ATTACHMENT 'A' TO AEP:NRC:00109

PROPOSED REVISIONS TO THE
DONALD C. COOK NUCLEAR PLANT UNIT NO. 1
APPENDIX 'A' TECHNICAL SPECIFICATIONS

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CHANGE NO. 1

Revision to Table 3.3-10; "Fire Detection Instrumentation"

This change involves a revision to Table 3.3-10 entitled, "Fire Detection Instrumentation" on page 3/4 3-52. The minimum number of thermistor detectors specified for the Containment Quadrants 1, 2, 3 and 4 do 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.f. 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 6-13; Unit 1)

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 6-13 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

CHANGE NO. 4 (CONT'D)

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

This change involves a revision to 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. 6

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 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. 7

This change involves a revision to Surveillance Requirement 4.3.3.6.1. 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 3-49. 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. 8

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 6-18. Also note that in order to have the proper numbering sequences, we have revised the valve numbering on page 3/4 6-18 thru 6-20 This change will not adversely affect the health and safety of the public.

CHANGE NO. 9 (EDITORIAL)

This change involves a revision to Table 2.2 -1 on page 2-9. Note 3 indicates "4 percent" and this is an editorial error. The attached revised page 2-9 has corrected this editorial error to read "2 percent." This change will not adversely affect the health and safety of the public.

CHANGE NO. 10 (EDITORIAL)

This change involves a revision to Table 3.3-1 on page 3/4 3-4. The words "same loop" should be added to Item 15, under Total No. of Channels as shown in the attached revised page 3/4 3-4. This change will not adversely affect the health and safety of the public.

CHANGE NO. 11 (EDITORIAL)

This change involves a revision to Bases section 3/4 2.2 on page B3/4 2-4. In paragraph a. the word "rod" should be changed to "rods" as shown in the attached revised page B3/4 2-4. This change is editorial. This change will not adversely affect the health and safety of the public.

CHANGE NO. 12

This change involves a revision to Figure 6.2-2 "Facility Organization" on page 6-3. We have had a system wide change in the titles of the staff members of our operating plant. Certain "Foreman" are now "Supervisors" and "Supervisors" are now "Superintendents." This change affects the Technical Specifications for the Cook Nuclear Plant as shown in the attached revised pages 6-3 and 6-4.

CHANGE NO. 13

This change revised 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 indicate 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 effect the health and safety of the public.

CHANGE NO. 14

CONTAINMENT AIR RECIRCULATION SYSTEMS

This change revised Technical Specifications 4.6.5.6(a) and (d) on page 3/4 6-35. 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 $\pm$ 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. Page 3/4 3-29 has also been revised to indicate the "new" maximum fan delay time. The above changes will not adversely effect the Health and Safety of the public.

CHANGE NO. 15

This change deletes specification 6.10.2.C on page 6-20. 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 effect teh health and safety of the public.

CHANGE NO. 16

This change revised the footnote to Table 3.2-1. The work 'increase' has been replaced with the word 'change' in two locations. 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 effect the health and safety of the public.

ATTACHMENT 'B' TO AEP:NRC:00109

PROPOSED REVISIONS TO THE
DONALD C. COOK NUCLEAR PLANT UNIT NO. 1
APPENDIX 'A' TECHNICAL SPECIFICATIONS

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CHANGE NO. 1

FIRE DETECTION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.7 As a minimum, the fire detection instrumentation for each fire detection zone shown in Table 3.3-10 shall be OPERABLE.

APPLICABILITY: Whenever equipment in that fire detection zone is required to be OPERABLE.

ACTION:

With the number of OPERABLE fire detection instruments less than required by Table 3.3-10:

1. Within 1 hour, establish a fire watch patrol to inspect the zone(s) with the inoperable instrument(s) at least once per hour, and

2. Restore the inoperable instrument(s) to OPERABLE status within 14 days or, in lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the instrument(s) to OPERABLE status.

3. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.7.1 Each of the above fire detection instruments shall be demonstrated OPERABLE at least once per 6 months by performance of a CHANNEL FUNCTIONAL TEST.

4.3.3.7.2 The NFPA Code 72D Class B supervised circuits supervision associated with the detector alarms of each of the above required fire detection instruments shall be demonstrated OPERABLE at least once per 6 months.

TABLE 3.3-10

FIRE DETECTION INSTRUMENTATION

INSTRUMENT LOCATION	MINIMUM INSTRUMENTS OPERABLE SMOKE (IONIZATION)	HEAT (THERMISTOR)
1. Containment		
Zone 6, Quadrant 1 Cable Tunnel	3	
Zone 7, Quadrant 2 Cable Tunnel	5	
Zone 8, Quadrant 3N Cable Tunnel	3	
Zone 9, Quadrant 3M Cable Tunnel	3	
Zone 10, Quadrant 3S Cable Tunnel	2	
Zone 11, Quadrant 4 Cable Tunnel	5	
Quadrant 1		14
Quadrant 2		3
Quadrant 3		18
Quadrant 4		11
1-HV-CFT-1 Charcoal Filters		1
1-HV-CFT-2 Charcoal Filters		1
2. Control Room		
Zone 22, Control Room	8	
3. Cable Spreading Room		
Zone 15, Switchgear Cable Vault	10	
Zone 16, Auxiliary Cable Vault	5	
Zone 17, Control Room Cable Vault	24	
Zone 18, Control Room Cable Vault	25	
4. Diesel Generator		
Diesel Generator Room 1AB		1
Diesel Generator Room 1CD		1
5. Diesel Fuel Oil Room		1

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CHANGE NO. 2

SURVEILLANCE REQUIREMENTS (Continued)

f. 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 $\geq$ 300 gpm
Loop 2 Boron Injection Flow 117.5 gpm	Loop 2 and 3 Cold Leg Flow $\geq$ 300 gpm
Loop 3 Boron Injection Flow 117.5 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 miniflow 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.

** Total SIS (single pump) flow, including miniflow, shall not exceed 650 gpm.

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CHANGE NO. 3

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 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 the all volatile treatment (AVT) of secondary coolant. However, even if a defect of similar type should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging will be required of all tubes with imperfections exceeding the plugging limit which, by the definition of Specification 4.4.5.4.a is 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 limitation 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.

The CONTROLLED LEAKAGE limitation restricts operation when the total flow supplied to the reactor coolant pump seals exceeds 52 GPM. 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 not isolated frm the RCS 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 as 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. Should PRESSURE BOUNDARY LEAKAGE occur through a component which can be isolated from the balance of the Reactor Coolant System, plant operation may continue provided the leaking component is promptly isolated from the Reactor Coolant System since isolation removes the source of potential failure.

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 the 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.

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CHANGE NO. 4

SURVEILLANCE REQUIREMENTS (Continued)

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

   1. Cycling each power operated (excluding automatic) valve in the flow path that is not testable during plant operation, through at least one complete cycle of full travel.

   2. Verifying that each automatic valve in the flow path actuates to its correct position on a Containment Pressure -- High-High signal.

d. At least once per 5 years by verifying a water flow rate of $\geq$ 20gpm and $\leq$ 50gpm from the spray additive tank to each containment spray system with the spray pump operating in the recirculation mode with a pump discharge pressure $\geq$ 225 psig.

3/4.6.3 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.3.1 The containment isolation valves specified in Table 3.6-1 shall be OPERABLE with isolation times as shown in Table 3.6-1.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one or more of the isolation valve(s) specified in Table 3.6-1 inoperable, either:

a. Restore the inoperable valve(s) to OPERABLE status within 4 hours, or

b. Isolate each affected penetration within 4 hours by use of at least one deactivated automatic valve secured in the isolation position, or

c. Isolate each affected penetration within 4 hours by use of at least one closed manual valve or blind flange; or

d. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.3.1.1 The isolation valves specified in Table 3.6-1 shall be demonstrated OPERABLE:

a. At least once per 92 days by cycling each OPERABLE power operated or automatic valve testable during plant operation through at least one complete cycle of full travel.

b. Immediately prior to returning the valve to service after maintenance, repair or replacement work is performed on the

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CHANGE NO. 5

COOLANT CIRCULATION

LIMITING CONDITION FOR OPERATION

3.9.8 At least one residual heat removal loop shall be in operation.

APPLICABILITY: MODE 6.

ACTION:

a. With less than one residual heat removal loop in operation, except as provided in b. below, suspend all operations involving an increase in the reactor decay heat load or a reduction in boron concentration of the Reactor Coolant System. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

b. The residual heat removal loop may be removed from operation for up to 1 hour per 8 hour period during the performance of CORE ALTERATIONS in the vicinity of the reactor pressure vessel hot legs.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.8 A residual heat removal loop shall be determined to be in operation and circulating reactor coolant at a flow rate of $\geq$ 3000 gpm at least once per 24 hours.

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 signal from each of the containment radiation monitoring instrumentation channels.

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CHANGE NO. 6

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 disintegration (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.

TABLE 1.1

OPERATIONAL MODES

MODE	REACTIVITY CONDITION, K_{eff}	% RATED THERMAL POWER*	AVERAGE COOLANT TEMPERATURE
1. POWER OPERATION	$\geq$ 0.99	> 5%	$\geq$ 350°F
2. STARTUP	$\geq$ 0.99	$\leq$ 5%	$\geq$ 350°F
3. HOT STANDBY	< 0.99	0	$\geq$ 350°F
4. HOT SHUTDOWN	< 0.99	0	350°F > T_{avg} > 200°F
5. COLD SHUTDOWN	< 0.99	0	$\leq$ 200°F
6. REFUELING**	$\leq$ 0.95	0	$\leq$ 140°F

* Excluding decay heat.

** Reactor vessel head unbolted or removed and fuel in the vessel.

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CHANGE NO. 7

AXIAL POWER DISTRIBUTION MONITORING SYSTEM

LIMITING CONDITION FOR OPERATION

3.3.3.6 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

4.3.3.6.1 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 orignial 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: 1) when incore maps are being taken as part of the Augmented Startup Test Program, or 2) when surveillance for determining power distribution maps is being performed.

SURVEILLANCE REQUIREMENTS (Continued)

a. If the absolute value of $\frac{R_{ij}-\overline{R}_j}{\overline{R}_j}$ is greater than $2\sigma_j$, another map shall be completed to verify the new $\overline{R}_j$. If the second map shows the first to be in error, the first map shall be disregarded. If the second map confirms the new $\overline{R}_j$, four more maps (including rodded configurations allowed by the insertion limits) will be completed so that a new $\overline{R}_j$ and σ_j can be defined from the six new maps.

4.3.3.6.2 The APDMS shall be demonstrated OPERABLE:

a. By performance of a CHANNEL FUNCTIONAL TEST within 7 days prior to its use and at least once per 31 days thereafter when used for monitoring $F_j(Z)$.

b. At least once per 18 months, during shutdown or below 5% of RATED THERMAL POWER, by performance of a CHANNEL CALIBRATION.

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CHANGE NO. 8

TABLE 3.6-1 (Continued)

VALVE NUMBER		FUNCTION	TESTABLE DURING PLANT OPERATION	ISOLATION TIME IN SECONDS
A.	PHASE "A" ISOLATION (Continued)			
27.	ECR-10	Cont. H_2 Sample Return	Yes	10
28.	ECR-11	Cont. H_2 Sample - Air to Rec. E	Yes	10
29.	ECR-12	Cont. H Sample - Air from Rec. E	Yes	10
30.	ECR-13	Cont. H_2 Sample - Low. Cont. Vol.	Yes	10
31.	ECR-14	Cont. H_2 Sample - Low Cont. Vol.	Yes	10
32.	ECR-15	Cont. H_2 Sample - Up Cont. Vol.	Yes	10
33.	ECR-16	Cont. H_2 Sample - Up Cont. Vol.	Yes	10
34.	ECR-17	Cont. H_2 Sample - Air to Rec. W	Yes	10
35.	ECR-18	Cont. H_2 Sample - Air from Rec. W	Yes	10
36.	ECR-19	Cont. H_2 Sample - Cont. Dome Vol.	Yes	10
37.	ECR-20	Cont. H_2 Sample-Return	Yes	10
38.	ECR-21	Cont. H_2 Sample - Air to Rec. E.	Yes	10
39.	ECR-22	Cont. H_2 Sample - Air fr. Rec. E	Yes	10
40.	ECR-23	Cont. H_2 Sample - Low Cont. Vol.	Yes	10
41.	ECR-24	Cont. H_2 Sample - Low Cont. Vol.	Yes	10
42.	ECR-25	Cont. H_2 Sample - Up Cont. Vol.	Yes	10
43.	ECR-26	Cont. H_2 Sample - Up Cont. Vol.	Yes	10
44.	ECR-27	Cont. H_2 Sample - Air to Rec. W.	Yes	10
45.	ECR-28	Cont. H_2 Sample - Air Fr. Rec. W.	Yes	10
46.	ECR-29	Cont. H_2 Sample - Cont. Dome Vol.	Yes	10
47.	GCR-301	N_2 Supply to Pressurizer Relief Tank	Yes	10
48.	GCR-314	N_2 Supply to Accumulators	Yes	10
49.	ICR-5	Accumulators Sample	Yes	10
50.	ICR-6	Accumulators Sample	Yes	10
51.	MCR-251	Sample Line from Steam Gen. Outlet #1	Yes	10
52.	MCR-252	Sample Line from Steam Gen. Outlet #2	Yes	10
53.	MCR-253	Sample Line from Steam Gen. Outlet #3	Yes	10
54.	MCR-254	Sample Line from Steam Gen. Outlet #4	Yes	10
55.	NCR-105	Hot Leg Sample	Yes	10
56.	NCR-106	Hot Leg Sample	Yes	10

TABLE 3.6-1 (Continued)

VALVE NUMBER		FUNCTION	TESTABLE DURING PLANT OPERATION	ISOLATION TIME IN SECONDS
B.	PHASE "B" ISOLATION (Continued)			
10.	WCR-901	NESW to Low Containment Vent #1	Yes	10
11.	WCR-903	NESW from Low Containment Vent #1	Yes	10
12.	WCR-905	NESW to Low Containment Vent #2	Yes	10
13.	WCR-907	NESW from Low Containment Vent #2	Yes	10
14.	WCR-909	NESW to Low Containment Vent #3	Yes	10
15.	WCR-911	NESW from Low Containment Vent #3	Yes	10
16.	WCR-913	NESW to Low Containment Vent #4	Yes	10
17.	WCR-915	NESW from Low Containment Vent #4	Yes	10
18.	WCR-921	NESW to Up Containment Vent #1	Yes	10
19.	WCR-923	NESW from Up Containment Vent #1	Yes	10
20.	WCR-925	NESW to Up Containment Vent #2	Yes	10
21.	WCR-927	NESW from Up Containment Vent #2	Yes	10
22.	WCR-929	NESW to Up Containment Vent #3	Yes	10
23.	WCR-931	NESW from Up Containment Vent #3	Yes	10
24.	WCR-933	NESW to Up Containment Vent #4	Yes	10
25.	WCR-935	NESW from Up Containment Vent #4	Yes	10
26.	WCR-945	NESW from RCP Motor Air Cooler	Yes	10
27.	WCR-946	NESW from RCP Motor Air Cooler	Yes	10
28.	WCR-947	NESW from RCP Motor Air Cooler	Yes	10
29.	WCR-948	NESW from RCP Motor Air Cooler	Yes	10
30.	WCR-951	NESW to RCP Motor Air Cooler Vent #1	Yes	10
31.	WCR-952	NESW to RCP Motor Air Cooler Vent #2	Yes	10
32.	WCR-953	NESW to RCP Motor Air Cooler Vent #3	Yes	10
33.	WCR-954	NESW to RCP Motor Air Cooler Vent #4	Yes	10
34.	WCR-955	NESW from RCP Motor Air Cooler Vent #1	Yes	10
35.	WCR-956	NESW from RCP Motor Air Cooler Vent #2	Yes	10
36.	WCR-957	NESW from RCP Motor Air Cooler Vent #3	Yes	10
37.	WCR-958	NESW from RCP Motor Air Cooler Vent #4	Yes	10

TABLE 3.6-1 (Continued)

VALVE NUMBER		FUNCTION	TESTABLE DURING PLANT OPERATION	ISOLATION TIME IN SECONDS
B.	PHASE "B" ISOLATION (Continued)			
38.	WCR-961	NESW to Instr. Rm. East Vent	Yes	10
39.	WCR-963	NESW from Instr. Rm. West Vent	Yes	10
40.	WCR-965	NESW to Instr. Rm. East Vent	Yes	10
41.	WCR-967	NESW from Instr. Rm. West Vent	Yes	10
42.	WCR-902	NESW from Lower Containment Vent #1	Yes	10
43.	WCR-906	NESW from Lower Containment Vent #2	Yes	10
44.	WCR-910	NESW from Lower Containment Vent #3	Yes	10
45.	WCR-914	NESW from Lower Containment Vent #4	Yes	10
46.	WCR-922	NESW from Upper Containment Vent #1	Yes	10
47.	WCR-926	NESW from Upper Containment Vent #2	Yes	10
48.	WCR-930	NESW from Upper Containment Vent #3	Yes	10
49.	WCR-934	NESW from Upper Containment Vent #4	Yes	10
50.	WCR-962	NESW from Instrument Room East Vent	Yes	10
51.	WCR-966	NESW from Instrument Room West Vent	Yes	10
C.	CONTAINMENT PURGE AND EXHAUST			
1.	VCR-101	Instr. Room Purge Air Inlet	Yes	10
2.	VCR-102	Instr. Room Purge Air Outlet	Yes	10
3.	VCR-103	Lower Comp. Purge Air Inlet	Yes	10
4.	VCR-104	Lower Comp. Purge Air Outlet	Yes	10
5.	VCR-105	Upper Comp. Purge Air Inlet	Yes	10
6.	VCR-106	Upper Comp. Purge Air Outlet	Yes	10
7.	VCR-107*	Cont. Press. Relief Fan Isolation	Yes	10
8.	VCR-201	Instr. Room Purge Air Inlet	Yes	10
9.	VCR-202	Instr. Room Purge Air Outlet	Yes	10
10.	VCR-203	Lower Comp. Purge Air Inlet	Yes	10
11.	VCR-204	Lower Comp. Purge Air Outlet	Yes	10

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CHANGE NO. 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 $\leq$ 567.8°F
K_4	=	1.075
K_5	=	0.0177/°F for increasing average temperature and 0 for decreasing average temperature
K_6	=	0.0012 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
$f_2(\Delta I)$	=	f_1 (ΔI) as defined in Note 1 above.

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

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CHANGE NO. 10

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
8. Overpower ΔT					
Four Loop Operation	4	2	3	1, 2	2
Three Loop Operation	4	1**	3	1, 2	9
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

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
14. Steam Generator Water Level--Low-Low	3/loop	2/loop in any operating loops	2/loop in each operating loop	1, 2	7
15. Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level	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
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	4	1	8
19. Safety Injection Input from ESF	2	1	2	1, 2	1

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CHANGE NO. 11

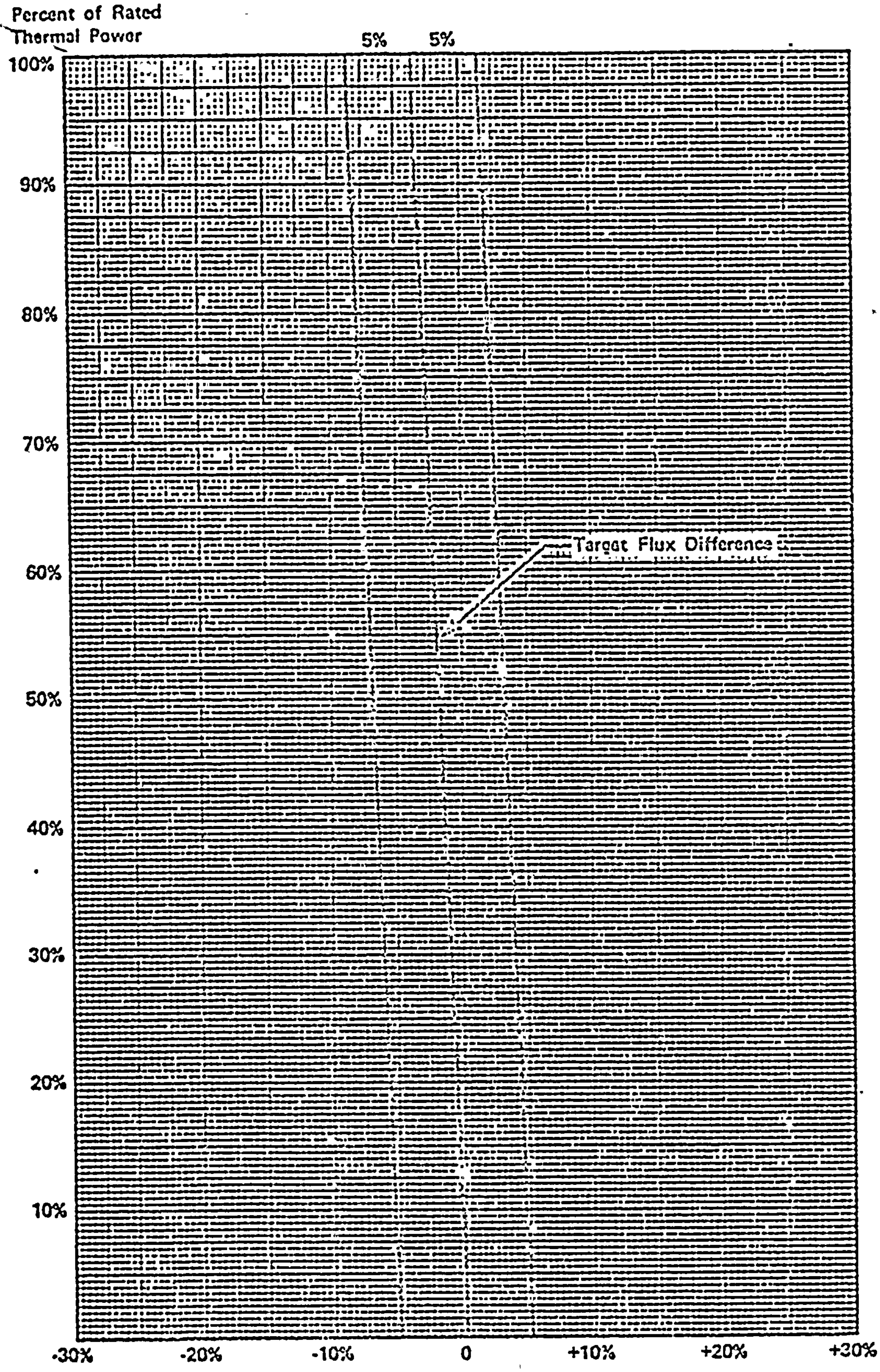

Figure B 3/4 2-1 TYPICAL INDICATED AXIAL FLUX DIFFERENCE VERSUS THERMAL POWER AT BOL

3/4.2.2 and 3/4.2.3 HEAT FLUX AND NUCLEAR ENTHALPY HOT CHANNEL FACTORS- $F_Q(Z)$ and $F^N_{\Delta H}$

The limits on heat flux and nuclear enthalpy hot channel factors 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 hot channel factors are 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 hot channel factor limits are maintained provided:

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

b. Control rod groups are sequenced with overlapping groups as described in Specification 3.1.3.5.

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.

The relaxation in $F^N_{\Delta H}$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits. $F^N_{\Delta H}$ will be maintained within its limits provided conditions a thru d above, are maintained.

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 $F^N_{\Delta H}$ is measured, experimental error must be allowed for and 4% is the appropriate allowance for a full core map taken with the incore detection system. The specified limit for $F^N_{\Delta H}$ also contains an 8% allowance for uncertainties which mean that normal operation will result in $F^N_{\Delta H} \leq 1.51/1.08$. The 8% allowance is based on the following considerations:

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CHANGE NO. 12

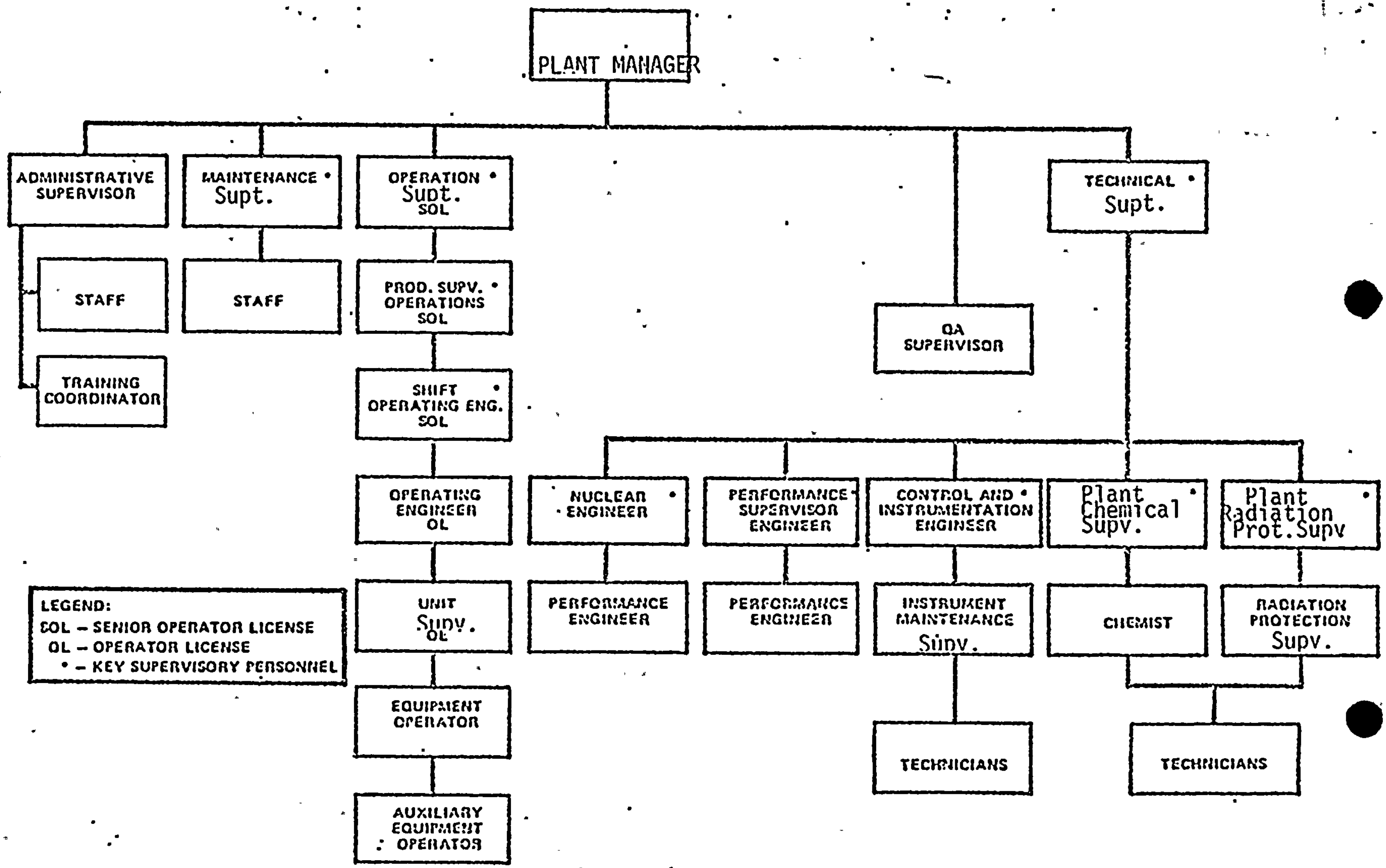

FIGURE 6.2-2 Facility Organization – Donald C. Cook – Unit No. 1

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CHANGE NO. 13

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 Director 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 in Technical Specifications or licenses.

e. Violations of applicable statutes, 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. REPORTABLE OCCURRENCES requiring 24 hour 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.

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

d. Proposed changes in Technical Specifications or licenses.

e. Violations of applicable statutes, 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. REPORTABLE OCCURRENCES requiring 24 hour 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 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. 14

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 72 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 3 months on a STAGGERED TEST BASIS by:

a. Verifying that the return air fan starts on an auto-start signal after a 9 ± 1 minute delay and operates for at least 15 minutes,

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

c. Verifying that with the fan off, the return air fan damper opens when a force of ≤ 11 lbs is applied to the counter-weight, and

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

FLOOR-DRAINS

LIMITING CONDITION FOR OPERATION

3.6.5.7 The ice condenser floor drains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the ice condenser floor drain inoperable, restore the floor drain to OPERABLE status prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.5.7 Each ice condenser floor drain shall be demonstrated OPERABLE at least once per 18 months during shutdown by:

a. Verifying that valve gate opening is not impaired by ice, frost or debris,

b. Verifying that the valve seat is not damaged,

c. Verifying that the valve gate opens when a force of $\leq$ 100 lbs is applied, and

d. Verifying that the 12 inch drain line from the ice condenser floor to the containment lower compartment is unrestricted.

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
6. Steam Flow in Two Steam Lines-High Coincident with Steam Line Pressure-Low	
a. Safety Injection (ECCS)	$\leq$ 13.0#/23.0##
b. Reactor Trip (from SI)	$\leq$ 3.0
c. Feedwater Isolation	$\leq$ 8.0
d. Containment Isolation-Phase "A"	$\leq$ 18.0#/28.0##
e. Containment Purge and Exhaust Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	Not Applicable
g. Essential Service Water System	$\leq$ 14.0#/48.0##
h. Steam Line Isolation	$\leq$ 8.0
7. Containment Pressure--High-High	
a. Containment Spray	$\leq$ 45.0
b. Containment Isolation-Phase "B"	Not Applicable
c. Steam Line Isolation	$\leq$ 7.0
d. Containment Air Recirculation Fan	$\leq$ 600.0
8. Steam Generator Water Level--High-High	
a. Turbine Trip-Reactor Trip	$\leq$ 2.5
b. Feedwater Isolation	$\leq$ 11.0

TABLE 3.3-5 (Continued)

TABLE NOTATION

* Diesel generator starting and sequence loading delays included. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps, SI and RHR pumps.

\# Diesel generator starting and sequence loading delays not included. Offsite power available. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps.

\#\# Diesel generator starting and sequence loading delays included. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps.

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CHANGE NO. 15

6.9 REPORTING REQUIREMENTS (Continued)

e. Seismic event analysis, Specification 4.3.3.3.2.

f. Sealed Source leakage on excess of limits, Specification 4.7.9.1.3.

g. Fire Detection Instrumentation, Specification 3.3.3.7.

h. Fire Suppression Systems, Specifications 3.7.9.1, 3.7.9.2, 3.7.9.3 and 3.7.9.4.

6.10 RECORD RETENTION

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

a. Records and logs of unit 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 reactor tests and experiments.

f. Records of changes made to the procedures required by Specification 6.8.1.

g. Records of radioactive shipments.

h. Records of sealed source leak tests and results.

i. 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 unit 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. Records of radiation exposure for all individuals entering radiation control areas.

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

e. Records of transient or operational cycles for those facility components identified in Table 5.9-1.

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

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

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

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

j. 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:

a. A High Radiation Area in which the intensity of radiation is greater than 100 mrem/hr but less than 1000 mrem/hr shall be barricaded and conspicuously posted as a High Radiation Area and entrance thereto shall be controlled by issuance of a Radiation Work Permit and any individual or group of individuals permitted to enter such areas shall be provided with a radiation monitoring device which continuously indicates the radiation dose rate in the area.

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CHANGE NO. 16

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

c. Reactor Coolant System Total Flow Rate

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

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

4.2.5.2 The Reactor Coolant System total flow rate shall be determined to be within its limit by measurement at least once per 18 months.

TABLE 3.2-1

DNB PARAMETERS

PARAMETER	LIMITS: 4 Loops In Operation	LIMITS: 3 Loops in Operation
Reactor Coolant System T_{avg}	$\leq$ 571.8°F	$\leq$ 571.8°F
Pressurizer Pressure	$\geq$ 2220 psia*	$\geq$ 2220 psia*
Reactor Coolant System Total Flow Rate	$\geq 1.350 \times 10^8$ lbs/hr	$\geq 0.9917 \times 10^8$ lbs/hr

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