MAY 19 1978

Docket No. 50-338

Virginia Electric & Power Company
ATTN: Mr. W. L. Proffitt
Senior Vice President - Power
P. O. Box 26666
Richmond, Virginia 23261

Gentlemen:

SUBJECT: ISSUANCE OF AMENDMENT NO. 5 TO FACILITY OPERATING LICENSE NPF-4
NORTH ANNA POWER STATION, UNIT NO. 1

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 5 to Facility Operating License NPF-4.

This amendment approves your proposed design modifications for the final solution to the NPSH problem for the recirculation spray pumps as presented in your letter, dated April 14, 1978. It also makes appropriate changes to Appendix A to the Technical Specifications related to conditions 2.D.(3)d and 2.D.(3)e which were originally contained in Amendment No. 3 to Facility Operating License NPF-4. These conditions are related to the final solution to the NPSH problem for the recirculation spray pumps.

The amendment also changes Appendix A to the Technical Specifications related to the heat flux hot channel factor (F_Q) limit. The allowable (F_Q) limit has been increased to 2.21 from 2.05 in accordance with your letter, dated May 5, 1978.

In addition, the amendment makes appropriate editorial changes to the Appendix A to the Technical Specifications. These page changes for Facility Operating License NPF-4 are attached to this license amendment.

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A copy of the Federal Register Notice concerning issuance of Amendment No. 5 and the related Safety Evaluation supporting Amendment No. 5 to Facility Operating License No. NPF-4 are enclosed.

Sincerely,

Original Signed by
O. D. Parr

Olan D. Parr, Chief
Light Water Reactors Branch No. 3
Division of Project Management

Enclosures:
As Stated

cc w/enclosures:
See next page

OFFICE	LWR #3:LA	LWR #3:PM	LWR #3:BC			
SURNAME	MRushbrook:LLM	ADromerick	OParr			
DATE	5/18/78	5/18/78	5/19/78			

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U.S. Nuclear Regulatory Commission
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cc: Mr. A. D. Johnson, Chairman
Board of Supervisors of
Louisa County
Trevillians, Virginia 23170

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Richmond, Virginia 23219

Mr. George Pence
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6th & Walnut Streets
Philadelphia, Pennsylvania 19106

Director, Technical Assessment Division
Office of Radiation Programs (AW-459)
US EPA
Crystal Mall #2
Arlington, Virginia 20460

VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-338

NORTH ANNA POWER STATION, UNIT NO. 1

FACILITY OPERATING LICENSE

License No. NPF-4
Amendment No. 5

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

   A. The applications for amendment by Virginia Electric and Power Company (the licensee), dated April 14, 1978, and May 5, 1978, comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the license, as amended, the provisions of the Act, and the rules and regulations of the Commission;

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

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

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

2. Accordingly, the license is amended by changing the following conditions to read as follows:

   a. Add the following:

      2.D.(2)f. Additional page changes to Appendix A of the Technical Specifications issued with Amendment 5 of Facility Operating license NPF-4 and attached thereto become a part of the license.

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DATE ➤						

b. Condition 2.D.(3)d shall read:

Prior to startup following the first regularly scheduled refueling outage, the Virginia Electric and Power Company shall implement the final design modification of the recirculation spray system with respect to the available net positive suction head in accordance with their submittal of April 14, 1978, concerning the final solution to the NPSH problem for the recirculation spray pumps.

c. Condition 2.D.(3)e shall read:

Until Technical Specification 3.1.2.8.b.3, 4.1.2.8.b, 3.5.5.c, 4.5.5.b, 3.6.2.2, 4.6.2.2.1.b, 4.6.2.2.1.c, 4.6.2.2.2, Table 3.6-1, Items 32, 33, 34 and 35 of Item e, Figure 3.6-1 and Table 3.7-4 (snubbers 112 and 113) become effective (1) the refueling water storage tank water temperature shall not exceed 40 degrees Fahrenheit, (2) the service water temperature shall be maintained between 35 degrees Fahrenheit and 80 degrees Fahrenheit, (3) the containment atmosphere temperature shall be maintained between 86 degrees Fahrenheit and 105 degrees Fahrenheit and (4) the containment air partial pressure shall be maintained in accordance with Figure 1 (attached to Amendment No. 3 to Facility Operating License NPF-4) to assure that adequate net positive suction head is available to the recirculation spray pumps.

d. Condition 2.D(3)i shall read:

Prior to the startup following the first regularly scheduled refueling outage, the Virginia Electric and Power Company shall install and have operational the area ambient temperature monitoring system outside containment.

Prior to the installation of the area ambient temperature monitoring system outside containment, the Virginia Electric and Power Company shall monitor and log the temperature on a daily basis of areas outside containment, as specified in Table 1 (attached to Amendment No. 3 to Facility Operating License NPF-4). Should the temperature in these Class IE areas exceed the associated equipment rating during this period, the licensee is required to report such an occurrence and provide an analysis to demonstrate acceptability of the Class IE equipment in that area.

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H. This amended license is effective as of the date of issuance except that Technical Specifications 3.1.2.8.b.3, 4.1.2.8.b, 3.5.5.c, 4.5.5.b, 3.6.2.2, 4.6.2.2.1.b, 4.6.2.2.1.c, 4.6.2.2.2, Table 3.6-1, Items 32, 33, 34 and 35 of Item e, Figure 3.6-1 and Table 3.7-4 (snubbers 112 and 113) will become effective at the time the Office of Inspection and Enforcement verifies that the modifications to the recirculation spray pumps are completed and the system operable.

FOR THE NUCLEAR REGULATORY COMMISSION

Original Signed by
O. D. Parr

O. D. Parr, Chief
Light Water Reactors Branch No. 3
Division of Project Management

Attachment:
Appendix A Technical
Specification page changes

Date of Issuance: MAY 19 1978

subject to modification on original

OFFICE ➤	DPM/LWR-3	DPM/LWR-3	OELD DS	DPM/LWR-3 ODP	DOR EW for	
SURNAME ➤	MRushbrook	ADromerick	DSWANSON	ODParr	JMcGough	
DATE ➤	5/18/78	5/18/78	5/19/78	5/19/78	5/19/78	

NRC FORM 318 (9-76) NRCM 0240 ☆ U.S. GOVERNMENT PRINTING OFFICE: 1976 – 626-624

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-338

VIRGINIA ELECTRIC AND POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 5 to the Facility Operating License No. NPF-4, issued to Virginia Electric and Power Company, which changes certain conditions contained in Facility Operating License NPF-4 Amendment No. 3. The amendment is effective as of its date of issuance.

The amendment implements the final design modification related to the NPSH problem for the recirculation spray system, raises the heat flux hot channel factor (F_Q) limit in accordance with the revised Appendix A Technical Specifications attached to the amendment, and makes appropriate editorial changes to Appendix A to the Technical Specifications.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made

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this determination, it has further been concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR Section 51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

For further details with respect to this action, see (1) Virginia Electric and Power Company letters, dated April 14, 1978, and May 5, 1978, (2) Amendment No. 5 to License No. NPF-4 with Appendix A Technical Specification page changes, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D.C. 20555 and at the Board of Supervisor's Office, Louisa County Courthouse, Louisa, Virginia 23093 and at the Alderman Library, Manuscripts Department, University of Virginia, Charlottesville, Virginia 22901. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Project Management.

Dated at Bethesda, Maryland this 19th day of May, 1978

FOR THE NUCLEAR REGULATORY COMMISSION

Original Signed by
O. D. Parr

Olan D. Parr, Chief
Light Water Reactors Branch No. 3
Division of Project Management

*See previous yellow for concurrences

OFFICE➤	DPM/LWR #3	*DPM/LWR #3	*OELD	DPM/LWR #3		
SURNAME➤	MRushbrook/LM	ADromerick	DSwanson	OParr		
DATE➤	5/ /78	5/ /78	5/ /78	5/19/78		

For further details with respect to this action, see (1) Virginia Electric and Power Company letters, dated April 14, 1978, and May 5, 1978, (2) Amendment No. 5 to License No. NPF-4 with Appendix A Technical Specification page changes, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D.C. 20555 and at the Board of Supervisor's Office, Louisa County Courthouse, Louisa, Virginia 23093 and at the Alderman Library, Manuscripts Department, University of Virginia, Charlottesville, Virginia 22901. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Project Management.

Dated at Bethesda, Maryland this day of

FOR THE NUCLEAR REGULATORY COMMISSION

Olan D. Parr, Chief
Light Water Reactors Branch No. 3
Division of Project Management

subject to modifications on original

OFFICE ➤	DPM/LWR-3	DPM/LWR-3	OELD	DPM/LWR-3		
SURNAME ➤	MRushbrook	ADromerick	D SWANSON	ODParr		
DATE ➤	5/18/78	5/18/78	5/19/78	5/ /78		

MAY 19 1978

SAFETY EVALUATION BY THE OFFICE OF

NUCLEAR REACTOR REGULATION SUPPORTING AMENDMENT NO. 5

TO LICENSE NO. NPF-4

(VIRGINIA ELECTRIC AND POWER COMPANY)

A. Evaluation of Emergency Core Cooling System Performance

In a letter dated March 30, 1978, the licensee provided an analysis of a double-ended cold leg break with a discharge coefficient of 0.4. This analysis was performed to correct a metal/water reaction calculation error in the previous analysis. Results showed a peak cladding temperature of 2070 degrees Fahrenheit, maximum local metal/water reaction of 5.6 percent, and a total corewide metal/water reaction of less than 0.3 percent, assuming a total peaking factor (F_Q) of 2.05. As stated in Section 6.3.8 of Supplement No. 9 to the North Anna Power Station, Units 1 and 2 Safety Evaluation Report, we reviewed the analysis and concluded that the worst break continued to be a double ended cold leg break with a discharge coefficient of 0.4, and that the emergency core cooling system performance conforms to the acceptance criteria in paragraph 50.46 of CFR 50.

In a letter dated May 5, 1978, the licensee submitted a reanalysis of the same break (double ended cold leg break) using the same input, models, and assumptions as the March 30, 1978 analysis, but with the total peaking factor increased to 2.21. Results reported for this analysis are a peak cladding temperature of 2198 degrees Fahrenheit, maximum local metal/water reaction of 7.906 percent, and total corewide metal/water reaction of less than 0.3 percent. These results are less that the limits specified in 10 CFR 50, paragraph 50.46 of 2200 degrees Fahrenheit, 17 percent, and 1.0 percent, respectively.

Based on our review of the results given above and the conclusions of our review as stated in Supplement No. 9 to the North Anna Power Station, Unit 1 and 2 Safety Evaluation Report, we conclude that the emergency core cooling system performance conforms to the requirements of paragraph 50.46 of CFR 50 and is, therefore, acceptable. Appendix A to the Technical Specifications is being changed to incorporate the appropriate changes.

B. Evaluation of Modified Containment Heat Removal Systems

Supplement Nos. 8 and 9 to the North Anna Power Station, Units 1 and 2, Safety Evaluation Report discuss the interim solution to the net positive suction head problem for the recirculation spray pumps. The interim system modifications consisted of diverting 150 gallons per minute of quench spray system water to the suction side of each inside recirculation spray pump and limiting the flow of each outside recirculation spray pump to 2000 gallons per minute by installing a flow restricting orifice in the discharge line of each pump.

We stated in Supplement No. 8 to the Safety Evaluation Report that the licensee's analysis of the available net positive suction head to the recirculation spray pumps and the containment depressurization analysis, based on the interim system modifications, were acceptable provided the licensee's proposed plant operating restrictions remained in effect until a final resolution of the net positive suction head problem had been implemented in the plant design. We also stated in Supplement No. 8 that the licensee was considering alternative system design changes for the recirculation spray system as a final resolution of the net positive suction head matter.

The final system modifications proposed by the licensee to satisfy net positive suction head requirements of the recirculation spray system pumps, and the core and containment cooling requirements are discussed below:

B.1 Inside Recirculation Spray Pumps

For the inside recirculation spray pumps, the licensee proposes to retain the interim fix modification described above and in Supplement No. 8 to the Safety Evaluation Report. We have previously concluded on the acceptability of this design change in Supplement No. 8 to the Safety Evaluation Report.

B.2 Outside Recirculation Spray Pumps

For the outside recirculation spray pumps, the licensee proposes, as a final solution, to remove the flow restricting orifice in the discharge line of each outside recirculation spray pump, and reinstall spray header nozzles that had been removed for the interim fix. Since the outside recirculation spray pump flow rates will increase from 2000 gallons per minute to 3,640 gallons per minute, the required net positive suction head will also increase and therefore it will be necessary to increase the available net positive suction head.

The licensee proposed to increase the available net positive suction head to the outside recirculation spray pumps by injecting 800 gallons per minute of 50 degrees Fahrenheit water into each outside recirculation spray pump suction line. A new system, designated the casing cooling subsystem will be used to accomplish this. The subsystem will consist of a separate supply tank, with a usable capacity of 110,000 gallons of chilled water, and two separate and independent flow paths for injecting cooling water into each outside recirculation spray pump suction line. The usable capacity of the tank is greater than that needed to satisfy the outside recirculation spray pump net positive suction head requirements over the short time interval of interest (several hundred seconds) at the beginning of a loss-of-coolant accident transient.

The containment depressurization actuation signal will initiate operation of the casing cooling subsystem by starting the casing cooling pumps, opening the valves closest to the containment and provide a signal to the normally open valves to assure that they are open. Low supply tank level will automatically close the supply line valves closest to the containment. Low pump discharge flow, concurrent with a containment depressurization actuation signal, will automatically close the second valve in each line. The casing cooling subsystem can also be manually operated from the control room. The two motor operated valves in each discharge line will be powered from redundant safety related motor control centers to ensure containment isolation when required.

Based on the above discussion, we have concluded that the licensee has adequately considered single active failures in the design of the casing cooling subsystem. Therefore, we have concluded that the system design is acceptable from a functional standpoint, including the capability to subsequently isolate the system. Furthermore, the system is designed to seismic Category I requirements, and component codes and standards, which we find acceptable.

B.3 <u>Cross-Connect Between the Outside Recirculation Spray System and the Low Head Safety Injection System</u>

The licensee proposes to retain the system modifications that will permit the outside recirculation spray pumps to provide backup (after 24 hours) emergency core cooling capability to the low head safety injection pumps. The modifications that have been made to interconnect the outside recirculation spray system and the low head injection system are described in Supplement No. 9 to the North Anna Power Station, Units 1 and 2 Safety Evaluation Report. We have previously concluded on the acceptability of this design change in Supplement No. 9.

B.4 Supporting Analyses

The following is an evaluation of the licensee's containment response analyses as they affect (1) the calculation of pump available net positive suction head, (2) the containment depressurization time and the long term capability to maintain a subatmospheric condition in the containment and (3) the minimum containment pressure for the emergency core cooling system performance evaluation. The licensee's calculations are based on the final system modifications, without the operating restrictions specified in Supplement No. 8 to the Safety Evaluation Report.

(1) Mass and Emergency Release Data Used in Analyses

The mass and energy release rates were calculated using the LOCTIC code which has not been reviewed by us. In a letter dated May 11, 1978, the Virginia Electric and Power Company provided a comparison of the mass and energy release data calculated by the LOCTIC code with data calculated using the SATAN-V, W REFLOOD and FROTH codes as described in Topical Report WCAP-8312A, "Westinghouse Mass and Energy Release Data for Containment Design." This topical report was approved by us in a Topical Report Evaluation attached to a letter to Westinghouse dated March 12, 1975.

The results of the comparison indicated that the LOCTIC code produced essentially the same mass and energy release data as the methods of WCAP-8312A. The total energy release was about 2 percent higher in the WCAP-8312A analysis than in the LOCTIC analysis for periods less than three hours. For periods longer than three hours, the LOCTIC code produced higher energy releases. The licensee's analyses of the long term containment response indicated no difference between the two methods. We have therefore concluded that the LOCTIC code is an acceptable method of calculating mass and energy release rates for the containment analysis of North Anna Power Station, Units 1 and 2.

(2) Containment Depressurization Analysis

In view of the system modifications that were found necessary to satisfy the net positive suction head requirements of the recirculation spray pumps, the licensee has redone the containment depressurization analysis to verify the acceptability of the depressurization time used in evaluating the offsite radiological consequences following a postulated loss-of-coolant accident. The analysis was based on a cold leg (pump suction) double-ended rupture at 102 percent of full power; minimum engineered safeguards were assumed. We have reviewed the input parameters used by the licensee to perform the depressurization analysis and conclude that the analysis results in a reasonably conservative calculation of the containment depressurization time.

The licensee's analysis shows that the containment would be depressurized within one hour and would remain subatmospheric. The licensee also provided an analysis to justify that only one outside recirculation spray pump is needed to satisfy both the containment spray and core cooling requirements 24 hours after a postulated loss-of-coolant accident.

We have also done a confirmatory analysis which shows that the containment would be depressurized in one hour and remain subatmospheric. We have also confirmed that one outside recirculation spray pump is capable of maintaining a subatmospheric condition in the containment with a portion of the flow diverted for core cooling purposes. We, therefore, conclude that the licensee's containment depressurization analysis is acceptable.

(3) Net Positive Suction Head Analysis

The licensee has also performed a sensitivity study to identify the pipe break location and single active failure that will result in the lowest available net positive suction head for the recirculation spray pumps, to justify the acceptability of the final recirculation spray system modifications. Other factors used in the calculation of the available net positive suction head, namely, depth of sump water and suction piping resistance to flow, have a lesser effect on the revised analyses. A postulated hot leg double-ended rupture, with normal engineered safeguards was found to result in the lowest available net positive suction head. The licensee has calculated that the available net positive suction head will be 11.5 feet and 16.6 feet for the inside recirculation spray and outside recirculation spray pumps, respectively. The results of the recirculation spray pump testing indicate that the required net positive suction head is 9.4 feet for the inside recirculation spray pumps at the design flow rate of 3300 gallons per minute, and 11.0 feet for the outside recirculation spray pumps at the design flow rate of 3640 gallons per minute. Therefore, adequate margin will exist.

The licensee has also calculated that the available net positive suction head to the outside recirculation spray pumps will be 20.9 feet when the recirculation spray system is operating in the cross-connect mode which provides an acceptable margin above the required net positive suction head stated above.

The analytical model used by the licensee to determine the available net positive suction head is described in Supplement No. 8 to the Safety Evaluation Report. We have previously concluded, in Supplement No. 8, that the analytical model is acceptable.

We have done a confirmatory analysis using a modified version of the CONTEMPT-LT (MOD 26) computer code, as described in Supplement No. 8, to permit containment analyses to be based on the pressure flash method. This method minimizes the calculated containment pressure and maximizes containment sump temperature. These assumptions minimize the calculated net positive suction head. Our calculations of the containment pressure and sump water temperature versus time and in good agreement with the licensee's results. We, therefore, conclude that the licensee's net positive suction head analysis is acceptable.

(4) Minimum Containment Pressure Analysis for ECCS Performance Evaluation

In view of the changes that were made to solve the net positive suction head problem, there is no change in the previously accepted minimum containment pressure analysis as stated in Supplement Nos. 8 and 9 to the North Anna Power Station, Units 1 and 2 Safety Evaluation Report.

C. Environmental Consideration

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR Section 51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

D. Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered or a significant decrease in any safety margin, it does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public. Also, we reaffirm our conclusions as otherwise stated in our Safety Evaluation Report and its Supplements.

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

VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-338

NORTH ANNA POWER STATION, UNIT NO. 1

FACILITY OPERATING LICENSE

License No. NPF-4
Amendment No. 5

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

   A. The applications for amendment by Virginia Electric and Power Company (the licensee), dated April 14, 1978, and May 5, 1978, comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the license, as amended, the provisions of the Act, and the rules and regulations of the Commission;

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

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

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

2. Accordingly, the license is amended by changing the following conditions to read as follows:

   a. Add the following:

      2.D.(2)f. Additional page changes to Appendix A of the Technical Specifications issued with Amendment 5 of Facility Operating license NPF-4 and attached thereto become a part of the license.

b. Condition 2.D.(3)d shall read:

Prior to startup following the first regularly scheduled refueling outage, the Virginia Electric and Power Company shall implement the final design modification of the recirculation spray system with respect to the available net positive suction head in accordance with their submittal of April 14, 1978, concerning the final solution to the NPSH problem for the recirculation spray pumps.

c. Condition 2.D.(3)e shall read:

Until Technical Specification 3.1.2.8.b.3, 4.1.2.8.b, 3.5.5.c, 4.5.5.b, 3.6.2.2, 4.6.2.2.1.b, 4.6.2.2.1.c, 4.6.2.2.2, Table 3.6-1, Items 32, 33, 34 and 35 of Item e, Figure 3.6-1 and Table 3.7-4 (snubbers 112 and 113) become effective (1) the refueling water storage tank water temperature shall not exceed 40 degrees Fahrenheit, (2) the service water temperature shall be maintained between 35 degrees Fahrenheit and 80 degrees Fahrenheit, (3) the containment atmosphere temperature shall be maintained between 86 degrees Fahrenheit and 105 degrees Fahrenheit and (4) the containment air partial pressure shall be maintained in accordance with Figure 1 (attached to Amendment No. 3 to Facility Operating License NPF-4) to assure that adequate net positive suction head is available to the recirculation spray pumps.

d. Condition 2.D(3)i shall read:

Prior to the startup following the first regularly scheduled refueling outage, the Virginia Electric and Power Company shall install and have operational the area ambient temperature monitoring system outside containment.

Prior to the installation of the area ambient temperature monitoring system outside containment, the Virginia Electric and Power Company shall monitor and log the temperature on a daily basis of areas outside containment, as specified in Table 1 (attached to Amendment No. 3 to Facility Operating License NPF-4). Should the temperature in these Class IE areas exceed the associated equipment rating during this period, the licensee is required to report such an occurrence and provide an analysis to demonstrate acceptability of the Class IE equipment in that area.

H. This amended license is effective as of the date of issuance except that Technical Specifications 3.1.2.8.b.3, 4.1.2.8.b, 3.5.5.c, 4.5.5.b, 3.6.2.2, 4.6.2.2.1.b, 4.6.2.2.1.c, 4.6.2.2.2, Table 3.6-1, Items 32, 33, 34 and 35 of Item e, Figure 3.6-1 and Table 3.7-4 (snubbers 112 and 113) will become effective at the time the Office of Inspection and Enforcement verifies that the modifications to the recirculation spray pumps are completed and the system operable.

FOR THE NUCLEAR REGULATORY COMMISSION

Olan D. Parr

O. D. Parr, Chief
Light Water Reactors Branch No. 3
Division of Project Management

Attachment:
Appendix A Technical
Specification page changes

Date of Issuance: MAY 19 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 5

FACILITY OPERATING LICENSE NO. NPF-4

DOCKET NO. 50-338

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages as indicated. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

Pages

XV
XIX
3/4 1-16
3/4 1-17
3/4 2-1
3/4 2-2
3/4 2-4
3/4 2-5
3/4 2-8
3/4 2-16
3/4 2-17
3/4 2-18
3/4 3-33
3/4 3-55
3/4 4-13
3/4 5-9
3/4 6-6
3/4 6-12
3/4 6-12a (added)
3/4 6-27a (added)
3/4 7-56

Pages

B 3/4 1-3
B 3/4 2-1
B 3/4 2-2
B 3/4 2-6
B 3/4 4-11
6-11
6-12
6-13
6-14
6-15
6-16

INDEX

BASES

SECTION | PAGE

INDEX

BASES

SECTION PAGE

ADMINISTRATIVE CONTROLS

BORATED WATER SOURCES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.7 As a minimum, one of the following borated water sources shall be OPERABLE:

a. A boric acid storage system and associated heat tracing with:

1. A minimum contained borated water volume of 835 gallons,
2. Between 20,000 and 22,500 ppm of boron, and
3. A minimum solution temperature of 145°F.

b. The refueling water storage tank with:

1. A minimum contained borated water volume of 9690 gallons,
2. Between 2000 and 2100 ppm of boron, and
3. A minimum solution temperature of 35°F.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no borated water source OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until at least one borated water source is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.1.2.7 The above required borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the boron concentration of the water,
2. Verifying the contained borated water volume of the tank, and
3. Verifying the boric acid storage tank solution temperature when it is the source of borated water.

b. At least once per 24 hours by verifying the RWST temperature when it is the source of borated water and the outside air temperature is < 35°F.

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.8 Each of the following borated water sources shall be OPERABLE:

a. A boric acid storage system and associated heat tracing with:

1. A contained borated water volume of between 4450 and 16,280 gallons,

2. Between 20,000 and 22,500 ppm of boron, and

3. A minimum solution temperature of 145°F.

b. The refueling water storage tank with:

1. A contained borated water volume of between 450, 000 and 464,000 gallons,

2. Between 2000 and 2100 ppm of boron, and

3. A solution temperature between 40°F and 50°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With the boric acid storage system inoperable, restore the storage system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to at least 1.77% $\Delta k/k$ at 200°F; restore the boric acid storage system to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the refueling water storage tank inoperable, restore the tank to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.8 Each borated water source shall be demonstrated OPERABLE:

SURVEILLANCE REQUIREMENTS (Continued)

a. At least once per 7 days by:

1. Verifying the boron concentration in each water source,

2. Verifying the contained borated water volume of each water source, and

3. Verifying the boric acid storage system solution temperature.

b. At least once per 24 hours by verifying the RWST temperature.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

GROUP HEIGHT

LIMITING CONDITION FOR OPERATION

3.1.3.1 All full length (shutdown and control) rods and all part length rods which are inserted in the core shall be OPERABLE and positioned within ± 12 steps (indicated position) of their group step counter demand position.

APPLICABILITY: MODES 1* and 2*

ACTION:

a. With one or more full length rods inoperable due to being immovable as a result of excessive friction or mechanical interference or known to be untrippable, determine, within 1 hour that the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied and be in HOT STANDBY within 6 hours.

b. With more than one full or part length rod inoperable or misaligned from the bank step counter demand position by more than ± 12 steps (indicated position), be in HOT STANDBY within 6 hours.

c. With one full or part length rod inoperable due to causes other than those addressed by ACTION "a" above or misaligned from its bank step counter demand height by more than ± 12 steps (indicated position), POWER OPERATION may continue provided that within one hour either:

   1. The rod is restored to OPERABLE status within the above alignment requirements, or

   2. The rod is declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. POWER OPERATION may then continue provided that:

      a) A reevaluation of each accident analysis of Table 3.1-1 is performed within 5 days. This reevaluation shall confirm that the previous analyzed results of these accidents remain valid for the duration of operation under these conditions, and

*See Special Test Exceptions 3.10.2 and 3.10.3.

AXIAL FLUX DIFFERENCE (AFD)

LIMITING CONDITION FOR OPERATION

3.2.1 The indicated AXIAL FLUX DIFFERENCE (AFD) shall be maintained within a ±5% target band (flux difference units) about the target flux difference.

APPLICABILITY: MODE 1 ABOVE 50% RATED THERMAL POWER*

ACTION:

a. With the indicated AXIAL FLUX DIFFERENCE outside of the ± 5% target band about the target flux difference and with THERMAL POWER:

1. Above 85% of RATED THERMAL POWER, within 15 minutes:

   a) Either restore the indicated AFD to within the target band limits, or

   b) Reduce THERMAL POWER to less than 85% of RATED THERMAL POWER.

2. Between 50% and 85% of RATED THERMAL POWER:

   a) POWER OPERATION may continue provided:

      1) The indicated AFD has not been outside of the ± 5% target band for more than 1 hour penalty deviation cumulative during the previous 24 hours, and

      2) The indicated AFD is within the limits shown on Figure 3.2-1. Otherwise, reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 30 minutes and reduce the Power Range Neutron Flux-High Trip Setpoints to ≤ 55% of RATED THERMAL POWER within the next 4 hours.

   b) Surveillance testing of the Power Range Neutron Flux Channels may be performed pursuant to Specification 4.3.1.1.1 provided the indicated AFD is maintained within the limits of Figure 3.2-1. A total of 16 hours of operation may be accumulated with the AFD outside of the target band during this testing without penalty deviation.

---

*See Special Test Exception 3.10.2.

LIMITING CONDITION FOR OPERATION (Continued)

b. THERMAL POWER shall not be increased above 85% of RATED THERMAL POWER unless the indicated AFD is within the ± 5% target band and ACTION 2.a.1, above has been satisfied.

c. THERMAL POWER shall not be increased above 50% of RATED THERMAL POWER unless the indicated AFD has not been outside of the ± 5% target band for more than 1 hour penalty deviation cumulative during the previous 24 hours.

SURVEILLANCE REQUIREMENTS

4.2.1.1 The indicated AXIAL FLUX DIFFERENCE shall be determined to be within its limits during POWER OPERATION above 15% of RATED THERMAL POWER by:

a. Monitoring the indicated AFD for each OPERABLE excore channel:

1. At least once per 7 days when the AFD Monitor Alarm is OPERABLE, and

2. At least once per hour for the first 24 hours after restoring the AFD Monitor Alarm to OPERABLE status.

b. Monitoring and logging the indicated AXIAL FLUX DIFFERENCE for each OPERABLE excore channel at least once per hour for the first 24 hours and at least once per 30 minutes thereafter, when the AXIAL FLUX DIFFERENCE Monitor Alarm is inoperable. The logged values of the indicated AXIAL FLUX DIFFERENCE shall be assumed to exist during the interval preceding each logging.

4.2.1.2 The indicated AFD shall be considered outside of its ± 5% target band when at least 2 of 4 or 2 of 3 OPERABLE excore channels are indicating the AFD to be outside the target band. Penalty deviation outside of the ± 5% target band shall be accumulated on a time basis of:

a. One minute penalty deviation for each one minute of POWER OPERATION outside of the target band at THERMAL POWER levels equal to or above 50% of RATED THERMAL POWER, and

b. One-half minute penalty deviation for each one minute of POWER OPERATION outside of the target band at THERMAL POWER levels between 15% and 50% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS (Continued)

4.2.1.3 The target flux difference of each OPERABLE excore channel shall be determined by measurement at least once per 92 Effective Full Power Days with all part length control rods fully withdrawn. The provisions of Specification 4.0.4 are not applicable.

4.2.1.4 The target flux difference shall be updated at least once per 31 Effective Full Power Days by either determining the target flux difference pursuant to 4.2.1.3 above or by linear interpolation between the most recently measured value and 0 percent at the end of the cycle life. The provisions of Specification 4.0.4 are not applicable.

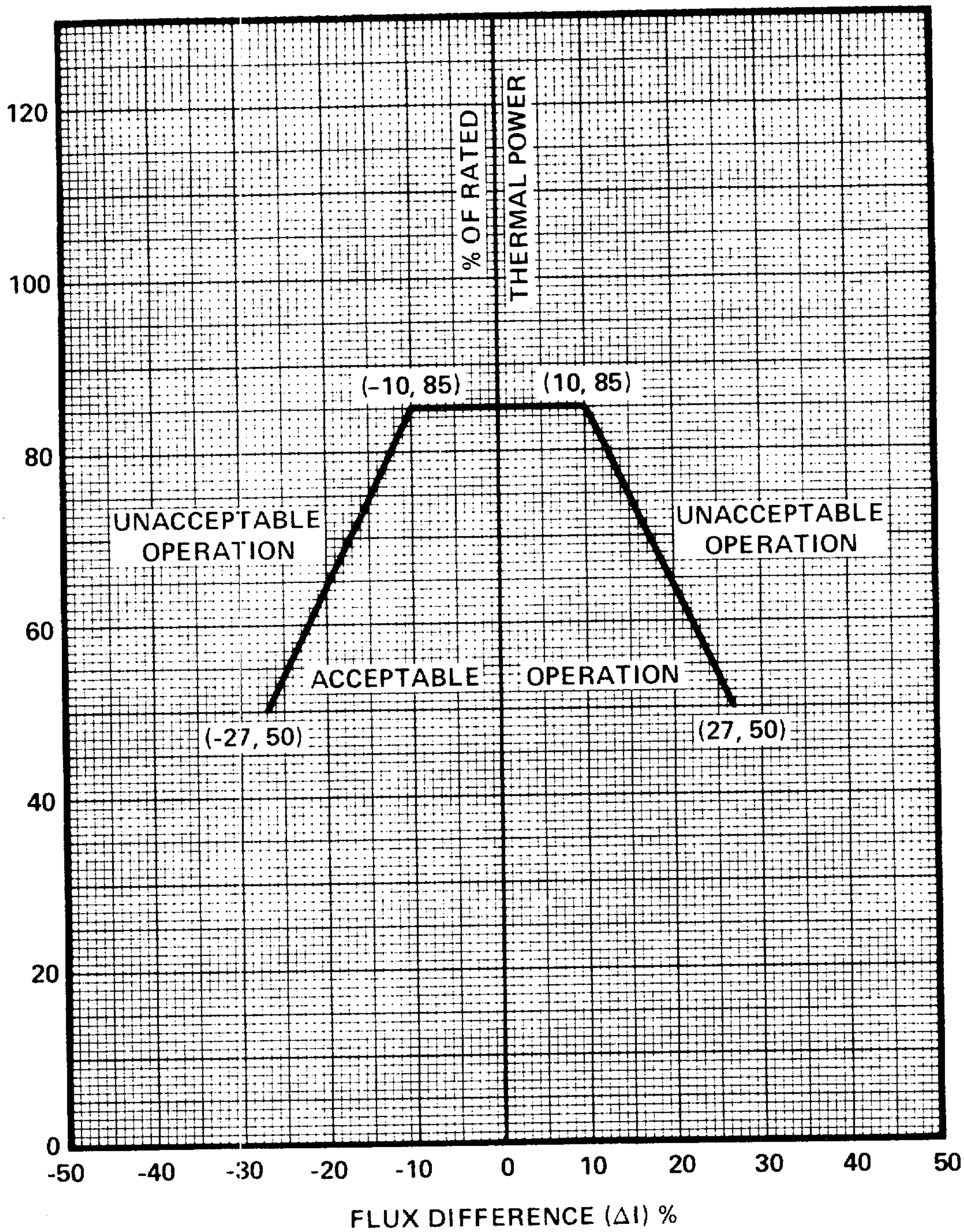

FIGURE 3.2-1 AXIAL FLUX DIFFERENCE LIMITS AS A FUNCTION OF RATED THERMAL POWER

POWER DISTRIBUTION LIMITS

HEAT FLUX HOT CHANNEL FACTOR-$F_Q(Z)$

LIMITING CONDITION FOR OPERATION

3.2.2 $F_Q(Z)$ shall be limited by the following relationships:

$$F_Q(Z) \leq \left[\frac{2.21}{P}\right] [K(Z)] \text{ for } P > 0.5$$

$$F_Q(Z) \leq [4.42] [K(Z)] \text{ for } P \leq 0.5$$

$$\text{where } P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$$

and K(Z) is the function obtained from Figure 3.2-2 for a given core height location.

APPLICABILITY: MODE 1

ACTION:

With $F_Q(Z)$ exceeding its limit:

a. Comply with either of the following ACTIONS:

1. Reduce THERMAL POWER at least 1% for each 1% $F_Q(Z)$ exceeds the limit within 15 minutes and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours; POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower ΔT Trip Setpoints have been reduced at least 1% for each 1% $F_Q(Z)$ exceeds the limit. The Overpower ΔT Trip Setpoint reduction shall be performed with the reactor in at least HOT STANDBY.

2. Reduce THERMAL POWER as necessary to meet the limits of Specification 3.2.6 using the APDMS with the latest incore map and updated $\bar{R}$.

b. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER above the reduced limit required by a, above; THERMAL POWER may then be increased provided $F_Q(Z)$ is demonstrated through incore mapping to be within its limit.

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 F_{xy} shall be evaluated to determine if $F_Q(Z)$ is within its limit by:

a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER greater than 5% of RATED THERMAL POWER.

b. Increasing the measured F_{xy} component of the power distribution map by 3% to account for manufacturing tolerances and further increasing the value by 5% to account for measurement uncertainties.

c. Comparing the F_{xy} computed (F_{xy}^C) obtained in b, above to:

1. The F_{xy} limits for RATED THERMAL POWER (F_{xy}^{RTP}) for the appropriate measured core planes given in e and f, below, and

2. The relationship:

$$F_{xy}^L = F_{xy}^{RTP} [1 + 0.2(1-P)]$$

where F_{xy}^L is the limit for fractional THERMAL POWER operation expressed as a function of F_{xy}^{RTP} and P is the fraction of RATED THERMAL POWER at which F_{xy} was measured.

d. Remeasuring F_{xy} according to the following schedule:

1. When F_{xy}^C is greater than the F_{xy}^{RTP} limit for the appropriate measured core plane but less than the F_{xy}^L relationship, additional power distribution maps shall be taken and F_{xy}^C compared to F_{xy}^{RTP} and F_{xy}^L:

a) Either within 24 hours after exceeding by 20% of RATED THERMAL POWER or greater, the THERMAL POWER at which F_{xy}^C was last determined, or

SURVEILLANCE REQUIREMENTS (Continued)

b) At least once per 31 EFPD, whichever occurs first.

2. When the F_{xy}^{C} is less than or equal to the F_{xy}^{RTP} limit for the appropriate measured core plane, additional power distribution maps shall be taken and F_{xy}^{C} compared to F_{xy}^{RTP} and F_{xy}^{L} at least once per 31 EFPD.

e. The F_{xy} limits for RATED THERMAL POWER within specific core planes shall be:

1. $F_{xy}^{RTP} \leq 1.71$ for all core planes containing bank "D" control rods and/or any part length rods, and

2. $F_{xy}^{RTP} \leq 1.55$ for all unrodded core planes.

f. The F_{xy} limits of e, above, are not applicable in the following core plane regions as measured in percent of core height from the bottom of the fuel:

1. Lower core region from 0 to 15%, inclusive.

2. Upper core region from 85 to 100%, inclusive.

3. Grid plane regions at 17.8 ± 2%, 32.1 ± 2%, 46.4 ± 2%, 60.6 ± 2% and 74.9 ± 2%, inclusive (17 x 17 fuel elements).

4. Core plane regions within ± 2% of core height (± 2.88 inches) about the bank demand position of the bank "D" or part length control rods.

g. With F_{xy}^{C} exceeding F_{xy}^{L} the effects of F_{xy} on $F_Q(Z)$ shall be evaluated to determine if $F_Q(Z)$ is within its limit.

4.2.2.3 When $F_Q(Z)$ is measured for other than F_{xy} determination, an overall measured $F_Q(Z)$ shall be obtained from a power distribution map and increased by 3% to account for manufacturing tolerances and further increased by 5% to account for measurement uncertainty.

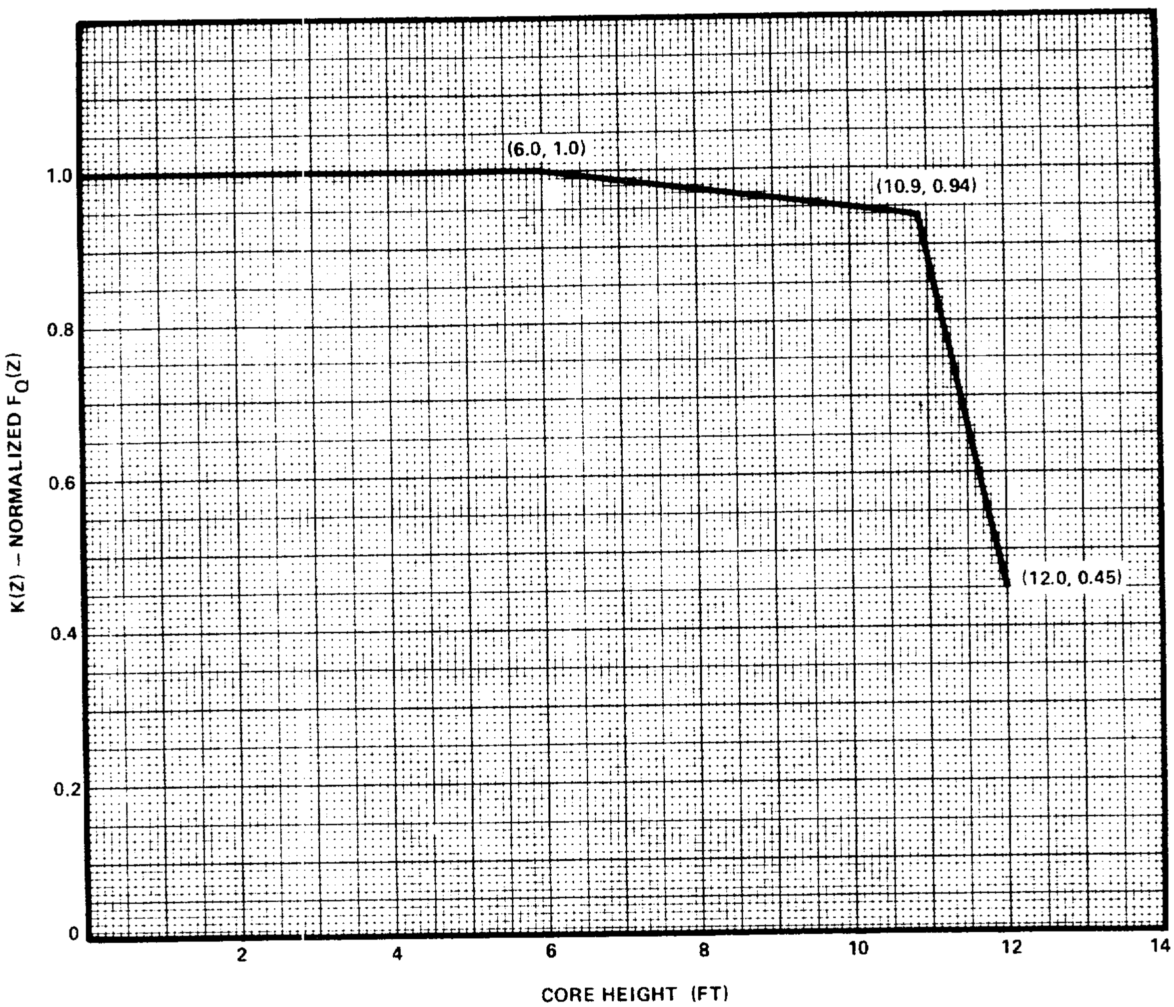

Figure 3.2–2 K(Z) – Normalized $F_Q(Z)$ as a Function of Core Height

TABLE 3.2-1

DNB PARAMETERS

PARAMETER	LIMITS 3 Loops In Operation	2 Loops In Operation** & Loop Stop Valves Open	2 Loops In Operation** & Isolated Loop Stop Valves Closed
Reactor Coolant System T_{avg}	$\leq$ 585°F		
Pressurizer Pressure	$\geq$ 2205 psig*		
Reactor Coolant System Total Flow Rate	$\geq$ 278,400 gpm		

*Limit not applicable during either a THERMAL POWER ramp increase in excess of 5% RATED THERMAL POWER per minute or a THERMAL POWER step increase in excess of 10% RATED THERMAL POWER.

**Values dependent on NRC approval of ECCS evaluation for these conditions

LIMITING CONDITION FOR OPERATION

3.2.6 The axial power distribution shall be limited by the following relationship:

$$[F_j(Z)]_S = \frac{[2.21]\,[K(Z)]}{(\overline{R}_j)(P_L)(1.03)(1 + \sigma_j)(1.07)}$$

Where:

a. $F_j(Z)$ is the normalized axial power distribution from thimble j at core elevation Z.

b. P_L is the fraction of RATED THERMAL POWER.

c. K(Z) is the function obtained from Figure 3.2-2 for a given core height location.

d. $\overline{R}_j$, for thimble j, is determined from at least n=6 in-core flux maps covering the full configuration of permissible rod patterns above 95% of RATED THERMAL POWER in accordance with:

$$\overline{R}_j = \frac{1}{n} \sum_{i=1}^{n} R_{ij}$$

Where:

$$R_{ij} = \frac{F_{Qi}^{Meas}}{[F_{ij}(Z)]_{Max}}$$

and $[F_{ij}(Z)]_{Max}$ is the maximum value of the normalized axial distribution at elevation Z from thimble j in map i which had a measured peaking factor without uncertainties or densification allowance of F_Q^{Meas}.

LIMITING CONDITION FOR OPERATION (Continued)

σ_j is the standard deviation associated with thimble j, expressed as a fraction or percentage of $\overline{R}_j$, and is derived from n flux maps from the relationship below, or 0.02, (2%) whichever is greater.

$$\sigma_j = \frac{\left[\frac{1}{n-1}\sum_{i=1}^{n}(\overline{R}_j - R_{ij})^2\right]^{1/2}}{\overline{R}_j}$$

The factor 1.07 is comprised of 1.02 and 1.05 to account for the axial power distribution instrumentation accuracy and the measurement uncertainty associated with F_Q using the movable detector system, respectively.

The factor 1.03 is the engineering uncertainty factor.

APPLICABILITY: MODE 1 above 95% OF RATED THERMAL POWER#.

ACTION:

a. With a $F_j(Z)$ factor exceeding $[F_j(Z)]_S$ by $\leq$ 4 percent, reduce THERMAL POWER one percent for every percent by which the $F_j(Z)$ factor exceeds its limit within 15 minutes and within the next two hours either reduce the $F_j(Z)$ factor to within its limit or reduce THERMAL POWER to 95% or less of RATED THERMAL POWER.

b. With a $F_j(Z)$ factor exceeding $[F_j(Z)]_S$ by > 4 percent, reduce THERMAL POWER to 95% or less of RATED THERMAL POWER within 15 minutes.

The APDMS may be out of service when surveillance for determining power distribution maps is being performed.

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 3.3.3.8 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 95% 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 95% 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.

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
4. STEAM LINE ISOLATION				
a. Manual	N.A.	N.A.	R	1, 2, 3
b. Automatic Actuation Logic	N.A.	N.A.	M(2)	1, 2, 3
c. Containment Pressure--Intermediate High-High	S	R	M	1, 2, 3
d. Steam Flow in Two Steam Lines--High Coincident with T_{avg} -- Low or Steam Line Pressure--Low	S	R	M	1, 2, 3
5. TURBINE TRIP AND FEEDWATER ISOLATION				
a. Steam Generator Water Level--High-High	S	R	M	1, 2, 3
6. AUXILIARY FEEDWATER PUMPS				
a. Steam Generator Water Level--Low-Low	S	R	M	1, 2, 3, 4
b. S.I.	See 1 above (all S.I. Surveillance Requirements)			
c. Station Blackout	N.A.	R	N.A.	1, 2, 3, 4
d. Main Feedwater Pump Trip	N.A.	N.A.	R	1, 2

TABLE 4.3-2 (Continued)

TABLE NOTATION

(1) Manual actuation switches shall be tested at least once per 18 months during shutdown. All other circuitry associated with manual safeguards actuation shall receive a CHANNEL FUNCTIONAL TEST at least once every other 31 days.

(2) Each train or logic channel shall be tested at least every other 31 days.

(3) The CHANNEL FUNCTIONAL TEST shall include exercising the transmitter by applying either a vacuum or pressure to the appropriate side of the transmitter.

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.

4.3.3.8.2 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, by performance of a CHANNEL CALIBRATION.

9. Preservice Inspection means an inspection of the full length of each tube in each steam generator performed by eddy current techniques prior to service to establish a baseline condition of the tubing. This inspection shall be performed after the field hydrostatic test and prior to initial POWER OPERATION using the equipment and techniques expected to be used during subsequent inservice inspection.

b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug all tubes exceeding the plugging limit and all tubes containing through-wall cracks) required by Table 4.4-2.

4.4.5.5 Reports

a. Following each inservice inspection of steam generator tubes, the number of tubes plugged in each steam generator shall be reported to the Commission within 15 days.

b. The complete results of the steam generator tube inservice inspection shall be reported on an annual basis for the period in which this inspection was completed. This report shall include:

1. Number and extent of tubes inspected.

2. Location and percent of wall-thickness penetration for each indication of an imperfection.

3. Identification of tubes plugged.

c. Results of steam generator tube inspections which fall into Category C-3 and require prompt notification of the Commission shall be reported pursuant to Specification 6.9.1.8 prior to resumption of plant operation. The written followup of this report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

TABLE 4.4-1

MINIMUM NUMBER OF STEAM GENERATORS TO BE INSPECTED DURING INSERVICE INSPECTION

Preservice Inspection	No			Yes		
No. of Steam Generators per Unit	Two	Three	Four	Two	Three	Four
First Inservice Inspection	All			One	Two	Two
Second & Subsequent Inservice Inspections	One[1]			One[1]	One[2]	One[3]

Table Notation:

1. The inservice inspection may be limited to one steam generator on a rotating schedule encompassing 3 N % of the tubes (where N is the number of steam generators in the plant) if the results of the first or previous inspections indicate that all steam generators are performing in a like manner. Note that under some circumstances, the operating conditions in one or more steam generators may be found to be more severe than those in other steam generators. Under such circumstances the sample sequence shall be modified to inspect the most severe conditions.

2. The other steam generator not inspected during the first inservice inspection shall be inspected. The third and subsequent inspections should follow the instructions described in 1 above.

3. Each of the other two steam generators not inspected during the first inservice inspections shall be inspected during the second and third inspections. The fourth and subsequent inspections shall follow the instructions described in 1 above.

REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.5.5 The refueling water storage tank (RWST) shall be OPERABLE with:

a. A contained borated water volume of between 450,000 and 464,000 gallons.

b. Between 2000 and 2100 ppm of boron, and

c. A solution temperature between 40°F and 50°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the refueling water storage tank inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.5.5 The RWST shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the contained borated water volume in the tank, and

2. Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the RWST temperature.

INTERNAL PRESSURE

LIMITING CONDITION FOR OPERATION

3.6.1.4 Primary containment internal air partial pressure shall be maintained $\geq$ 8.9 psia and within the acceptable operation range of the applicable RWST water temperature limit lines and bulk air temperature limit lines shown on Figure 3.6-1.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the containment internal air partial pressure < 8.9 psia or above the applicable RWST water temperature limit line or bulk air temperature line shown on Figure 3.6-1, restore the internal air partial pressure to within the limits 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

4.6.1.4 The primary containment internal air partial pressure shall be determined to be within the limits at least once per 12 hours.

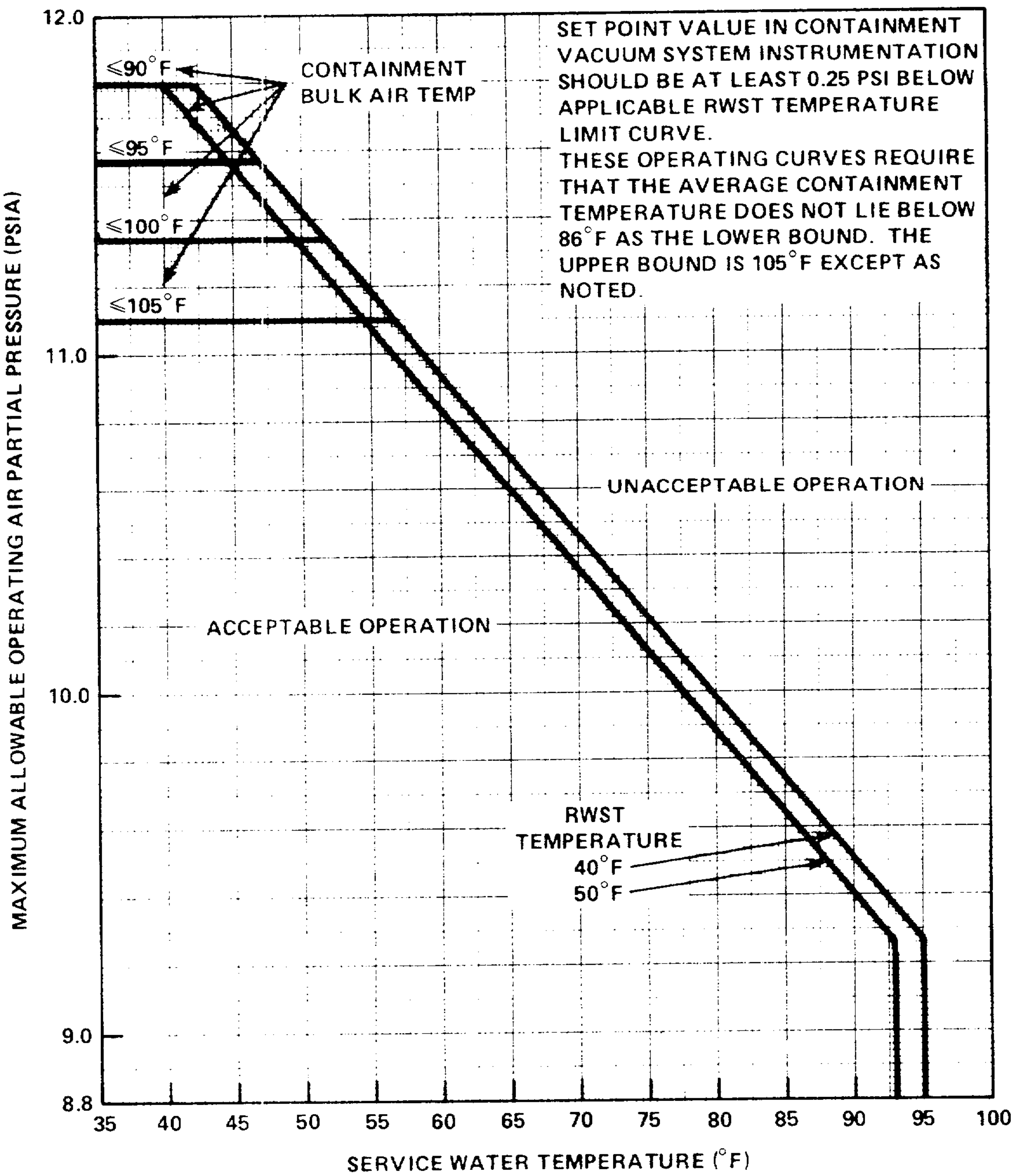

Figure 3.6-1 Maximum Allowable Primary Containment Air Partial Pressure Versus Service Water Temperature and RWST Water Temperature

SURVEILLANCE REQUIREMENTS (Continued)

d. At least once per 5 years by performing an air or smoke flow test through each spray header and verifying each spray nozzle is unobstructed.

CONTAINMENT RECIRCULATION SPRAY SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.2 The containment recirculation spray system shall be OPERABLE with:

a. Four separate and independent containment recirculation spray subsystems, each composed of a spray pump, associated heat exchanger and flow path.

b. Two separate and independent outside recirculation spray pump casing cooling subsystems, each composed of a casing cooling pump, and flow path capable of transferring fluid from the casing cooling tank to the suction of the outside recirculation spray pumps.

c. One casing cooling tank shall be OPERABLE with:

1. Contained borated water volume of at least 116,500 gallons.
2. Between 2000 and 2100 ppm boron concentration.
3. A solution temperature $\geq 35°$ and $\leq 50°F$.

APPLICABILITY: Modes 1, 2, 3 and 4.

ACTION:

a. With one containment recirculation spray subsystem or casing cooling subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours; restore the inoperable subsystem to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the next 30 hours.

b. With the casing cooling tank inoperable, restore the tank 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.

CONTAINMENT RECIRCULATION SPRAY SYSTEM

SURVEILLANCE REQUIREMENTS

4.6.2.2.1 Each containment recirculation spray subsystem and casing cooling subsystem 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. Verifying, that on recirculation flow, each outside recirculation spray pump develops a discharge pressure of $\geq$ 115 psig and each casing cooling pump develops a discharge pressure of __*__ when tested pursuant to Specification 4.0.5.

c. At least once per 18 months by:

1. Verifying that on a Containment Pressure--High-High signal, each casing cooling pump starts automatically without time delay, and each recirculation spray pump starts automatically with the following time delays: inside 195 $\pm$ 9.75 seconds, outside 210 $\pm$ 21 seconds.

2. Verifying that each automatic valve in 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 performing an air or smoke flow test through each spray header and verifying each spray nozzle is unobstructed.

4.6.2.2.2 The casing cooling tank shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the contained borated water volume in the tank, and

2. Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the casing cooling tank temperature.

*To be supplied following final testing.

TABLE 3.6-1 (Cont.)

VALVE NUMBER		FUNCTION	ISOLATION TIME (SEC.)
17.	TV-CV100*	Containment Air Ejector Suction	NA
18.	MOV-1869A*	High Head Safety Injection to RCS Except Boron Injection Line	NA
19.	MOV-1836*	High Head Safety Injection to RCS Except Boron Injection Line	NA
20.	MOV-1869B*	High Head Safety Injection to RCS Except Boron Injection Line	NA
21.	HCV-1142*	Reactor Coolant Letdown Line From RHR System	NA
22.	TV-SS107A*	Residual Heat Removal System Sample Lines	NA
23.	TV-SS107B*	Residual Heat Removal System Sample Lines	NA
24.	MOV-1890A*	LHSI Pump Discharge to Reactor Coolant System Hot Legs	NA
25.	MOV-1890B*	LHSI Pump Discharge to Reactor Coolant System Hot Legs	NA
26.	MOV-1890C*	LHSI Pump Discharge to Reactor Coolant System Cold Legs	NA
27.	MOV-1890D*	LHSI Pump Discharge to Reactor Coolant System Cold Legs	NA
28.	FCV-1160*	Loop Fill Header	NA
29.	MOV-1289A*	Charging Line	NA
30.	MOV-1867C*	High Head Safety Injection, Boron Injection Tank	NA
31.	MOV-1867D*	High Head Safety Injection, Boron Injection Tank	NA

TABLE 3.6-1 (Cont.)

VALVE NUMBER		FUNCTION	ISOLATION TIME (SEC.)
32.	MOV-RS-100A*	Casing Cooling to Outside Recirculation Spray Pump	NA
33.	MOV-RS-100B*	Casing Cooling to Outside Recirculation Spray Pump	NA
34.	MOV-RS-101A*	Casing Cooling to Outside Recirculation Spray Pump	NA
35.	MOV-RS-101B*	Casing Cooling to Outside Recirculation Spray Pump	NA

TABLE 3.6-1 (Cont.)

VALVE NUMBER		FUNCTION	ISOLATION TIME (SEC.)
F. CHECK			
1.	1-CC-193	Component Cooling Water to RHR System and Excess Letdown Heat Exchanger	NA
2.	1-CC-198	Component Cooling Water to RHR System and Excess Letdown Heat Exchanger	NA
3.	1-SI-79	High Head Safety Injection, Boron Injection to RCS	NA
4.	1-CC-572	Component Cooling Water to Containment Air Recirculation Coils	NA
5.	1-CC-559	Component Cooling Water to Containment Air Recirculation Coils	NA
6.	1-CC-546	Component Cooling Water to Containment Air Recirculation Coils	NA
7.	1-CH-322	Charging Line	NA
8.	1-CC-154	Component Cooling Water to Reactor Coolant Pumps	NA
9.	1-CC-119	Component Cooling Water to Reactor Coolant Pumps	NA
10.	1-CC-84	Component Cooling Water to Reactor Coolant Pumps	NA
11.	1-CH-402	Reactor Coolant Pumps, Seal Water Return	NA
12.	1-SI-110	Safety Injection Accumulator Make Up	NA
13.	1-SI-185	High Head Safety Injection to RCS except Boron Injection Line	NA

Table 3.7-4 (Continued)

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER No.	SYSTEM SNUBBER INSTALLED ON, LOCATION (ELEVATION-AREA)#	ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION ZONE** (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)
104B	SI-256-SG	A	No	No
105	SI-256-SG	A	No	No
106	SI-256-SG	A	No	No
100	SI-257-B	A	No	Yes
101	SI-257-B	I	No	Yes
101D	SI-257-B	I	No	Yes
101C	SI-257-B	I	No	Yes
102C	SI-221-B	A	No	No
103A	SI-238-C	I	Yes	Yes
103B	SI-238-C	I	Yes	Yes
103C	SI-238-C	I	Yes	Yes
104	SI-221-C	A	No	No

Table 3.7-4 (Continued)

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER No.	SYSTEM SNUBBER INSTALLED ON, LOCATION (ELEVATION-AREA)#	ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION ZONE** (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)
105A	SI-238-A	I	No	Yes
105B	SI-238-A	I	No	Yes
105C	SI-238-A	I	No	Yes
106A	SI-221-A	A	No	No
107	SI-256-SG	A	No	No
108	SI-256-SG	A	No	No
109	SI-250-A	I	Yes	No
110	SI-219-A	A	No	No
111	SI-241-1	A	No	No
112	SI-256-SG	A	No	No
113	SI-256-SG	A	No	No

3/4.1.2 BORATION SYSTEMS (Continued)

With the RCS average temperature above 200°F, a minimum of two separate and redundant boron injection systems are provided to ensure single functional capability in the event an assumed failure renders one of the systems inoperable. Allowable out-of-service periods ensure that minor component repair or corrective action may be completed without undue risk to overall facility safety from injection system failures during the repair period.

The boration capability of either system is sufficient to provide a SHUTDOWN MARGIN from expected operating conditions of 1.77% $\Delta k/k$ after xenon decay and cooldown to 200°F. This expected boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires 4450 gallons of 20,000 ppm borated water from the boric acid storage tanks or 70,000 gallons of 2000 ppm borated water from the refueling water storage tank.

With the RCS temperature below 200°F, one injection system is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity change in the event the single injection system becomes inoperable.

The boron capability required below 200°F is sufficient to provide a SHUTDOWN MARGIN of 1.77% $\Delta k/k$ after xenon decay and cooldown from 200°F to 140°F. This condition requires either 835 gallons of 20,000 ppm borated water from the boric acid storage tanks or 9690 gallons of 2000 ppm borated water from the refueling water storage tank.

The contained water volume limits include allowance for water not available because of discharge line location and other physical characteristics. The OPERABILITY of one boron injection system during REFUELING insures that this system is available for reactivity control while in MODE 6.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 8.5 and 11.0 for the solution recirculated within the containment after a LOCA. This pH minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corosion on mechanical systems and components.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) limit the potential effects of rod misalignment on associated accident analyses. OPERABILITY of the control rod position indicators is required to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original design criteria are met. Misalignment of a rod requires measurement of peaking factors or a restriction in THERMAL POWER; either of these restrictions provides assurance of fuel rod integrity during continued operation. In addition those accident analyses affected by a misaligned rod are reevaluated to confirm that the results remain valid during future operation.

Control rod positions and OPERABILITY of the rod position indicators are required to be verified on a nominal basis of once per 12 hours with more frequent verifications required if an automatic monitoring channel is inoperable. These verification frequencies are adequate for assuring that the applicable LCO's are satisfied.

The restriction prohibiting part length rod insertion ensures that adverse power shapes and rapid local power changes which may effect DNB considerations do not occur as a result of part length rod insertion during operation.

The maximum rod drop time restriction is consistent with the assumed rod drop time used in the accident analyses. Measurement with $T_{avg} \geq 500°F$ and with all reactor coolant pumps operating ensures that the measured drop times will be representative of insertion times experienced during a reactor trip at operating conditions.

BASES

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (a) maintaining the minimum DNBR in the core $\geq$ 1.30 during normal operation and in short term transients, and (b) limiting the fission gas release, fuel pellet temperature & cladding mechanical properties to within assumed design criteria. In addition, limiting the peak linear power density during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the ECCS acceptance criteria limit of 2200°F is not exceeded.

The definitions of certain hot channel and peaking factors as used in these specifications are as follows:

$F_Q(Z)$ Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods.

$F^N_{\Delta H}$ Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power.

$F_{xy}(Z)$ Radial Peaking Factor, is defined as the ratio of peak power density to average power density in the horizontal plane at core elevation Z.

3/4.2.1 AXIAL FLUX DIFFERENCE (AFD)

The limits on AXIAL FLUX DIFFERENCE assure that the $F_Q(Z)$ upper bound envelope of 2.21 times the normalized axial peaking factor is not exceeded during either normal operation or in the event of xenon redistribution following power changes.

Target flux difference is determined at equilibrium xenon conditions with the part length control rods withdrawn from the core. The full length rods may be positioned within the core in accordance with their respective insertion limits and should be inserted near their normal position for steady state operation at high power levels. The value of the target flux difference obtained under these conditions divided by the fraction of RATED THERMAL POWER is the target flux difference at RATED THERMAL POWER for the associated core burnup conditions. Target flux differences for other THERMAL POWER levels are obtained by multiplying the RATED THERMAL POWER value by the appropriate fractional THERMAL POWER level. The periodic updating of the target flux difference value is necessary to reflect core burnup considerations.

BASES

Although it is intended that the plant will be operated with the AXIAL FLUX DIFFERENCE within the $\pm$ 5% target band about the target flux difference, during rapid plant THERMAL POWER reductions, control rod motion will cause the AFD to deviate outside of the target band at reduced THERMAL POWER levels. This deviation will not affect the xenon redistribution sufficiently to change the envelope of peaking factors which may be reached on a subsequent return to RATED THERMAL POWER (with the AFD within the target band) provided the time duration of the deviation is limited. Accordingly, a 1 hour penalty deviation limit cumulative during the previous 24 hours is provided for operation outside of the target band but within the limits of Figure 3.2-1 while at THERMAL POWER levels between 50% and 85% of RATED THERMAL POWER. For THERMAL POWER levels between 15% and 50% of rated THERMAL POWER, deviations of the AFD outside of the target band are less significant. The penalty of 2 hours actual time reflects this reduced significance.

Provisions for monitoring the AFD on an automatic basis are derived from the plant process computer through the AFD Monitor Alarm. The computer determines the one minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for at least 2 of 4 or 2 of 3 OPERABLE excore channels are outside the target band and the THERMAL POWER is greater than 85% of RATED THERMAL POWER. During operation at THERMAL POWER levels between 50% and 85% and 15% and 50% RATED THERMAL POWER, the computer outputs an alarm message when the penalty deviation accumulates beyond the limits of 1 hour and 2 hours, respectively.

Figure B 3/4 2-1 shows a typical monthly target band.

a. abnormal perturbations in the radial power shape, such as from rod misalignment, effect $F^N_{\Delta H}$ more directly than F_Q,

b. although rod movement has a direct influence upon limiting F_Q to within its limit, such control is not readily available to limit $F^N_{\Delta H}$, and

c. errors in prediction for control power shape detected during startup physics tests can be compensated for in F_Q by restricting axial flux distributions. This compensation for $F^N_{\Delta H}$ is less readily available.

3/4.2.4 QUADRANT POWER TILT RATIO

The quadrant power tilt ratio limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during start-up testing and periodically during power operation.

The limit of 1.02 at which corrective action is required provides DNB and linear heat generation rate protection with x-y plane power tilts.

The two hour time allowance for operation with a tilt condition greater than 1.02 but less than 1.09 is provided to allow identification and correction of a dropped or misaligned rod. In the event such action does not correct the tilt, the margin for uncertainty on F_Q is reinstated by reducing the power by 3 percent for each percent of tilt in excess of 1.0.

3/4.2.5 DNB PARAMETERS

The limits on the DNB related parameters assure that each of the parameters are maintained within the normal steady state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR of 1.30 throughout each analyzed transient.

The 12 hour periodic surveillance of these parameters thru instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation. The 18 month periodic measurement of the RCS total flow rate is adequate to detect flow degradation and ensure correlation of the flow indication channels with measured flow such that the indicated percent flow will provide sufficient verification of flow rate on a 12 hour basis.

3/4.2.6 AXIAL POWER DISTRIBUTION

The limit on axial power distribution ensures that F_Q will be controlled and monitored on a more exact basis through use of the APDMS when operating above 95% of RATED THERMAL POWER. This additional limitation on F_Q is necessary in order to provide assurance that peak clad temperatures will remain below the ECCS acceptance criteria limit of 2200°F in the event of a LOCA.

vessel inside radius are essentially identical, the measured transition shift for a sample can be applied with confidence to the adjacent section of the reactor vessel. The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule is different from the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

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

The number of reactor vessel irradiation surveillance specimens and the frequencies for removing and testing these specimens are provided in Table 4.4-3 to assure compliance with the requirements of Appendix H to 10 CFR Part 50.

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

3/4.4.10 STRUCTURAL INTEGRITY

3/4.4.10.1 ASME CODE CLASS 1, 2 and 3 COMPONENTS

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

3/4.10.2 STEAM GENERATOR SUPPORTS

For the A572 material, operation above 225° provides a conservative temperature limit and thus toughness level in the steel. This assures the safety of the A572 material even under the worst postulated accident conditions. The points to be monitored were determined during hot functional testing, which indicated the top level corner lags the middle level corner during heatup; however, once the material achieved 225°F the top level corner exceeded the temperature of the middle level corner. The latter thus becomes the controlling zone during operation.

STEAM GENERATOR SUPPORTS (Continued)

For the monitored top level corner of the steam generator supports, operation below 355°F provides assurance that no other region of the supports will exceed this temperature. The monitored top level corner is the highest temperature region in the supports. With the temperature of the supports less than 355°F all materials will be within allowable stress limits even, under the worst postulated accident conditions.

g. Any other area of facility operation considered appropriate by the SyNSOC or the Vice President-Power Supply and Production Operations.

h. The Station 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 SyNSOC shall report to and advise the Vice President - Power Supply and Production Operations on those areas of responsibility specified in Sections 6.5.2.7 and 6.5.2.8.

RECORDS

6.5.2.10 Records of SyNSOC activities shall be prepared, maintained and dissesminated as indicated below within 14 working days of each meeting or following completion of the review or audit.

1. Senior Vice President-Power
2. Vice President-Power Supply and Production Operations
3. Nuclear Power Station Managers
4. Director Nuclear Operations
5. Members of the SyNSOC
6. Others that the Chairman of the SyNSOC may designate.

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 SNSOC and submitted to the SyNSOC and the Director of Nuclear Operations.

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The facility shall be placed in at least HOT STANDBY within one hour.

b. The Safety Limit violation shall be reported to the Commission, the Director, Nuclear Operations and to the SyNSOC within 24 hours.

c. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the SNSOC. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems or structures, and (3) corrective action taken to prevent recurrence.

d. The Safety Limit Violation Report shall be submitted to the Commission, the SyNSOC and the Director, Nuclear Operations within 14 days of the violation.

6.8 PROCEDURES

6.8.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, November, 1972.

b. Refueling operations.

c. Surveillance and test activities of safety related equipment.

d. Security Plan implementation.

e. Emergency Plan implementation.

f. Fire Protection Program Implementation.

6.8.2 Each procedure of 6.8.1 above, and changes thereto, shall be reviewed by the SNSOC and approved by the Station Manager prior to implementation and reviewed periodically as set forth in administrative procedures.

6.8.3 Temporary changes to procedures of 6.8.1 above may be made provided:

a. The intent of the original procedure is not altered.

b. The change is approved by two members of the plant supervisory staff, at least one of whom holds a Senior Reactor Operator's License on the unit affected.

c. The change is documented, reviewed by the SNSOC and approved by the Station Manager within 14 days of implementation.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS AND REPORTABLE OCCURRENCES

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Director of the Regional Office of Inspection and Enforcement unless otherwise noted.

STARTUP REPORTS

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant.

6.9.1.2 The startup report shall address each of the tests identified in the FSAR and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details requested in license conditions based on other commitments shall be included in this report.

6.9.1.3 Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial power operation), supplementary reports shall be submitted at least every three months until all three events have been completed.

ANNUAL REPORTS[1/]

6.9.1.4 Annual reports covering the activities of the unit as described below for the previous calendar year shall be submitted prior to March 1 of each year. The initial report shall be submitted prior to March 1 of the year following initial criticality.

6.9.1.5 Reports required on an annual basis shall include:

a. A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man-rem exposure according to work and job functions,[2/] e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignments to various duty functions may be estimated based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20 percent of the individual total dose need not be accounted for. In the aggregate, at least 80 percent of the total whole body dose received from external sources should be assigned to specific major work functions.

b. The complete results of the steam generator tube inservice inspections performed during the report period (Reference Specification 4.4.5.5.b.).

[1/] A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station.

[2/] This tabulation supplements the requirements of §20.407 of 10 CFR Part 20.

MONTHLY OPERATING REPORT

6.9.1.6 Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis to the Director, Office of Management and Program Analysis, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Regional Office, of Inspection and Enforcement, no later than the 15th of each month following the calendar month covered by the report.

REPORTABLE OCCURRENCES

6.9.1.7 The REPORTABLE OCCURRENCES of Specifications 6.9.1.8 and 6.9.1.9 below, including corrective actions and measures to prevent recurrence, shall be reported to the NRC. Supplemental reports may be required to fully describe final resolution of occurrence. In case of corrected or supplemental reports, a licensee event report shall be completed and reference shall be made to the original report date.

PROMPT NOTIFICATION WITH WRITTEN FOLLOWUP

6.9.1.8 The types of events listed below shall be reported within 24 hours by telephone and confirmed by telegraph, mailgram, or facsimile transmission to the Director of the Regional Office, or his designate no later than the first working day following the event, with a written followup report within 14 days. The written followup report shall include, as a minimum, a completed copy of a licensee event report form. Information provided on the licensee event report form shall be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

a. Failure of the reactor protection system or other systems subject to limiting safety-system settings to initiate the required protective function by the time a monitored parameter reaches the setpoint specified as the limiting safety-system setting in the technical specifications or failure to complete the required protective function.

NORTH ANNA POWER STATION, UNIT NO. 1 - MAY 19 1978

AMENDMENT NO. 5 TO NPF - 4

Distribution w/enclosure:
Docket File
NRC PDR
Local PDR (2)
LWR #3 File
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