_3 S}$	=	The function generated by the rate lag controller for T_{avg} dynamic compensation
τ_3	=	Time constant utilized in the rate lag controller for T_{avg} τ_3 = 10 secs.
S	=	Laplace transform operator
$f_2(\Delta I)$	=	0 for all ΔI

Note 3: The channel's maximum trip point shall not exceed its computed trip point by more than 2 percent excluding transmitter.

AEP:NRC:00111

CHANGE NO. 15

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - T_{avg} > 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be ≥ 1.6% Δk/k.

APPLICABILITY: MODES 1, 2,* 3, and 4.

ACTION:

With the SHUTDOWN MARGIN < 1.6% Δk/k, immediately initiate and continue boration at ≥ 10 gpm of 20,000 ppm boric acid solution or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be ≥ 1.6% Δk/k:

a. Within one hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the above required SHUTDOWN MARGIN shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable control rod(s).

b. When in MODES 1 or 2#, at least once per 12 hours by verifying that control bank withdrawal is within the limits of Specification 3.1.3.5.

c. When in MODE 2##, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical control rod position is within the limits of Specification 3.1.3.5.

---

*See Special Test Exception 3.10.1

#With $K_{eff} \geq 1.0$

##With $K_{eff} < 1.0$

SURVEILLANCE REQUIREMENTS (Continued)

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of e below, with the control banks at the maximum insertion limit of Specification 3.1.3.6.

e. When in MODES 3 or 4, at least once per 24 hours by consideration of the following factors:

1. Reactor coolant system boron concentration,
2. Control rod position,
3. Reactor coolant system average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration, and
6. Samarium concentration.

4.1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within $\pm$ 1% $\Delta k/k$ at least once per 31 Effective Full Power Days (EFPD). This comparison shall consider at least those factors stated in Specification 4.1.1.1.1.e, above. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 Effective Full Power Days after each fuel loading.

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CHANGE NO. 16

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
4. STEAM LINE ISOLATION		
a. Manual	Not Applicable	Not Applicable
b. Automatic Actuation Logic	Not Applicable	Not Applicable
c. Containment Pressure--High-High	$\leq$ 2.9 psig	$\leq$ 3.0 psig
d. Steam Flow in Two Steam lines--High Coincident with T_{avg} --Low-Low	$\leq$ A function defined as follows: A Δp corresponding to 1.47×10^6 lbs/hr steam flow between 0% and 20% load and then a Δp increasing linearly to a Δp corresponding to 110% of full steam flow at full load.	$\leq$ A function defined as follows: A Δp corresponding to 1.62×10^6 lbs/hr steam flow between 0% and 20% load and then a Δp increasing linearly to a Δp corresponding to 111.5% of full steam flow at full load.
	$T_{avg} \geq 541°F$	$T_{avg} \geq 539°F$
e. Steam Line Pressure--Low	$\geq$ 600 psig steam line pressure	$\geq$ 580 psig steam line pressure
5. TURBINE TRIP AND FEED WATER ISOLATION		
a. Steam Generator Water level--High-High	$\leq$ 67% of narrow range instrument span each steam generator	$\leq$ 68% of narrow range instrument span each steam generator

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CHANGE NO. 17

CONTAINMENT SYSTEMS

DIVIDER BARRIER PERSONNEL ACCESS DOORS AND EQUIPMENT HATCHES

LIMTING CONDITION FOR OPERATION

3.6.5.5 The personnel access doors and equipment hatches between the containment's upper and lower compartments shall be OPERABLE and closed.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With a personnel access door or equipment hatch inoperable or open except for personnel transit entry and $T_{avg} > 200°F$, restore the door or hatch to OPERABLE status or to its closed position (as applicable) within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.5.5.1 The personnel access doors and equipment hatches between the containment's upper and lower compartments shall be determined closed by a visual inspection prior to increasing the Reactor Coolant System T_{avg} above 200°F and after each personnel transit entry when the Reactor Coolant System T_{avg} is above 200°F.

4.6.5.5.2 The personnel access doors and equipment hatches between the containment's upper and lower compartments shall be determined OPERABLE by visually inspecting the seals and sealing surfaces of these penetrations and verifying no detrimental misalignments, cracks or defects in the sealing surfaces, or apparent deterioration of the seal material:

a. Prior to final closure of the penetration each time it has been opened, and

b. At least once per 10 years for penetrations containing seals fabricated from resilient materials.

CONTAINMENT AIR RECIRCULATION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.6.5.6 Two independent containment air recirculation systems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one containment air recirculation system inoperable, restore the inoperable system to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.5.6 Each containment air recirculation system shall be demonstrated OPERABLE at least once per 92 days on a STAGGERED TEST BASIS by:

a. Verifying that the return air fan starts on an auto-start signal after a 9 $\pm$ 1 minute delay and operate for at least 15 minutes.

b. Verifying that with the return air fan dampers closed, the fan motor current is 56 $\pm$ 5 amps when the fan speed is 880 $\pm$ 20 RPM.

c. Verifying that with the fan off, the return air fan damper opens when a force of $\leq$ 11 lbs is applied to the counterweight.

d. Verifying that the motor operated valve in the suction line to the containment's lower compartment opens after a 9 $\pm$ 1 minute delay.

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CHANGE NO. 18

6.10 RECORD RETENTION

6.10.1 The following records shall be retained for at least five years:

a. Records and logs of facility operation covering time interval at each power level.

b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety.

c. ALL REPORTABLE OCCURRENCES submitted to the Commission.

d. Records of surveillance activities, inspections and calibrations required by these Technical Specifications.

e. Records of changes made to Operating Procedures.

f. Records of radioactive shipments.

g. Records of sealed source and fission detector leak tests and results.

h. Records of annual physical inventory of all sealed source material of record.

6.10.2 The following records shall be retained for the duration of the Facility Operating License:

a. Records and drawing changes reflecting facility design modifications made to systems and equipment described in the Final Safety Analysis Report.

b. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories.

c. Record of radiation exposure for all individuals entering radiation control areas.

[illegible]

d. Records of gaseous and liquid radioactive material released to the environs.

e. Records of transient of operational cycles for those facility components identified in Table 5.7-1.

f. Records of reactor tests and experiments.

g. Records of training and qualification for current members of the plant staff.

h. Records of in-service inspections performed pursuant to these Technical Specifications.

i. Records of Quality Assurance activities required by the QA Manual.

j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.

k. Records of meetings of the PNSRC and the NSDRC.

6.11 RADIATION PROTECTION PROGRAM

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

6.12 HIGH RADIATION AREA

6.12.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.203(c)(2) of 10 CFR 20, each high radiation area in which the intensity of radiation is 1000 mrem/hr or less shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit*. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

*Health Physics personnel shall be exempt from the RWP issuance requirement during the performance of their assigned radiation protection duties, provided they comply with approved radiation protection procedures for entry into high radiation areas.

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CHANGE NO. 19

DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB related parameters shall be maintained within the limits shown on Table 3.2-1:

a. Reactor Coolant System T_{avg}.

b. Pressurizer Pressure.

APPLICABILITY: MODE 1

ACTION:

With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.5 Each of the parameters of Table 3.2-1 shall be verified to be within their limits at least once per 12 hours.

TABLE 3.2-1

DNB PARAMETERS

PARAMETER	LIMITS 4 Loops In Operation	LIMITS 3 Loops In Operation
Reactor Coolant System T_{avg}	$\leq$ 576.2°F	$\leq$ 569.8°F
Pressurizer Pressure	$\geq$ 2220 psia*	$\geq$ 2220 psia*

"Limit not applicable during either THERMAL POWER ramp changes or THERMAL POWER step changes in excess of 10% RATED THERMAL POWER."

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CHANGE NO. 20

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 REACTOR COOLANT LOOPS

NORMAL OPERATION

LIMITING CONDITION FOR OPERATION

3.4.1.1 All reactor coolant loops shall be in operation.

APPLICABILITY: As noted below, but excluding MODE 6.*

ACTION:

Above P-7, comply with either of the following ACTIONS:

a. With one reactor coolant loop and associated pump not in operation, STARTUP and/or continued POWER OPERATION may proceed provided THERMAL POWER is restricted to less than 31% of RATED THERMAL POWER and the following ESF instrumentation channels associated with the loop not in operation, are placed in their tripped condition within 1 hour:

   1. T_{avg} -- Low-Low channel used in the coincidence circuit with Steam Flow - High for Safety Injection.

   2. Steam Line Pressure - Low channel used in the coincidence circuit with Steam Flow - High for Safety Injection.

   3. Steam Flow-High Channel used for Safety Injection.

   4. Differential Pressure Between Steam Lines - High channel used for Safety Injection (trip all bistables which indicate low active loop steam pressure with respect to the idle loop steam pressure).

b. With one reactor coolant loop and associated pump not in operation, subsequent STARTUP and POWER OPERATION above 31% of RATED THERMAL POWER may proceed provided:

   1. The following actions have been completed with the reactor in at least HOT STANDBY:

      a) Reduce the overtemperature ΔT trip setpoint to the value specified in Specification 2.2.1 for 3 loop operation.

*See Special Test Exception 3.10.4.

ACTION (Continued)

b) Place the following reactor trip system and ESF instrumentation channels, associated with the loop not in operation, in their tripped conditions:

1) Overpower ΔT channel.

2) Overtemperature ΔT channel.

3) T_{avg} -- Low-Low channel used in the coincidence circuit with Steam Flow - High for Safety Injection.

4) Steam Line Pressure - Low channel used in the coincidence circuit with Steam Flow - High for Safety Injection.

5) Steam Flow-High channel used for Safety Injection.

6) Differential Pressure Between Steam Lines - High channel used for Safety Injection (trip all bistables which indicate low active loop steam pressure with respect to the idle loop steam pressure).

c) Change the P-8 interlock setpoint from the value specified in Table 3.3-1 to $\leq$ 76% of RATED THERMAL POWER.

2. THERMAL POWER is restricted to $\leq$ 71% of RATED THERMAL POWER.

Below P-7:

a. With $K_{eff} \geq 1.0$, operation may proceed provided at least two reactor coolant loops and associated pumps are in operation.

b. With $K_{eff} < 1.0$, operation may proceed provided at least one reactor coolant loop is in operation with an associated reactor coolant or residual heat removal pump.*

c. The provisions of Specification 3.0.3 and 3.0.4 are not applicable.

---

* All reactor coolant pumps and residual heat removal pumps may be de-energized for up to 1 hour, provided no operations are permitted which could cause dilution of the reactor coolant system boron concentration.

3/4.4 REACTOR COOLANT SYSTEM

BASES

3/4.4.1 REACTOR COOLANT LOOPS

The plant is designed to operate with all reactor coolant loops in operation, and maintain calculated DNBR above the design DNBR value during Condition I and II events. With one reactor coolant loop not in operation, THERMAL POWER is restricted to < 51 percent of RATED THERMAL POWER until the Overtemperature ΔT trip is reset. Either action ensures that the calculated DNBR will be maintained above the design DNBR value. A loss of flow in two loops will cause a reactor trip if operating above P-7 (11 percent of RATED THERMAL POWER) while a loss of flow in one loop will cause a reactor trip if operating above P-8 (31 percent of RATED THERMAL POWER).

A single reactor coolant loop provides sufficient heat removal capability for removing core decay heat while in HOT STANDBY; however, single failure considerations require placing a RHR loop into operation in the shutdown cooling mode if component repairs and/or corrective actions cannot be made within the allowable out-of-service time.

3/4.4.2 and 3/4.4.3 SAFETY VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. Each safety valve is designed to relieve 420,000 lbs per hour of saturated steam at the valve set point. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating RHR loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization.

During operation, all pressurizer code safety valves must be OPERABLE to prevent the RCS from being pressurized above its safety limit of 2735 psig. The combined relief capacity of all of these valves is greater than the maximum surge rate resulting from a complete loss of load assuming no reactor trip until the first Reactor Protective System trip set point is reached (i.e., no credit is taken for a direct reactor trip on the loss of load) and also assuming no operation of the power operated relief valves or steam dump valves.

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.

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 3.3-1 (Continued)

ACTION 8 - With the number of OPERABLE channels one less than the Total Numbers of Channels and with the THERMAL POWER level above P-7, place the inoperable channel in the tripped condition within 1 hour; operation may continue until performance of the next required CHANNEL FUNCTIONAL TEST.

ACTION 9 - With a channel associated with an operating loop inoperable, restore the inoperable channel to OPERABLE status within 2 hours or be in HOT STANDBY within the next 6 hours; however, one channel associated with an operating loop may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1.1.

ACTION 10 - With one channel inoperable, restore the inoperable channel to OPERABLE status within 2 hours or reduce THERMAL POWER to below P-8 within the next 2 hours. Operation below P-8 may continue pursuant to ACTION 11.

ACTION 11 - With less than the Minimum Number of Channels OPERABLE, operation may continue provided the inoperable channel is placed in the tripped condition within 1 hour.

ACTION 12 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in HOT STANDBY within the next 6 hours and/or open the reactor trip breakers.

REACTOR TRIP SYSTEM INTERLOCKS

DESIGNATION	CONDITION AND SETPOINT	FUNCTION
P-6	With 2 of 2 Intermediate Range Neutron Flux Channels < 6×10^{-11} amps.	P-6 prevents or defeats the manual block of source range reactor trip.

TABLE 3.3-1 (Continued)

DESIGNATION	CONDITION AND SETPOINT	FUNCTION
P-7	With 2 of 4 Power Range Neutron Flux Channels $\geq$ 11% of RATED THERMAL POWER or 1 of 2 Turbine impulse chamber pressure channels $\geq$ 55 psia.	P-7 prevents or defeats the automatic block of reactor trip on: Low flow in more than one primary coolant loop, reactor coolant pump under-voltage and under-frequency, turbine trip, pressurizer low pressure, and pressurizer high level.
P-8	With 2 of 4 Power Range Neutron Flux channels $\geq$ 31% of RATED THERMAL POWER.	P-8 prevents or defeats the automatic block of reactor trip on low coolant flow in a single loop.
P-10	With 3 of 4 Power range neutron flux channels < 9% of RATED THERMAL POWER.	P-10 prevents or defeats the manual block of: Power range low setpoint reactor trip, Intermediate range reactor trip, and intermediate range rod stops. Provides input to P-7.

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CHANGE NO. 21

SURVEILLANCE REQUIREMENTS

4.2.6.1 $F_j(Z)$ shall be determined to be within its limit by:

a. Either using the APDMS to monitor the thimbles required per Specification 3.3.3.7 at the following frequencies.

1. At least once per 8 hours, and

2. Immediately and at intervals of 10, 30, 60, 90, 120, 240 and 480 minutes following:

a) Increasing the THERMAL POWER above 94% of RATED THERMAL POWER, or

b) Movement of control bank "D" more than an accumulated total of 5 steps in any one direction.

b. Or using the movable incore detectors at the following frequencies when the APDMS is inoperable:

1. At least once per 8 hours, and

2. At intervals of 30, 60, 90, 120, 240 and 480 minutes following:

a) Increasing the THERMAL POWER above 94% of RATED THERMAL POWER, or

b) Movement of control bank "D" more than an accumulated total of 5 steps in any one direction.

4.2.6.2 When the movable incore detectors are used to monitor $F_j(Z)$, at least 2 thimbles shall be monitored and an $F_j(Z)$ accuracy equivalent to that obtained from the APDMS shall be maintained.

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CHANGE NO. [illegible]3

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
9. Pressurizer Pressure-Low	4	2	3	1, 2	6#
10. Pressurizer Pressure--High	4	2	3	1, 2	6#
11. Pressurizer Water Level--High	3	2	2	1, 2	7#
12. Loss of Flow - Single Loop (Above P-8)	3/loop	2/loop in any operating loop	2/loop in each operating loop	1	7#
13. Loss of Flow - Two Loops (Above P-7 and below P-8)	3/loop	2/loop in two operating loops	2/loop each operating loop	1	7#
14. Steam Generator Water Level--Low-Low	3/loop	2/loop in any operating loop	2/loop each operating loop	1, 2	7#
15. Steam/Feedwater Flow Mismatch and Low Steam Generator Water	2/loop-level and 2/loop-flow mismatch in same loop	1/loop-level coincident with 1/loop-flow mismatch in same loop	1/loop-level and 2/loop-flow mismatch or 2/loop-level and 1/loop-flow mismatch	1, 2	7#

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
16. Undervoltage-Reactor Coolant Pumps	4-1/bus	2	3	1	6#
17. Underfrequency-Reactor Coolant Pumps	4-1/bus	2	3	1	6#
18. Turbine Trip					
A. Low Fluid Oil Pressure	3	2	2	1	7#
B. Turbine Stop Valve Closure	4	4	3	1	6#
19. Safety Injection Input from ESF	2	1	2	1, 2	1
20. Reactor Coolant Pump Breaker Position Trip					
A. Above P-8	1/breaker	1	1/breaker	1	10
B. Above P-7	1/breaker	2	1/breaker per operating loop	1	11
21. Reactor Trip Breakers	2	1	2	1, 2 and *	1
22. Automatic Trip Logic	2	1	2	1, 2 and *	1

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CHANGE NO. 23

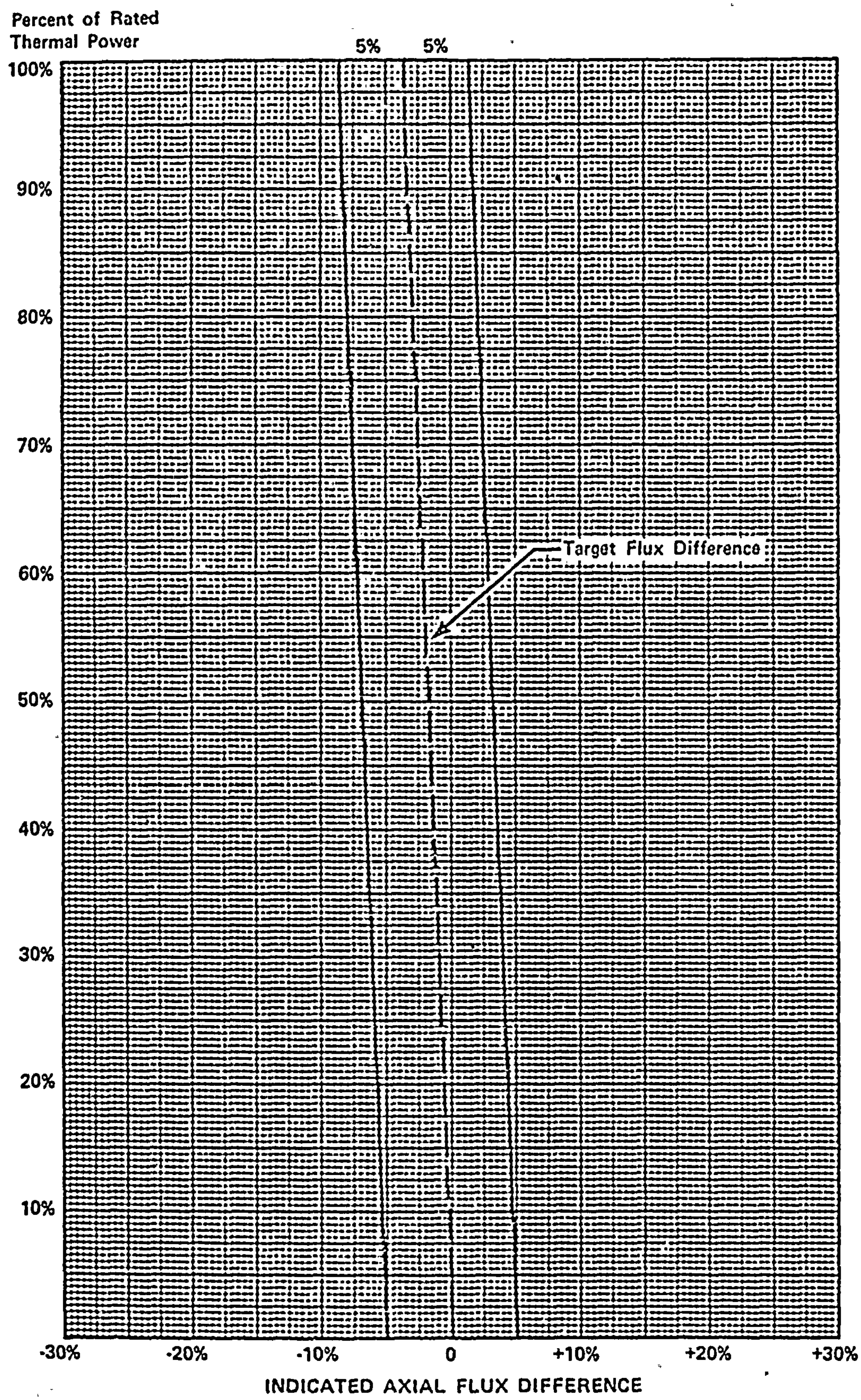

Figure B 3/4 2-1 TYPICAL INDICATED AXIAL FLUX DIFFERENCE VERSUS THERMAL POWER

3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR, RCS FLOWRATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

The limits on heat flux hot channel factor, RCS flowrate, and nuclear entholpy rise hot channel factor ensure that 1) the design limits on peak local power density and minimum DNBR are not exceeded and 2) in the event of a LOCA the peak fuel clad temperature will not exceed the 2200°F ECCS acceptance criteria limit.

Each of these is measurable but will normally only be determined periodically as specified in Specifications 4.2.2 and 4.2.3. This periodic surveillance is sufficient to insure that the limits are maintained provided:

a. Control rodsin a single group move together with no individual rod insertion differing by more than ± 12 steps from the group demand position.

b. Control rod groups are sequenced with overlapping groups as described in Specification 3.1.3.6.

c. The control rod insertion limits of Specifications 3.1.3.5 and 3.1.3.6 are maintained.

d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE, is maintained within the limits.

$F^N_{\Delta H}$ will be maintained within its limits provided conditions a. through d. above are maintained. As noted on Figures 3.2-3 and 3.2-4, RCS flow rate and $F^N_{\Delta H}$ may be "traded off" against one another (i.e., a low measured RCS flow rate is acceptable if the measured $F^N_{\Delta H}$ is also low) to ensure that the calculated DNBR will not be below the design DNBR value. The relaxation of $F^N_{\Delta H}$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits.

When an F_Q measurement is taken, both experimental error and manufacturing tolerance must be allowed for. 5% is the appropriate allowance for a full core map taken with the incore detector flux mapping system and 3% is the appropriate allowance for manufacturing tolerance.

When RCS flow rate and $F^N_{\Delta H}$ are measured, no additional allowances are necessary prior to comparison with the limits of Figures 3.2-3 and 3.2-4. Measurement errors of 3.5% for RCS total flow rate and 4% for $F^N_{\Delta H}$ have been allowed for in determination of the design DNBR value.

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CHANGE NO. 24

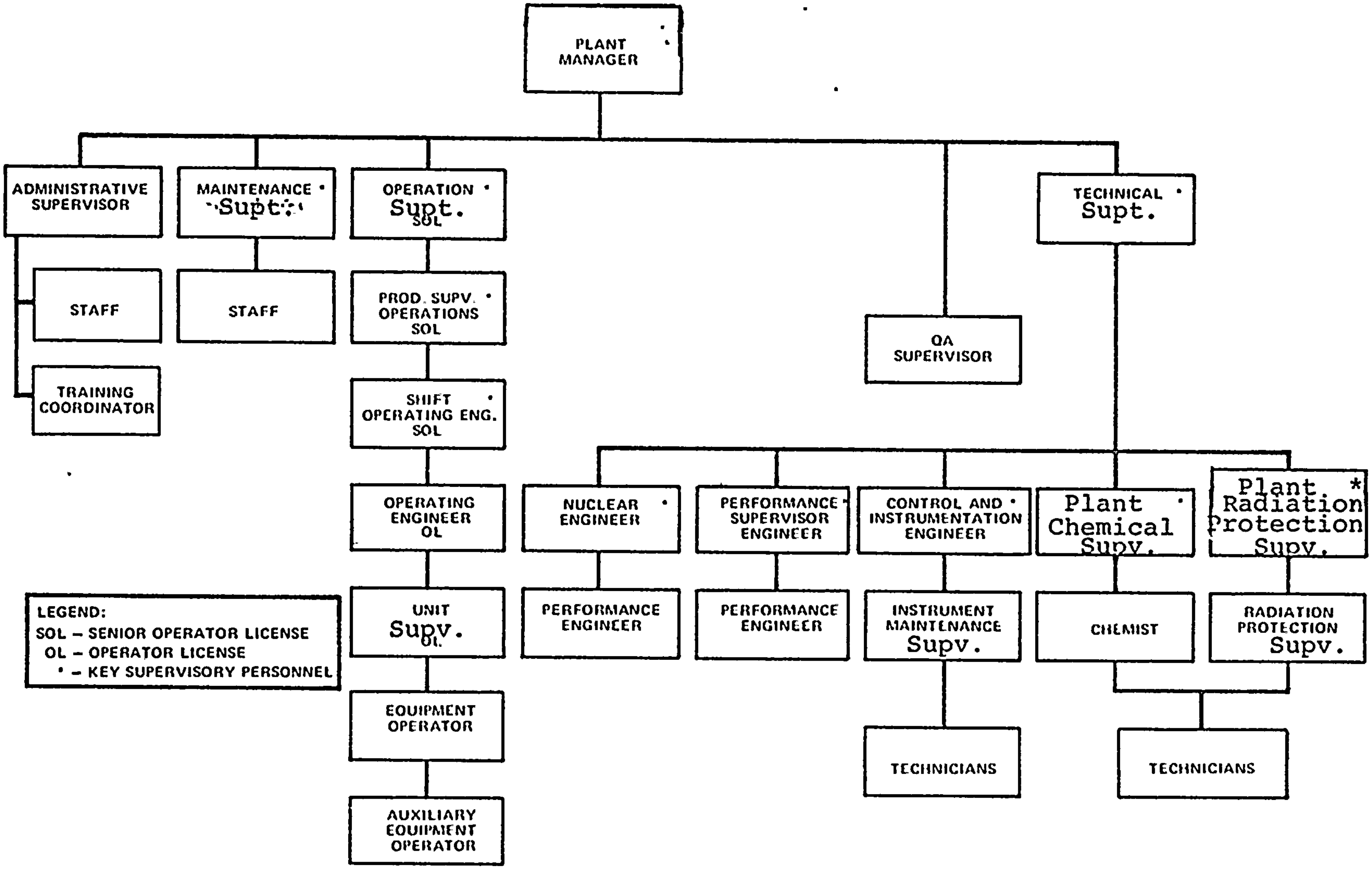

FIGURE 6.2-2 Facility Organization – Donald C. Cook – Unit No. 2

TABLE 6.2-1

MINIMUM SHIFT CREW COMPOSITION#

LICENSE CATEGORY	APPLICABLE MODES	
	1, 2, 3 & 4	5 & 6
SOL	2**	1*
OL	2	1
Non-Licensed	2	1

*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