AUG 2 8 1975

Docket No. 50-324

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Carolina Power & Light Company
ATTN: Mr. J. A. Jones
Executive Vice President
336 Fayetteville Street
P.O. Box 1551
Raleigh, North Carolina 27602

Amendment No. 5
Change No. 5
License No. DPR-62

Gentlemen:

The Commission has issued the enclosed Amendment No. 5 to Facility License No. DPR-62 for the Brunswick Steam Electric Plant, Unit 2. This amendment includes Change No. 5 to the Technical Specifications and is in response to Carolina Power & Light Company's request dated May 9, 1975, as supplemented by letters dated July 11, 22, 28, 1975.

This amendment incorporates operating limits in the Technical Specifications for the facility based on an acceptable evaluation model that conforms with the requirements of Section 50.46 of 10 CFR Part 50. In addition this amendment incorporates operating limits in the Technical Specifications based on the new General Electric Thermal Analysis Basis (GETAB).

The Commission's staff has evaluated the potential for environmental impact associated with operation of Brunswick Steam Electric Plant, Unit 2 in the proposed manner. From this evaluation, the staff has determined that there will be no change in effluent types or total amounts, no change in authorized power level and no significant environmental impact attributable to the proposed action. Having made this determination, the Commission has further concluded pursuant to 10 CFR Section 51.5(c)(1) that no environmental impact statement need be prepared for this action. Copies of the related Negative Declaration and supporting Environmental Impact Appraisal are enclosed. As required by 10 CFR Part 51, the Negative Declaration is being filed with the Office of the Federal Register for publication.

bcc: J. R. Buchanan, ORNL
Thmmas B. Abernathy, DTIE
A. Rosenthal, ASLAB
N. H. Goodrich, ASLBP

OFFICE ➤						
SURNAME ➤						
DATE ➤						

Form AEC-318 (Rev. 9-53) AECM 0240 ☆ U. S. GOVERNMENT PRINTING OFFICE: 1974-526-166

AUG 2 8 1975

Copies of the related Safety Evaluation, the Federal Register Notice and Amendment No. 5 with Technical Specification Change No. 5 are enclosed.

Sincerely,

Original signed by
Walter Butler

Walter R. Butler, Chief
Light Water Reactors Branch 1-2
Division of Reactor Licensing

Enclosures:
1. Amendment No. 5
   w/Change No. 5
2. Negative Declaration with Supporting
   Environmental Impact Appraisal
3. Safety Evaluation
4. Federal Register Notice

cc: See page 3

OFFICE ➤	L:LWR 1-2	L:LWR 1-2	OELD	L:LWR 1-2		
SURNAME ➤	MRushbrook/red	RPowell	STRIDIRON	WRButler WB		
DATE ➤	7791 8/22/75	8/22/75	8/27/75	8/28/75	8/28/75	8/28

Form AEC-318 (Rev. 9-53) AECM 0240 ☆ U. S. GOVERNMENT PRINTING OFFICE: 1974-526-166

AUG 2 8 1975

cc: Richard E. Jones, Esq.
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Chairman, Board of County
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OFFICE ➤						
SURNAME ➤						
DATE ➤						

CAROLINA POWER & LIGHT COMPANY
DOCKET NO. 50-324
BRUNSWICK STEAM ELECTRIC PLANT, UNIT 2
AMENDMENT TO FACILITY LICENSE

Amendment No. 5
License No. DPR-62

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

   A. The application for amendment by Carolina Power and Light Company (the licensee) dated May 9, 1975, and supplements dated July 11, 22, and 28, 1975, 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 application, 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.

2. Accordingly, the license is amended by a change to the Technical Specifications as indicated in the attachment to this license amendment and paragraphs 2C.(2), 2C.(4) and 2C.(5) of Facility License No. DPR-62 are hereby amended and added (respectively) to read as follows:

   2C.(2) Technical Specifications
   The Technical Specifications contained in Appendices A and B, as revised, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications, as revised by issued changes thereto through Change No. 5

   2C.(4) Equalizer Valve Restriction
   The valves in the equalizer piping between the recirculation loops shall be closed at all times during reactor operation.

2C.(5) Recirculation Loop Inoperable
The reactor shall not be operated with one recirculation loop out of service.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

R. S. Boyd, Acting Director
Division of Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Change No. 5 to the
Technical Specifications

Date of Issuance: August 28, 1975

NEGATIVE DECLARATION

REGARDING PROPOSED CHANGES TO THE

TECHNICAL SPECIFICATIONS OF LICENSE DPR-62

BRUNSWICK STEAM ELECTRIC PLANT UNIT 2

DOCKET NO.: 50-324

The Nuclear Regulatory Commission (the Commission) has considered the issuance of changes to the Technical Specifications of Facility Operating License No. DPR-62. These changes would authorize the Carolina Power and Light Company (CP&L) (the licensee) to operate the Brunswick Steam Electric Plant Unit 2 (located near the town of Southport, Brunswick County, North Carolina) with changes to the limiting conditions for operation resulting from application of the Acceptance Criteria for Emergency Core Cooling System (ECCS) in conjunction with a reactor core using 7 x 7 fuel in Unit 2.

The U. S. Nuclear Regulatory Commission, Division of Reactor Licensing, has prepared an environmental impact appraisal for the proposed changes to the Technical Specifications of License No. DPR-62, Brunswick Unit 2, described above. On the basis of this appraisal, the Commission has concluded that an environmental impact statement for this particular action is not warranted because there will be no environmental impact attributable to the proposed action other than that which has already been predicted and described in the Commission's Final Environmental Statement for Brunswick Units 1 and 2 published in January 1974. The environmental impact appraisal is available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C., and at Southport-Brunswick County Library, 109 W. Moore Street, Southport, North Carolina.

Dated at Rockville, Maryland, this 16th day of July.

FOR THE NUCLEAR REGULATORY COMMISSION

Gordon K. Dicker

Gordon K. Dicker, Chief
Environmental Projects Branch 2
Division of Reactor Licensing

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

JUL 1 6 1975

ENVIRONMENTAL IMPACT APPRAISAL BY THE DIVISION OF REACTOR LICENSING

SUPPORTING AMENDMENT NO. 5 TO DPR-62

CHANGE NO. 5 TO THE TECHNICAL SPECIFICATIONS

CAROLINA POWER AND LIGHT COMPANY

BRUNSWICK STEAM ELECTRIC PLANT UNIT 2

ENVIRONMENTAL IMPACT APPRAISAL

1. Description of Proposed Action

By letters dated May 9, 1975, and July 11, 1975, the Carolina Power and Light Company (CP&L) submitted proposed changes to the Technical Specifications Appendix A to License No. DPR-62. The proposed changes were requested to incorporate limiting conditions for operation associated with fuel assembly specific power (average planar linear heat generation rate) resulting from the application of the Acceptance Criteria for Emergency Core Cooling System (ECCS) in conjunction with a core using 7 x 7 fuel. The staff has independently reviewed this matter and the conclusions are set forth below.

The licensee is presently licensed to possess and operate Brunswick Steam Electric Plant Unit 2 located in Brunswick County, North Carolina, at power levels up to 2,436 megawatt thermal (MWt) using a full core of 7 x 7 fuel (containing U-235). The proposed change to incorporate the ECCS Acceptance Criteria does not result in an increase or decrease in power levels of the unit. The restrictions on heat generation rates will require careful control of fuel operating history. However, there should be no reduction on total burnup resulting from the revised ECCS evaluation methods. Since neither power level nor fuel burnup is affected by the action, the action does not affect the benefits of electric power production considered for the captioned facility in the Commission's Final Environmental Statement (FES) for Brunswick Steam Electric Plant, Docket Nos. 50-324 and 50-325 dated January 1974.

AMERICAN REVOLUTION BICENTENNIAL 1776-1976

2. Environmental Impacts of Proposed Action

Potential environmental impacts associated with the proposed action are those which may be associated with incorporation of the ECCS Acceptance Criteria and utilization of nuclear fuel for this facility.

It is particularly noted that in the absence of any significant change in power levels, there will be no change in cooling water requirements and consequently no increase in environmental impact from radioactive effluents and thermal effluents for normal operation or post-accident conditions which in turn could not lead to significant increases in radiation doses or thermal stress to the public or to biota in the environment.

For normal operating conditions, no environmental impact other than as described in the Commission's Final Environmental Statement (FES) for Brunswick Steam Electric Plant, Docket Nos. 50-324 and 50-325 dated January 1974, can be predicted for the proposed action. The Commission's calculated releases for radioactive effluents, both gaseous and liquid, are based on expected release rates to the environment and are quantified on the basis of the total quantity of nuclear fuel within the reactor. The estimates of radionuclides and release rates will not be affected by the proposed action, and since the total quantity of nuclear fuel is unchanged, no increase in the calculated release of radioactive effluents is predicted. Consequently, no increases in radiation doses to man or other biota are predicted.

3. Conclusion and Basis for Negative Declaration

On the basis of the foregoing analysis, it is concluded that there will be no environmental impact attributable to the proposed action other than has already been predicted and described in the Commission's FES for Brunswick Steam Electric Plant Units 1 and 2. Having made this conclusion, the Commission has further concluded that no environmental impact statement for the proposed action need be prepared and that a negative declaration to this effect is appropriate.

DATE: JUL 1 6 1975

SAFETY EVALUATION

of the

BRUNSWICK UNIT 2

GETAB AND ECCS APPENDIX K ANALYSIS

BY

Reactor Systems Branch

August 1975

TABLE OF CONTENTS

SER

BRUNSWICK UNIT 2 - INITIAL CORE

SAFETY EVALUATION REPORT

1.0 Introduction

The licensee has submitted the analyses supporting the proposed GETAB-based Technical Specifications and the loss-of-coolant accident analysis in conformance to Appendix K of 10 CFR Part 50. The analyses are based on the initial core loading of the Brunswick Unit 2 reactor with 7 x 7 fuel. The licensee submitted information consisting of a main letter dated May 9, 1975, and of supporting letters dated July 11, 1975, July 22, 1975, and July 28, 1975. We have reviewed the submitted information and report our safety evaluation herein.

2.0 Evaluation

2.1 GETAB

To apply GETAB to the Technical Specifications involves 1) establishing the fuel damage safety limit, 2) establishing limiting conditions of operation such that the safety limit is not exceeded for normal operation and anticipated transients, and 3) establishing limiting conditions for operation such that the initial conditions assumed in accident analyses are satisfied. We have evaluated and report herein the thermal margins developed for Brunswick-2 based on the generic NEDO-10958 report[1] and the plant specific input information provided by the licensee.

2.1.1 Fuel Cladding Integrity Safety Limit MCPR

The fuel cladding integrity safety limit, the minimum critical power ratio or MCPR, for the 7 x 7 fuel is 1.05. It is based on the GETAB statistical analysis which assures that 99.9% of the fuel rods in the core are expected to avoid boiling transition. The uncertainties in the core and system operating parameters and the GEXL correlation, Table 1 of the licensee submittal,[2] combined with the relative bundle power distribution in the core form the basis for the GETAB statistical determination of the safety limit MCPR. The tabulated lists of uncertainties for Brunswick Unit 2 are the same as those reported in NEDO-10958 and NEDO-20340[3] which are acceptable.

The reactor core selected for the GETAB statistical analyses that incorporates the operating parameters, fuel design (R factor*), and GEXL correlation uncertainties is a typical 251/764 core. This selected core is under the same reactor class as is the Brunswick Unit 2 core and it is larger. Thus, the GETAB analysis results provide a fuel cladding integrity safety limit MCPR of 1.05 which is conservatively applied to the Brunswick Unit 2 reactor. Comparison of the licensee submitted bundle power distributions(4) used for the GETAB application with that of the actual operation planned for the Brunswick Unit 2 reactor illustrates that the use of more high power bundles in the GETAB analysis indicates that the calculated safety limit MCPR based on the 99.9% statistical criteria is a conservative value.

We conclude that the proposed fuel integrity safety limit, a MCPR of 1.05, is acceptable to prevent fuel damage for the current fuel cycle for Brunswick Unit 2.

2.1.2 Operating Limit MCPR

Various transient events will reduce the required operating limit MCPR. To assure that the fuel cladding integrity safety limit (MCPR of 1.05) is not exceeded during anticipated abnormal operational transients, the most limiting transients have been analyzed to determine which one results in the largest reduction in critical power ratio (ΔMCPR). The licensee has submitted the

*The R factor is a parameter which characterizes the local peaking pattern with respect to the most limiting rod.

results of the transient analyses which lead to significant decreases in MCPR. The types of transients evaluated were loss of flow, pressure and power increase, and coolant temperature decrease. The more limiting transients in the stated categories were 2-pump trip, load rejection without bypass, and loss of feedwater heating. Of these three, the most limiting transient was load rejection without bypass, which results in a ΔMPCR of 0.17. Addition of this ΔMCPR to the safety limit MCPR gives the minimum operating limit MCPR of 1.22 which is required to avoid violation of the safety limit, should this limiting transient occur.

The transient analyses were evaluated with the end-of-cycle scram reactivity insertion rates that include a design conservatism factor of 0.80. The initial condition parameters[2] used for the worst operational transient analysis as submitted by the licensee are acceptable. The initial MCPR assumed in the transient analyses was equal to or greater than the established operating limit MCPR of 1.22.

Conservatism was applied in the determination of the required operating limit MCPR because the axial and local peaking were assumed to take place at the beginning of the fuel cycle and the peak of the axial power shape was assumed to occur in the mid-plane (node 12; APF of 1.40). This is the worst consistent set of parameters that is supported by a GE study[1] which has shown the required operating MCPR to be a function of the location of axial peak, and the required MCPR's are essentially independent of peak location for axials peaked in the middle and upper portions of the core whereas for bottom peaked axials the required MCPR is reduced.

The applied R factor of 1.084 for 7 x 7 fuel is taken at the beginning of cycle to reasonably bound the expected operating conditions. During the cycle the local peaking and therefore the R factor is reduced while the peak in the axial shape moves toward the bottom of the core. Although the operating limit MCPR would be increased by approximately 1% of the reduced end-of-cycle R factor, this is offset by the reduction in MCPR resulting from the relocation of the axial peak to below the midplane.

2.1.3 Correction to the Operating Limit MCPR Due to Change in Void Coefficient

The required minimum operating limit MCPR of 1.22 was based on the addition of the largest ΔMCPR (caused by the load rejection without bypass transient) to the safety limit MCPR of 1.05, in which we found this to be acceptable. However, due to the change in method of calculating void reactivity coefficients (Neutron Effective Voids [NEV]) --where the new method provides better agreement between the calculated and plant instrument power distributions--a relative change in ΔMCPR was also affected by corresponding changes in void coefficient values. GE performed a generic void coefficient sensitivity study on a 251 size BWR4 plant, and found that relative changes in ΔMCPR due to a change in void coefficient was most sensitive to a generator load rejection with failure to bypass transient. The licensee complied with the conclusions of this neutron effective void correction analysis by applying the ΔMCPR correction for the load rejection without bypass transient[2]. The resultant corrected largest ΔMCPR was 0.213 and therefore the minimum required operating limit MCPR will have to be 1.27. We find this to be acceptable.

The calculated change in MCPR for the second most severe abnormal operational transient--the loss of feedwater heating--was 0.15 for 7 x 7 fuel without the neutron effective void correction (NEV). Since the corrective change in ΔMCPR due to NEV for the loss of feedwater heating transient is less sensitive than the load rejection without bypass transient, the largest change in ΔMCPR (assuming NEV) was based on the latter transient. (At the staff's request, GE will provide at a later date a sensitivity study of relative change in ΔMCPR due to change in void coefficients for the loss of feedwater heating transient.)

2.1.4 Rod Withdrawal Error Transient

The licensee discussed the rod withdrawal error transient in terms of worst case conditions.[4] The analysis shows that the local power range monitor subsystem (LPRM's) will detect high local powers and alarm. However, if the operator ignores the LPRM alarm, the rod block monitor subsystem (RBM) will stop rod withdrawal while the critical power ratio is still greater than the 1.05 MCPR safety limit, and the cladding is under the one percent plastic strain limit. We conclude that the consequences of this localized transient are acceptable.

2.1.5 Operating MCPR Limits for Less than Rated Power and Flow

For the limiting transient of recirculation pump speed control failure at lower than rated power and flow condition, the licensee will conform to Technical Specification limiting conditions for operation, Paragraph 3.1C. This requires the licensee to maintain the required operating MCPR greater than 1.27 times K_f factor for core flows less than rated. The K_f factor curves were generically derived which assures that the most limiting transient occurring at less than rated flow will not exceed the safety limit MCPR of 1.05.

We conclude that the calculated consequences of the anticipated abnormal transients do not violate the thermal and plastic strain limits of the fuel or the pressure limits of the reactor coolant boundary.

2.2 ECCS Appendix K Analysis

On December 27, 1974, the then Atomic Energy Commission issued an Order for Modification of License implementing the requirements of 10 CFR 50.46 "Acceptance Criteria and Emergency Core Cooling Systems for Light Water Nuclear Power Reactors." One of the requirements of the Order was that prior to any license amendment authorizing any core reloading"...the licensee shall submit a reevaluation of ECCS cooling performance calculated in accordance with an acceptable evaluation model which conforms with the provisions of 10 CFR Part 50, 50.46." The Order also required that the evaluation shall be accompanied by such proposed changes in Technical Specifications or license amendments as may be necessary to implement the evaluation results.

On May 9, 1975 the licensee submitted an evaluation of the ECCS performance for the design basis piping break for Brunswick Unit 2 along with an amendment requesting changes to the Technical Specifications for Brunswick Unit 2 to implement the results of the evaluation. The licensee incorporated further information relating to the details of the ECCS evaluation as an appropriate lead plant analysis by letters dated July 11, 1975 and July 28, 1975, to show compliance with the 10 CFR 50.46 criteria and Appendix K to 10 CFR Part 50.

The Order for Modifying of License issued December 27, 1974, stated that evaluation of ECCS cooling performance may be based on the vendor's evaluation model as modified in accordance with the changes described in the staff Safety Evaluation Report of the Dresden Nuclear Power Station dated December 27, 1974.

The background of the staff review of the General Electric (GE) ECCS models and of their application to Brunswick Unit 2 are described in the staff Safety Evaluation Report (SER) dated December 27, 1974, issued in connection with the Order. The bases for acceptance of the principal portions of the evaluation model are set forth in the staff's Status Report of October 1974 which are referenced in the December 27, 1974 SER. The December 27, 1974 SER also describes the various changes required in the earlier GE evaluation model. The December 27, 1974 SER and the Status Report with its Supplement, describes an acceptable ECCS evaluation model and the basis for the staff's acceptance of the model. The Brunswick Unit 2 evaluation which is covered by this SER properly conforms to the accepted model.

With respect to reflood and refill computations, the Brunswick Unit 2 analysis was based on a modified version of the SAFE computer code, with explicit consideration of the staff recommended limitations. There are described on pages 7 and 8 of the December 27, 1974 SER. The Brunswick evaluation did not attempt to include any further credit for other potential changes which the December 27, 1974 SER indicated were under consideration by GE at that time.

During the course of our review, we concluded that additional individual break sizes should be analyzed to substantiate the break spectrum curves submitted in connection with the evaluation provided in August 1974.

We also requested that other break locations be studied to substantiate that the limiting break location was the recirculation line.

The additional analyses supported the earlier submittal which concluded that the worst break was the complete severence of the recirculation line. These additional calculations provided further details with regard to the limiting location and size of break as well as worst single failure for the Brunswick Unit 2 design. The limiting break which is the design basis accident is the complete severence of the recirculation discharge line assuming a failure of the LPCI injection valve.

We have reviewed the evaluation of ECCS performance submitted by Carolina Power and Light Company for Unit 2 and conclude that the evaluation has been performed wholly in conformance with the requirements of 10 CFR 50.46(a). Therefore, operation of the reactor satisfies the requirements of 10 CFR 50.46 provided that operation is limited to the maximum average planar linear heat generation rates (MAPLHGR) of figures D5A and D5B of the Carolina Power and Light Company letter dated July 11, 1975, and to a minimum critical power ratio (MCPR) greater than 1.18.

However, certain changes must be made to the proposed technical specifications for conformance with the evaluation of ECCS performance. The largest recirculation break area assumed in the evaluation was 4.2 square feet. This break size is based on operation with a closed valve in the equalizer line between the two recirculation loops. Therefore, a license condition must be added which prohibits reactor operation unless the valve in the equalizer line is closed.

The ECCS performance analysis assumed that reactor operation will be limited to a MCPR of 1.18. However, a more limiting technical specification limits operation of the reactor to a MCPR of 1.27 for 7 x 7 fuel based on consideration of a turbine trip transient with failure of bypass valves. A statement must be added to the bases for the MCPR limiting condition of operation indicating that the MCPR value used in the ECCS performance evaluation has been appropriately considered.

The Technical Specifications should require the licensee to report as an abnormal occurrence, any operation in excess of the limiting MAPLHGR values, even if corrective action was taken upon discovery. We believe that such events should be reported in conformity with the Technical Specifications.

An evaluation was not provided for ECCS performance during reactor operation with one recirculation loop out of service. Therefore, reactor operation under such conditions should not be authorized until the necessary analyses have been performed, evaluated and determined to be acceptable.

The LOCA analysis assumed all ADS valves operated for small line breaks with HPCI failure. Since the licensee did not provide a LOCA analysis with one ADS valve out of service for small line breaks we require that the Technical Specifications be modified so as not to allow operation for more than seven days with any ADS valve out of service. The HPCI must be operable whenever any of the ADS valves is out of service.

3.0 Conclusions

We conclude that the submitted safety analyses of abnormal operational transients for Brunswick Unit 2 are acceptable. The proposed minimum operating limit MCPR established for Brunswick Unit 2 that is required to avoid violation of the Safety Limit MCPR, should the most limiting transient occur, is acceptable.

The licensee submitted ECCS LOCA analysis is in conformance to the requirements of Appendix K to 10 CFR Part 50. The reactor operation restrictions based on the submitted analysis are noted in this report.

References

1. "General Electric BWR Thermal Analysis Basis (GETAB) Data Correlation and Design Application," NEDO-10958 and NEDE-10958.

2. "Brunswick Steam Electric Plant, Unit No. 2, License No. DPR-62, Thermal Hydraulic Analysis", submitted by letter dated July 22, 1975.

3. General Electric "Process Computer Performance Evaluation Accuracy," NEDO-20340, and Amendment 1, NEDO-20340-1, dated June 1974 and December 1974, respectively.

4. "Brunswick Steam Electric Plant, Unit No. 2, License No. DPR-62, 10 CFR Part 50 Appendix K Calculations and Revised Technical Specifications", submitted by letter dated May 9, 1975.

UNITED STATES NUCLEAR REGULATORY COMMISSION
DOCKET NO. 50-324
CAROLINA POWER AND LIGHT COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO
FACILITY OPERATING LICENSE

Notice is hereby given that the U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 5 to Facility Operating License No. DPR-62 issued to Carolina Power and Light Company which revised Technical Specifications for operation of the Brunswick Steam Electric Plant Unit 2 located in Brunswick County, North Carolina. The amendment is effective as of its date of issuance.

The amendment incorporates operating limits in the Technical Specifications for the facility (1) based on an acceptable evaluation model that conforms with the requirements of Section 50.46 of 10 CFR Part 50 and (2) based on the new General Electric Thermal Analysis Basis in accordance with the Carolina Power and Light Company's request dated May 9, 1975.

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. Notice of Proposed Issuance of Amendment to Facility Operating License in connection with this action was published in the FEDERAL REGISTER on June 12, 1975 (40FR 25108). No request for a hearing or petition for leave to intervene was filed following notice of the proposed action.

For further details with respect to this action, see (1) the application for amendment dated May 9, 1975 and supplements thereto dated July 11, 22, 28, 1975, (2) Amendment No. 5 to License No. DPR-62 with change No. 5, to the Technical Specifications (3) the Commission's concurrently issued related Safety Evaluation, and (4) the Commission's Negative Declaration dated July 16, 1975 (which is also being published in the FEDERAL REGISTER) and associated Environmental Impact Appraisal.

All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W. Washington, D. C. and at the Southport-Brunswick County Library, 109 W Moore Street, Southport, North Carolina 28461. A single copy of items (2), (3) and (4) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Reactor Licensing.

Dated at Bethesda, Maryland this 28th day of August, 1975.

FOR THE NUCLEAR REGULATORY COMMISSION

Walter R. Butler

Walter R. Butler, Chief
Light Water Reactors Branch 1-2
Division of Reactor Licensing

TECHNICAL SPECIFICATIONS

CONTENTS

Change #5

AUGUST 1975

CONTENTS (Continued)

TABLES (Continued)

TECHNICAL SPECIFICATIONS

FIGURES

5

Change #5

to 105 percent of rated reactor power. The stated design power in megawatts thermal (MWt) is the result of a heat balance for a particular plant design. Design power for the Brunswick Steam Electric Plant is 2531 MWt (see also "Rated Power" definition).

N. Primary Containment Integrity - Primary containment integrity means that the drywell and pressure suppression chamber are intact and all of the following conditions are satisfied:

1. All manual containment isolation valves on lines connected to the reactor coolant system or opened to containment atmosphere which are not required to be open during normal operations are closed.

2. At least one door in each airlock is closed and sealed.

3. All automatic containment isolation valves are operable or deactivated in the isolated position.

4. All blind flanges and manways are closed.

O. Secondary Containment Integrity - Secondary containment integrity means that the Reactor Building is intact and the following conditions are met:

1. At least one door in each access opening is closed.

2. The standby gas treatment system is operable as specified in Subsection 3.7.B.

3. All automatic ventilation system isolation valves are operable or secured in the isolated position.

P. Operating Cycle - Interval between the end of one refueling outage and the end of the next subsequent refueling outage.

Q. Refueling Outage - Refueling outage is the period of time between shutdown of the Unit prior to refueling and startup of the Unit after that refueling. For the purpose of designating frequency of testing and surveillance, a refueling outage shall mean a regularly scheduled outage; however, where such outages occur within eight months of the completion of the previous refueling outage, the required surveillance testing need not be performed until the next regularly scheduled outage.

R. Alteration of the Reactor Core - The act moving any component in the region above the core support plate, below the upper gride and within the shroud. Normal control rod movement with the control rod drive hydraulic system is not defined as a core alteration. Normal movement of incore instrumentation is not defined as a core alteration.

S. Reactor Vessel Pressure - Unless otherwise indicated, reactor vessel pressures listed in the Technical Specifications are those measured by the reactor vessel steam space detectors.

T. Thermal Parameters

1. Minimum Critical Power Ratio (MCPR) - The value of the critical power ratio associated with the most limiting assembly in the reactor core. The critical power ratio of that power in a fuel assembly, which is calculated to cause some point in the assembly to experience boiling transition, to the actual assembly operating power. Change #5

2. Transition Boiling - Transition boiling means the boiling regime between nucleate and film boiling. Transition boiling is the regime in which both nucleate and film boiling occur intermittently with neither type being completely stable.

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTINGS

1.1 Fuel Cladding Integrity

Applicability:

Applies to the interrelated variables associated with fuel thermal behavior.

Objective:

To establish limits below which the integrity of the fuel cladding is preserved.

Specification:

A. When the reactor pressure is equal to or greater than 800 psia or core flow $\geq$ 10%, the minimum critical power ratio shall be $\geq$ 1.05.

Change #5

2.1 Fuel Cladding Integrity

Applicability:

Applies to trip settings of the instruments and devices which are provided to prevent the reactor system safety limits from being exceeded:

Objective:

To define the level of the process variables at which automatic protective action is initiated to prevent the fuel cladding integrity safety limits from being exceeded.

Specification:

The limiting safety system settings shall be as specified below:

A. Neutron Flux Scram

1. APRM - The APRM scram trip setpoint shall be as shown on Figure 2.1-1 and shall be:

$$S \leq (0.66W + 54) \frac{(2.60)}{MTPF}$$

with a maximum setpoint of 120 percent for core flow equal to 78.5 million lb/hr and greater.

where:

S = Setting in percent of rated power (2436 MWt)

W = Recirculation loop flow in percent of design

MTPF = Maximum Total Peaking Factor, 2.60 unless the combination of power and peak heat flux is above the curve in Figure 2.1-2 at which point the actual value of MTPF shall be used. 5

Change #5

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTINGS

1.1 Fuel Cladding Integrity (Cont'd)

B. When the reactor pressure is less than 800 psia, or core cooling flow is less than 10 percent of design, the reactor thermal power shall not exceed 25% rated power.

Change #5

To ensure that the safety limit established in Specification 1.1.A and 1.1.B is not exceeded, each required scram shall be initiated by its primary source signal. The safety limit shall be assumed to be exceeded when scram is accomplished by a means other than the primary source signal.

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2.1.A Neutron Flux Scram (Cont'd)

2. APRM - When the reactor mode switch is in the STARTUP position, the APRM scram shall be set at less than or equal to 15 percent of rated power.

3. IRM - the IRM flux scram setting shall be $\leq$ 120/125 of scale.

B. APRM Control Rod Block

The APRM Control Rod Block trip set point(s) shall be biased with flow as shown on Figure 2.1-1 and shall be less than or equal to:

$$S \leq (0.66W + 42)\ \frac{2.60}{MTPF}$$

5

The definitions used above for the APRM scram trip apply.

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING
1.1 Fuel Cladding Integrity (Cont'd)	2.1 Fuel Cladding Integrity (Cont'd)
C. Whenever the reactor is in the cold shutdown condition with irradiated fuel in the reactor vessel, the water level shall not be less than 18 inches above the top of the normal active fuel zone.	C. Reactor low water level #1 scram setting shall be ≥ 12.5" on level instruments. D. Turbine stop valve closure scram setting shall be ≤ 10 percent valve closure except that this is bypassed when power ≤ 30 percent. E. Turbine control valve 1. Fast closure - Results from low hydraulic oil pressure. 2. Loss of control oil pressure - setting shall be ≥ 850 psig. 3. For Brunswick Unit No. 2 - fast closure will initiate select rod insert but will not initiate a reactor protection system trip prior to determination of turbine bypass valve status. If the by-pass valves do not open, the reactor protection system will scram the reactor. F. Main steam isolation scram setting shall be ≤ 10 percent valve closure. G. Main steam isolation on main steam line low pressure at inlet to turbine valves. Pressure setting shall be ≥ 850 psig. H. Reactor low water level #3 initiation of LPCI, core spray and auto blow-down shall be set at or above -147.5 inches indicated level. I. Reactor low water level #2 initiation of HPCI and RCIC shall be set at or above -38 inches indicated level.

BASES:

1.1 FUEL CLADDING INTEGRITY SAFETY LIMIT

The fuel cladding integrity limit is set such that no calculated fuel damage would occur as a result of an abnormal operational transient. Because fuel damage is not directly observable, a step-back approach is used to establish a safety limit such that the minimum critical power ratio (MCPR) is no less than 1.05. MCPR > 1.05 represents a conservative margin relative to the conditions required to maintain fuel cladding integrity.

Change #5

The fuel cladding represents one of the physical barriers which separate radioactive materials from environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use-related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses which occur from reactor operation significantly above design conditions and the protection system safety settings. While fission product migration from cladding perforation is just as measurable as that from use-related cracking, the thermally-caused cladding perforations signal a threshold, beyond which still greater thermal stresses may cause gross rather than incremental cladding deterioration. Therefore, the fuel cladding safety limit is defined with margin to the conditions which would produce onset of transition boiling (MCPR of 1.0). These conditions represent a significant departure from the condition intended by design for planned operation.

5

Change #5

Onset of transition boiling results in a decrease in heat transfer from the clad and, therefore, elevated clad temperature and the possibility of clad failure. However, the existence of critical power, or boiling transition, is not a directly observable parameter in an operating reactor. Therefore, the margin to boiling transition is calculated from plant operating parameters such as core power, core flow, feedwater temperature, and core power distribution. The margin for each fuel assembly is characterized by the critical power ratio (CPR) which is the ratio of the bundle power which would produce onset of transition boiling divided by the actual bundle power. The minimum value of this ratio for any bundle in

BASES:

1.1 FUEL CLADDING INTEGRITY SAFETY LIMIT (Cont'd)

the core is the minimum critical power ratio (MCPR). It is assumed that the plant operation is controlled to the nominal protective setpoints via the instrumented variables, i.e., normal plant operation presented on Figure 1.1-1 by the nominal expected flow control line. The safety limit (MCPR of 1.05) has sufficient conservatism to assure that in the event of an abnormal operational transient initiated from a normal operating condition (MCPR > 1.27) more than 99.9% of the fuel rods in the core are expected to avoid boiling transition The margin between MCPR of 1.0 (onset of transition boiling) and the safety limit of 1.05 is derived from a detailed statistical analysis considering all of the uncertainties in monitoring the core operating state including uncertainty in the boiling transition correlation as described in Reference 1. Change #5

Because the boiling transition correlation is based on a large quantity of full scale data, there is a very high confidence that operation of a fuel assembly at the condition of MCPR = 1.05 would not produce boiling transition.

However, if boiling transition were to occur, clad perforation would not necessarily be expected. Cladding temperatures would increase to approximately 1100 F which is below the perforation temperature of the cladding material. This has been verified by tests in the General Electric Test Reactor (GETR) where fuel similar in design to Brunswick operated above the critical heat flux for a significant period of time without clad perforation.

If reactor pressure should ever exceed 1400 psia during normal power operation (the limit of applicability of the boiling transition correlation) it would be assumed that the fuel cladding integrity safety limit has been violated. Change #5

In addition to the boiling transition limit (MCPR = 1.05) operation is constrained to a maximum LHGR$\leq$18.5 Kw/ft. At 100% power this limit is reached with a maximum total peaking factor (MTPF) of 2.60. For the case of the MTPF exceeding 2.60, operation is permitted only at less than 100% of rated thermal

BASES:

1.1 FUEL CLADDING INTEGRITY SAFETY LIMIT (Cont'd)

power and only with reduced APRM scram settings as required by Specification 2.1.A.1.

Change #5

The actual power distribution in the core is established by specified control rod sequence and is monitored continuously by the incore local power range monitor (LPRM) system. However, to maintain applicability of the safety limit curves on Figure 2.1-1, the safety limits will be lowered according to the equations expressed in Specification 2.1 in the rare event of power operation with a total peaking factor in excess of 2.60.

At pressure below 800 psia, the core elevation pressure drop (0 power, 0 flow) is greater than 4.56 psi. At low powers and flows this pressure differential is maintained in the bypass region of the core. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low powers and flows will always be greater than 4.56 psi. Analyses show that with a flow of 28 x 10^3 lbs/hr bundle flow, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.56 psi driving head will be greater than 28 x 10^3 lbs/hr. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors this corresponds to a core thermal power of more than 50%. Thus, a core thermal power limit of 25% for reactor pressures below 800 psia is conservative.

Change #5

Plant safety analyses have shown that if a scram occurs when a limiting safety system scram setting is exceeded, the safety limit of Specifications 1.1.A or B will not be exceeded.

During transient operation, the heat flux would lag behind the neutron flux due to the inherent heat transfer time constant of the fuel, which is eight to nine seconds. Also, the limiting safety system scram settings are at values which

BASES:

1.1 FUEL CLADDING INTEGRITY SAFETY LIMIT (Cont'd)

will not allow the reactor to be operated above the safety limit during normal operation.

In addition, control rod scrams are such that, even for abnormal operating transients, the neutron flux transient is terminated before a significant increase in surface heat flux occurs. Scram times are checked periodically to assure the insertion times are adequate. The thermal power transient resulting when a scram is accomplished other than by the primary source signal (e.g., scram from neutron flux following closure of the main turbine stop valves) does not necessarily cause fuel damage. However, for this specification a safety limit violation will be assumed when a scram is only accomplished by means of a backup feature of the plant design. The concept of not approaching a safety limit provided primary source scram signals are operable is supported by the extensive plant safety analysis.

Change #5

The computer system has a sequence annunciation program that will indicate the sequence in which various scram initiation signals are generated. This program also indicates when each scram setpoint is cleared. This can provide information on how long a scram condition exists, allowing some measurement of the energy added during a transient. Thus, information usually will be available for analyzing scrams. If the computer information should not be available for any scram analysis, the note following Specification 1.1.B will be relied on to determine if a safety limit has been violated.

During periods when the reactor is shut down, consideration must also be given to water level requirements due to the effect of decay heat. If reactor water level should drop below the top of the active fuel during this time, the ability to cool the core is reduced. This reduction in core cooling capability could lead to elevated cladding temperatures and perforation.

BASES:

1.1 FUEL CLADDING INTEGRITY SAFETY LIMIT (Cont'd)

Establishment of the safety limit at 18 inches above the top of the fuel provides adequate margin.

References:

1. General Electric BWR Thermal Analysis Basis (GETAB) Data, Correlation and Design Application, NEDO 10958 and NEDE 10958.

Change #5

BASES:

2.1

LIMITING SAFETY SYSTEM SETTINGS RELATED TO FUEL CLADDING INTEGRITY

The abnormal operational transients applicable to operation of the Brunswick Plant have been analyzed throughout the spectrum of planned operating conditions up to the design thermal power condition of 2531 MWt at 100 percent recirculation flow. The analyses were based upon plant operation in accordance with the operating map given in Figure 3.7-1 of the FSAR. In addition, 2436 MWt is the licensed maximum power level of Brunswick, and this maximum steady-state power will never be knowingly exceeded.

Transient analyses were not performed for a power level that specifically included instrument errors. To permit appropriate conclusions from analyses which do not include instrument errors, conservatism was incorporated in the controlling factors such as void reactivity coefficient, control rod scram worth, scram delay time, peaking factors, axial power shapes, etc. These factors are all selected conservatively with respect to their effect on the applicable transient results as determined by the current analysis model. This transient model, evolved over many years, has been substantiated in operation as a conservative tool for evaluating reactor dynamics performance. Results obtained from a General Electric boiling water reactor have been compared with predictions made by the model. The comparisons and results are summarized in Reference 1. Change #5

The void reactivity coefficient utilized in the analysis is estimated to be about 25% more conservative than any value expected to occur during the core lifetime. The scram worth used has been derated to be equivalent to the scram worth of about 80% of the control rods. The effect of scram worth, scram delay time and rod insertion rate, all conservatively applied, are of greatest significance in the early portion of the negative reactivity insertion. The rapid insertion of negative reactivity is assured by the time requirements for 5% and 20% insertion. By the time the rods are 60% inserted, approximately four dollars of negative reactivity have been inserted which strongly turns the transient, and accomplishes the desired effect. Change #5

BASES:

2.1

LIMITING SAFETY SYSTEM SETTINGS RELATED TO FUEL CLADDING INTEGRITY (Cont'd)

The time for 50 percent and 90 percent insertions are given to assure proper completion of the insertion stroke, to further assure the expected performance in the earlier portion of the transient, and to establish the ultimate fully shutdown steady-state condition.

For analyses of the thermal consequences of the transients a MCPR of 1.27 is conservatively assumed to exist prior to initiation of the transients. Change #5

This choice of using conservative values of controlling parameters and initiating transients at the rated power level produces more pessimistic answers than would result by using expected values of control parameters and analyzing at higher power levels.

BASES:

2.1

LIMITING SAFETY SYSTEM SETTINGS RELATED TO FUEL CLADDING INTEGRITY (Cont'd)

The bases for individual setpoints are discussed below:

A. Neutron Flux Scram

The average power range monitoring (APRM) system, which is calibrated using heat balance data taken during steady-state conditions, reads in percent of rated power (2436 MWt). Because fission chambers provide the basic input signals, the APRM system responds directly to average neutron flux. During transients, the instantaneous rate of heat transfer from the fuel (reactor thermal power) is less than the instantaneous neutron flux due to the time constant of the fuel. Therefore, during abnormal operational transients, the thermal flux of the fuel will be less than that indicated by the neutron flux at the scram setting. Analyses have demonstrated that with a 120 percent scram trip setting, none of the abnormal operational transients analyzed violate the fuel safety limit and there is a substantial margin from fuel damage. Therefore, use of a flow-biased scram provides even additional margin.

Change #5

5

An increase in the APRM scram setting would decrease the margin present before the fuel cladding integrity safety limit is reached. The APRM scram setting was determined by an analysis of margins required to provide a reasonable range for maneuvering during operation. Reducing this operating margin would increase the frequency of spurious scrams, which have an adverse effect on reactor safety because of the resulting thermal stresses. Thus, the APRM scram trip setting was selected because it provides adequate margin for the fuel cladding integrity safety limit yet allows operating margin that reduces the possibility of unnecessary scrams.

BASES:

2.1

LIMITING SAFETY SYSTEM SETTINGS RELATED TO FUEL CLADDING INTEGRITY (Cont'd)

The scram trip setting must be adjusted to ensure that the LHGR transient peak is not increased for any combination of MTPF and reactor core thermal power. The scram setting is adjusted in accordance with the formula in Specification 2.1.A.1, when the maximum total peaking factor is greater than 2.60. Change #5

Analyses of the limiting transients show that no scram adjustment is required to assure MCPR >1.05 when the transient is initiated from MCPR > 1.27.

For operation in the startup mode while the reactor is at low pressure, APRM scram is set at ≤15 percent of rated power. This provides an adequate thermal margin between the setpoint and the safety limit, 25 percent rated power. The margin adequately accommodates anticipated maneuvers associated with plant startup. Effects of increasing pressure at zero or low void content are minor, cold water from sources available during startup is not much colder than that already in the system, temperature coefficients are small, and control rod patterns are constrained to be uniform by operating procedures backed up by the rod worth minimizer and the rod sequence control system. Change #5 5

Worth of individual rods is very low in a uniform rod pattern. Thus, of all possible sources of reactivity input, uniform control rod withdrawal is the most probable case of significant power rise. Because the flux distribution associated with uniform rod withdrawals does not involve high local peaks, and because several rods must be moved to change power by a significant percentage of rated power, the rate of power rise is slow. Generally, the heat flux is in near equilibrium with the fission rate. In an assumed uniform rod withdrawal approach to the scram level, the rate of power rise is no more

BASES:

2.1

LIMITING SAFETY SYSTEM SETTINGS RELATED TO FUEL CLADDING INTEGRITY (Cont'd)

than five percent of rated power per minute, and the APRM system would be more than adequate to assure a scram before power could exceed the safety limit. The APRM 15 percent power scram remains active until the mode switch is placed in the RUN position.

The IRM system consists of eight chambers, four in each of the reactor protection system logic channels. The IRM is a five-decade instrument which covers the range of power level between that covered by the SRM and the APRM. The five-decades are covered by the IRM by means of a range switch and the five decades are broken down into 10 ranges, each being one-half of a decade in size. The IRM scram setting of 120 divisions is active in each range of the IRM. For example, if the instrument were on range 1, the scram setting would be at 120 divisions for that range; likewise, if the instrument were on range 5, the scram would be 120 divisions on that range. Thus, as the IRM is ranged up to accommodate the increase in power level, the scram setting is also ranged up. In the startup mode, a scram at 120 divisions on the instrument is less than 15 percent power, except for range 10 on the instrument. Range 10 allows for IRM instruments to remain on scale at higher power levels to provide additional overlap and to permit IRM calibration at higher power levels. However, the APRM 15 percent scram prevents higher power operation without being in the RUN mode. The IRM scram provides protection for changes which occur both locally and over the entire core. The IRM, because of the scram arrangement discussed above, thus provides additional or backup protection to the APRM 15-percent

Change #5

BASES:

2.1

LIMITING SAFETY SYSTEM SETTINGS RELATED TO FUEL CLADDING INTEGRITY (Cont'd)

scram in the startup mode. The most significant sources of reactivity change during the power increase are due to control rod withdrawal. For insequence control rod withdrawal, the rate of change of power is slow enough due to the physical limitation of withdrawing control rods, that heat flux is in equilibrium with the neutron flux and an IRM scram would result in a reactor shutdown well before any safety limit is exceeded. In order to ensure that the IRM provided adequate protection against the single rod withdrawal error, a range of rod withdrawal accidents was analyzed. This analysis included starting the accident at various power levels. The most severe case involves an initial condition in which the reactor is just subcritical and the IRM system is not yet on scale. This condition exists at quarter-rod density. Additional conservatism was taken in this analysis by assuming that the IRM channel closest to the withdrawn rod is bypassed. The results of this analysis show that the reactor is scrammed and peak power limited to one percent of rated power, thus maintaining MCPR above 1.05. Based on the above analysis, the IRM provides protection against local control rod withdrawal errors and continuous withdrawal of control rods in sequence and provides backup protection for the APRM.

Change #5

Change #5

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B. APRM Control Rod Block (1)

Reactor power level may be varied by moving control rods or by varying the recirculation flow rate. The APRM system provides a control rod block to prevent rod withdrawal beyond a given point at constant recirculation flow rate, and thus to protect against

BASES:

2.1

LIMITING SAFETY SYSTEM SETTINGS RELATED TO FUEL CLADDING INTEGRITY (Cont'd)

MCPR less than 1.05. This rod block setpoint, which is automatically varied with recirculation loop flow rate, prevents an increase in the reactor power level to excessive values due to control rod withdrawal. The specified flow variable setpoint provides substantial margin against fuel damage, assuming a steady-state operation at the setpoint, over the entire recirculation flow range. The margin to the safety limit increases as the flow decreases for the specified trip point vs. flow relationship; therefore, the worst case MCPR during steady-state operation is at 108 percent of rated thermal power. The actual power distribution core is established by specified control rod sequences and is monitored continuously by the in-core LPRM system. As with the APRM scram setting, the APRM rod block setting is adjusted downward if peaking factors greater than 2.60 exist. The rod block setting is changed by changing the intercept point of the flow bias curve; thus, the entire curve will be shifted downward.

Change #5

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Change #5

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References

1. Linford, R. B., "Analytical Methods of Plant Transient Evaluations for the General Electric Boiling Water Reactor," NEDO-10802, Feb., 1973.

C. Reactor Low Water Level #1 Scram

The setpoint for low level scram is above the bottom of the separator skirt. This level has been used in transient analyses dealing with coolant inventory decrease. The results show that scram at this level adequately protects the fuel and the pressure barrier,

Change #5

BASES:

2.1 LIMITING SAFETY SYSTEM SETTINGS RELATED TO FUEL CLADDING INTEGRITY (Cont'd)

D. Turbine Stop Valve Closure Scram

The turbine stop valve closure scram anticipates the pressure, neutron flux, and heat flux increase that could result from rapid closure of the turbine stop valves. With a scram setting at 10 percent of valve closure, the resultant increase in surface heat flux is such that MCPR remains above 1.05 even during the worst-case transient that assumes the turbine bypass is closed. Turbine stop valve closure scram is bypassed when reactor power is $\leq$30 percent rated.

Change #5

E. Turbine Control Valve Scram

1. Turbine Control Valve Fast Closure Scram

The reactor protection system initiates a scram signal after the control valve hydraulic oil pressure decreases due to a load rejection exceeding the capacity of the bypass valves or due to hydraulic oil system rupture. The turbine hydraulic control system operates using high pressure oil. There are several points in this oil system where a loss of oil pressure could result in a fast closure of the turbine control valves. The control valve closure time is approximately twice as long as that for the stop valves which means that resulting transients, while similar, are less severe than for stop valve closure. No fuel damage occurs, and reactor system pressure does not exceed the safety relief valve setpoint. This is an anticipatory scram and results in reactor shutdown before any significant increase in pressure or neutron flux occurs. This scram is bypassed when turbine steamflow is below 30 percent of rated, as measured by turbine first-stage pressure.

BASES:

2.1

LIMITING SAFETY SYSTEM SETTINGS RELATED TO FUEL CLADDING INTEGRITY (Cont'd)

2. Select Rod Insert

Select rod insert is an operational aid designed to insert a predetermined group of control rods immediately following either a generator load rejection, loss of turbine control valve hydraulic pressure, or by manual operator action using a switch on the R-T-G board. The assignment of control rods to the select rod insert function is based on the startup and fuel warranty service associated with each control rod pattern, on RSCS considerations, and a dynamic function of both time and core patterns.

Approximately ten percent of the control rods in the reactor will be assigned to the select rod insert function by the operator. This selection will be accomplished by moving the rod scram test switch for those rods from the "NORMAL" position to the "SELECT ROD INSERT" position.

For Brunswick Unit No. 2, loss of turbine control valve hydraulic pressure will initiate the select rod insert function and the pre-selected group of control rods will be fully inserted. A reactor protection system trip will not be initiated prior to determination of turbine bypass valve status. Determination of the bypass valve status will be delayed by 200 msec referenced to the start of the low turbine control valve hydraulic pressure signal. If the bypass valves are not open, as determined by limit switches, the reactor protection system will scram the reactor. Any rod selected for Select Rod Insert should also have the other rods in its notch group selected to ensure that the RCSC (Rod Sequence Control System) criteria of plus-minus one notch

BASES:

2.1

LIMITING SAFETY SYSTEM SETTINGS RELATED TO FUEL CLADDING INTEGRITY (Cont'd)

position equality is met when the rod pattern is between 50% rod density and 20% reactor power. It is possible that a rod pattern within these limits may occur after the Select Rod Insert Function operates. Change #1

F & G. Main Steamline Isolation on Low Pressure and Main Steamline Isolation Scram

The low pressure isolation of the main steamlines at 850 psig was provided to protect against rapid reactor depressurization and the resulting rapid cooldown of the vessel. Advantage is taken of the scram feature that occurs when the main steamline isolation valves are closed, to provide for reactor shutdown so that high power operation at low reactor pressure does not occur, thus providing protection for the fuel cladding integrity safety limit. Operation of the reactor at pressures lower than 850 psig requires that the reactor mode switch be in the STARTUP position, where protection of the fuel cladding integrity safety limit is provided by the IRM high neutron flux scram. Thus, the combination of main steamline low pressure isolation and isolation valve closure scram assures the availability of neutron flux scram protection over the entire range of applicability of the fuel cladding integrity safety limit. In addition, the isolation valve closure scram anticipates the pressure and flux transients that occur during normal or inadvertent isolation valve closure.

BASES:

2.1

LIMITING SAFETY SYSTEM SETTINGS RELATED TO FUEL CLADDING INTEGRITY (Cont'd)

H & I. Reactor Low Water Level Setpoints for Initiation of HPCI and RCIC, Automatic Depressurization, and Starting LPCI and Core Spray Pumps

These systems maintain adequate cooling inventory and provide core cooling with the objective of preventing excessive clad temperatures. The design of these systems to adequately perform the intended function is based on the specified low level scram setpoint and initiation setpoints. Transient analyses demonstrate that these conditions result in adequate safety margins for both the fuel and the system pressure.

LIMITING CONDITIONS FOR OPERATION

3.1 Reactor Protection System

Applicability:

Applies to the operability of plant instrumentation and control systems required for reactor safety.

Objective:

To specify the limits imposed on plant operation by those instrument and control systems required for reactor safety.

Specification:

A. Plant Operation

Plant operation at any power level shall be permitted only in accordance with Table 3.1-1.

B. System Response

The designated system response time from actuation of the sensor contact or trip output to the de-energization of the scram solenoid relay shall not exceed 100 milliseconds.

C. Minimum Critical Power Ratio (MCPR)

During steady state power operation, MCPR shall be $\geq$ 1.27 at rated power and flow. For core flows other than rated the MCPR shall be > 1.27 times K_f, where K_f is as shown in Figure 3.1-1.

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SURVEILLANCE REQUIREMENTS

4.1 Reactor Protection System

Applicability:

Applies to the surveillance of the plant instrumentation and control systems required for reactor safety.

Objective:

To specify the type and frequency of surveillance to be applied to those instrument and control systems required for reactor safety.

Specification:

A. Plant Operation

Instrumentation systems shall be functionally tested and calibrated as indicated in Table 4.1-1.

B. System Response

The system response times will be checked prior to initial fuel loading.

C. Minimum Critical Power Ratio (MCPR)

MCPR shall be determined daily during reactor power operation at $\geq$ 25% rated thermal power and following any change in power level or distribution that would cause operation with a limiting control rod pattern as described in the bases for Specification 3.3.B.5.

Change #5

Change #5

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS

3.1 Reactor Protection System (Cont'd)

D. Average Planar Linear Heat Generation Rate (APLHGR)

During steady state power operation, the APLHGR for each type of fuel as a function of average planar exposure shall not exceed the limiting value shown in Figure 3.1-2A or 3.1-2B.

E. Local Linear Heat Generation Rate (LHGR)

During steady state power operation, the linear heat generation rate (LHGR) of any rod in any fuel assembly at any axial location shall not exceed the maximum allowable LHGR as calculated by the following equation:

$$LHGR_{max} \leq LHGR_d \left[1 - \{ (\Delta P/P)_{max} (L/L_T) \} \right]$$

$LHGR_d$ = Design LHGR = 18.5 KW/ft.

$(\Delta P/P)_{max}$ = Maximum power spiking penalty

= 0.026

L_T = Total core length = 12 feet

L = Axial position above bottom of core

4.1 Reactor Protection System (Cont'd)

D. Average Planar Linear Heat Generation Rate (APLHGR)

The maximum ratio of the limiting value for APLHGR as a function of average planar exposure to the APLHGR value (APLHGR RATIO) for each type of fuel shall be determined daily during reactor power operation at ≥25% rated thermal power.

E. Local Linear Heat Generation Rate (LHGR)

The LHGR as a function of core height shall be checked daily during reactor operation at ≥ 25% rated thermal power.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS

4.1 Reactor Protection System (Cont'd)

F. Heat Flux and Maximum Total Peaking Factor

Once a day during reactor power operation and at constant power $\geq$ 25% the maximum peak heat flux and the total peaking factor shall be checked and the SCRAM and APRM Rod Block settings given by Specifications 2.1.A.1 and 2.1.B shall be calculated if the peaking factor exceeds 2.60.

Change #5

G. Inoperable Channels

When an instrument channel monitoring any variable in the reactor protection system (RPS) fails, its associated RPS trip system must be manually tripped if the minimum number of operable instrument channels per trip system cannot be met.

The failed instrument channel may be bypassed to permit functional testing of the untripped RPS trip system providing that the remaining operable instrument channels monitoring the same variable in the tripped trip system are functionally tested immediately prior to bypassing the inoperable instrument channel.

In no case shall the inoperable instrument channel be bypassed for greater than eight hours per each functional test of the untripped trip system.

Change #1 & #5

Change #1

TABLE 3.1-1

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT REQUIREMENTS

Trip Function	Trip Settings	Modes in Which Functions Must be Operable Refuel (1)	Startup	Run	Min. No. Operable Instrument Channels Per Trip System (2)	Required Conditions When Minimum Conditions for Operation Are Not Satisfied (3)
1. Mode switch C72A-S1		X	X	X	1	A
2. Manual trip C72A-S3A, B		X	X	X	1	A
3. IRM High flux	$\le$120/125 of scale	X	X	(12)	3	A
Inoperative		X	X	(12)	3	A
4. APRM						
High Flux (4,14) (flow bias)	$\le(0.66W+54)\frac{(2.60)}{MTPF}$			X	2	B
(fixed) (14)	$\le$120% of rated power			X	2	B
Inoperative		X		X	2(5)	B
Downscale	$\ge$3/125 of scale			X(13)	2	B
Startup	$\le$15% of rated power	X	X		2	A
5. High reactor pressure B21-PS-N023A, B,C,D	$\le$1045 psig	X(10)	X	X	2	A
6. High drywell pressure C72-PS-N002A, B,C,D	$\le$2 psig	X(11)	X(11)	X	2	A
7. Reactor low water level #1 B21-LIS-N017A, B,C,D	$\ge$12.5 inch (6)	X	X	X	2	A

TABLE 3.1-1 (Cont'd)

Trip Function	Trip Settings	Modes in Which Functions Must be Operable: Refuel (1)	Startup	Run	Min. No. Operable Instrument Channels Per Trip System (2)	Required Conditions When Minimum Conditions for Operation Are Not Satisfied (3)
8. Scram discharge volume high level C11/C12-LSH-N013A,B,C,D	≤ 109 Gallons	X	X	X	2	A
9. Main steamline high radiation D12-RM-K603A,B,C,D	≤ 3x normal background at rated power	X	X	X	2	C
10. Main steamline isolation valve closure B21-ZS-F022A,B,C,D B21-ZS-F028A,B,C,D	≤ 10% valve closure (7)	X	X	X	4	C
11. Turbine stop valve closure EHC-SVOS-1,2,3,4	≤ 10% valve (9) closure			X	4	D

BASES:

4.1 Surveillance Requirement for Reactor Protection System

A. The scram sensor channels listed in Table 4.1-1 are divided into three groups: A, B, and C.

Group A sensors are of the on/off-type and will be tested and calibrated at indicated intervals.

Group B devices utilize an analog sensor followed by an amplifier and bistable trip circuit. This type of equipment incorporates control room mounted indicators and annunciator alarms. A failure in the sensor or amplifier may be detected by an alarm or by an operator who observes that one indicator does not track the others in similar channels. The bistable trip circuit failures are detected by the periodic testing.

Group C devices are active only during a given portion of the operating cycle. For example, the IRM is active during startup and inactive during full power operation. Testing of these instruments is only meaningful within a reasonable period prior to their use.

B. The system response times will be checked prior to initial fuel loading to ensure adequate reactor protection.

C. At core thermal power levels less than or equal to 25%, the reactor will be operating at minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience and thermal hydraulic analysis indicated that the resulting MCPR value is in excess of requirements by a considerable margin. With this low void content, any inadvertent core flow increase would only place operation in a more conservative mode relative to MCPR. During initial start-up testing of the plant, a MCPR evaluation will be made at the 25% thermal power level with minimum recirculation pump speed.

/ Change #5

BASES:

4.1.C Surveillance Requirement for Reactor Protection System (Cont'd)

The MCPR margin will thus be demonstrated such that future MCPR evaluations below this power level will be shown to be unnecessary. The daily requirement for calculating MCPR above 25% rated thermal power is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement for calculating MCPR when a limiting control rod pattern is approached ensures that MCPR will be known following a change in power or power shape (regardless of magnitude) that could place operation at a thermal limit.

D. This specification assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the limit specified in the 10CFR50, Appendix K.

The peak cladding temperature following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is only dependent secondarily on the rod to rod power distribution within an assembly. Since expected local variations in power distribution within a fuel assembly affect the calculated peak clad temperature by less than ± 20°F relative to the peak temperature for a typical fuel design, the limit on the average linear heat generation rate is sufficient to assure that calculated temperatures are within the 10CFR50 Appendix K limit. The limiting value for APLHGR is shown in Figure 3.1-2A for fuel types 1 and 3 and Figure 3.1-2B for fuel type 2.

E. This specification assures that the linear heat generation rate in any rod is less than the design linear heat generation if fuel pellet densification is postulated. The power spike penalty specified is based on the analysis presented in Section 3.2.1 of Reference 1 and in References 2 and 3, and assumes a linearly increasing variation in

Change #5

BASES:

4.1.E Surveillance Requirement for Reactor Protection System (Cont'd)

axial gaps between core bottom and top, and assures with a 95% confidence, that no more than one fuel rod exceeds the design linear heat generation rate due to power spiking. The LHGR as a function of core height shall be checked daily during reactor operation at $\geq$ 25% power to determine if fuel burnup, or control rod movement has caused changes in power distribution. For LHGR to be a limiting value below 25% rated thermal power, the MTPF would have to be greater than 10 which is precluded by a considerable margin when employing any permissible control rod pattern.

Change #5

F. The peak heat flux shall be checked once per day to determine if the APRM scram setpoint requires adjustment. This will normally be done by checking the LPRM readings. Only a small number of control rods are moved daily; thus, the peaking factors are not expected to change significantly. Consequently, the daily check of the peak heat flux is adequate.

References:

1. "Fuel Densification Effects on General Electric Boiling Water Reactor Fuel," Supplements 6, 7, and 8, NEDM-10735, August 1973.

2. Supplement 1 to Technical Report on Densifications of General Electric Reactor Fuels, December 14, 1973 (USA Regulatory Staff).

3. Communication: V. A. Moore to I. S. Mitchell, "Modified GE Model for Fuel Densification," Docket 50-321, March 27, 1974.

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TABLE 3.2-11 (Cont'd)

CONTROL ROD BLOCKS INITIATED FROM NEUTRON MONITORING SYSTEM

Trip Function	Minimum Number of Operable Instrument Channels (2)	Modes in Which Function Must Be Operable: Refuel	Startup	Run	Trip Setting	Remarks
c. Detector not in "full in" position, channels A through H, Relays C51-K9E through H, & J through M	6	X	X		Detector motor module limit switch LS-4 not closed (detector not full in)	Bypassed in run mode.
d. Downscale IRM channels A through H, Relay C51-K51	6	X	X		$\geq$ 3/125 of Scale	Bypassed in run mode and when IRM is in RANGE 1.
3. Average power range monitor						
a. Upscale APRM channels A through F, Relays K1 & K7	4			X	$\leq (0.66W+42)\ \frac{2.60}{MTPF}$	
b. Inoperative APRM channels A through F Relays K2 & K8	4	X	X	X	(1)	
c. Downscale APRM channels A through F Relays K3 & K9	4			X	$\geq$ 3/125 of Full Scale	Only active when mode switch is in RUN
d. Upscale startup APRM channels A through F Relay K18	4	X	X		$\leq$ 12% power	Bypassed when in run mode.

Change #1

Change #1

Change #1 & #5

Change #1

TABLE 3.2-11 (Cont'd)

CONTROL ROD BLOCKS INITIATED FROM NEUTRON MONITORING SYSTEM

Trip Function	Minimum Number of Operable Instrument Channels (2)	Modes in Which Function Must Be Operable: Refuel	Startup	Run	Trip Setting	Remarks
4. Rod block monitor						
a. Upscale RBM channels A,B Relay K1	1			X	$\leq (0.66W+42)\ \frac{2.60}{MTPF}$	
b. Downscale RBM channels A,B Relay K2	1			X	$\geq$ 3/125 of full scale	Only active when mode switch is in RUN and reactor power is $\geq$ 30%
c. Inoperative RBM channels A,B Relay K3				X	(1)	

Change #5

BSEP-1 & 2

NOTES:

(1) The inoperative trips are produced by the following conditions:

(a) SRM and IRM

1) Mode switch not in OPERATE
2) High voltage power supply voltage low
3) Circuit boards not in circuit

(b) APRM

1) Mode switch not in OPERATE
2) Less than 11 LPRM inputs
3) Circuit boards not in circuit

(c) RBM

1) Mode switch not in OPERATE
2) Circuit boards not in circuit
3) RBM fails to null
4) Less than required number of LPRM inputs for rod selected.

(2) If the minimum number of channels cannot be met for one out of two trip systems, seven days is allowed before requiring the affected trip system to be tripped. If both trip systems do not meet the minimum number of operable channels for operation, both trip systems shall be tripped.

Change #1

BASES:

3.2.B Core Standby Cooling System (CSCS) (Cont'd)

Section 3.5. Whenever an instrument in one subsystem is inoperable the limiting condition for operation as specified in Section 3.5 applies. If an instrument is in more than one subsystem of CSCS, then Section 3.5 is too restrictive and the inoperable channel shall be tripped using special jacks or other permanently installed circuits.

C. Control Rod Blocks

The control rod block functions are provided to prevent excessive control rod withdrawal so that MCPR does not decrease to the safety limit. The trip logic for this function is one out of n; e.g., any trip on one of the six APRMs, eight IRMs, or four SRMs will result in a rod block. The minimum instrument channel requirements for the IRM may be reduced by one for a short period of time to allow for maintenance, testing or calibration. The RBM is an operational guide and aid only and is not needed for rod withdrawal. Change #5

The APRM rod block trip is flow referenced and prevents a significant reduction in MCPR, especially during operation at reduced flow. The APRM provided gross core protection; i.e., limits the gross core power increase from withdrawal of control rods in the normal withdrawal sequence. The rod block trips are set so that MCPR is maintained greater than the safety limit. Change #5

The IRM rod block function provides local as well as gross core protection. The scaling arrangement is such that the trip setting is less than a factor of 10 above the indicated level. Analysis of the worst case accident results in rod block action before MCPR approaches the safety limit. Change #5 Change #5

BASES:

3.2.C Control Rod Blocks (Cont'd)

A downscale indication on an APRM or IRM is an indication the instrument has failed or the instrument is not sensitive enough. In either case, the instrument will not respond to changes in control rod motion; thus, control rod motion is prevented.

When minimum conditions for operation are not met, the required action is to leave the channel in the tripped condition until it is repaired.

D. Radiation Monitoring Systems - Isolation And Initiation Functions

Two radiation monitors are provided which initiate isolation of the Reactor Building and operation of the standby gas treatment system. The monitors are located in the Reactor Building ventilation duct. Any one upscale trip will initiate the isolation. Trip settings for the monitors in the ventilation duct are based upon initiation of the normal ventilation isolation and standby gas treatment system operation to limit the dose rate at the nearest site boundary to less than the dose rate allowed by 10CFR20. Change #1

If the minimum conditions for operation are not met, the Reactor Building ventilation system shall be isolated and the standby gas treatment system operated until the instrumentation is repaired.

E. Drywell Leak Detection Monitors

The instrumentation that monitors drywell leak detection provides the information to determine whether Specification 3.6.C. (Coolant Leakage) is met, therefore, the limiting condition for operation is the same as Specification 3.6.C.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS

3.3.B Control Rods (Cont'd)

4. Control rods shall not be withdrawn for startup or refueling unless at least two source range channels have an observed count rate equal to or greater than three counts per second. The minimum count rate may be reduced to 0.3 cps for the first core load when the source is at low strength.

Change #2

5. During reactor power operation with limiting control rod patterns, as determined by a Plant Engineer, either:

   a. Both RBM channels shall be operable; or

   b. Control rod withdrawal shall be blocked; or

   c. The operating power level shall be limited so that the MCPR will remain above 1.05 assuming a single error that results in complete withdrawal of any single operable control rod.

Change #5

6. In order to perform the required shutdown margin demonstrations subsequent to any fuel loading operations, to perform tests to verify shutdown margin due to inoperable control rod, or to perform control rod drive scram and/or friction testing and the initial startup test program, the relaxation of the following RSCS restraints is permitted. The sequence restraints imposed on control rod groups A_{12}, A_{34}, B_{12} or B_{34} may be removed for the test period by means of the individual rod position bypass switches.

Amendment 1

4.3.B Control Rods (Cont'd)

4. Prior to control rod withdrawal for startup or during refueling, verify that at least two source range channels have an observed count rate of at least three counts per second. The minimum count rate may be reduced to 0.3 cps for the first core load when the source is at low strength.

Change 5

5. When a limiting control rod pattern exists, an instrument functional test of the RBM shall be performed prior to withdrawal of the designated rod(s) and daily thereafter.

6. Prior to control rod withdrawal for startup, verify the conformance to specification 3.3.B.3d before a rod may be bypassed in the RSCS. The requirements to allow use of the individual rod position bypass switches within rod groups A_{12}, A_{34}, B_{12}, or B_{34} of the RSCS during shutdown margin, scram time or friction testing and the initial startup test program are:

   (a) RWM operable as per specification 3.3.B.3C.

   (b) After the bypassing of the rods in the RSCS groups A_{12}, A_{34}, B_{12} or B_{34} for test purposes, it shall be demonstrated that movemen of the rods in the 50 percent density to the preset power level range is blocked or limited to the single notch mode of withdrawal.

   (c) A second licensed operator shall verify the conformance to procedures and this Specification.

Amendment 1

LIMITING CONDITIONS FOR OPERATION

3.3.C Scram Insertion Times

1. The average scram insertion time, based on the deenergization of the scram pilot valve solenoids at time zero, of all operable control rods in the reactor power operation condition shall be no longer than:

Above 950 psig

% Inserted From Fully Withdrawn	Avg. Scram Insertion Times (sec)
5	0.375
20	0.90
50	2.0
90	3.5

2. The average of the scram insertion times for the three fastest control rods of all groups of four control rods in a two-by-two array shall be no longer than:

Above 950 psig

% Inserted From Fully Withdrawn	Avg. Scram Insertion Times (sec)
5	0.398
20	0.954
50	2.120
90	3.800

SURVEILLANCE REQUIREMENTS

4.3.C Scram Insertion Times

1. After each refueling outage all operable fully withdrawn insequence rods shall be scram time tested during operational hydrostatic testing or during startup from the fully withdrawn position with the nuclear system pressure above 800 psig. This testing shall be completed prior to synchronizing the main turbine generator initially following restart of the plant. Prior to exceeding 40% of rated power, all untested operable control rods shall be tested as described above.

2. At 16 week intervals, 10 percent of the control rods capable of movement with control rod drive pressure shall be scram timed above 950 psig. Whenever such scram time measurements are made, an evaluation shall be made to provide reasonable assurance that proper control rod drive performance is being maintained.

If a scram occurs and scram time measurements are available from the scram timing processor, the above 16 week time interval is to start from date of scram.

If a scheduled shutdown is planned near the midcycle period, at which time rod scram measurements will be taken for over 50 percent of the operable control rods, the above 16 week interval does not apply.

Amendment 1

BASES:

3.3.B and 4.3.B Control Rod Withdrawal (Cont'd)

4. The source range monitor (SRM) system performs no automatic safety system function; i.e., it has no scram function. It does provide the operator with a visual indication of neutron level. The consequences of reactivity accidents are functions of the initial neutron flux. The requirement of at least three counts per second assures that any transient, should it occur, begins at or above the initial value of 10^{-8} of rated power used in the analyses of transients from cold conditions. For the initial core, when the startup source strength is low, the minimum requirement will be 0.3 counts per second, which assures any transient would begin at or above 10^{-12} of rated power. One operable SRM channel would be adequate to monitor the approach to criticality, using homogeneous patterns of scattered control rod withdrawal. A minimum of two operable SRMs are provided as an added conservatism. Change #2

5. The rod block monitor (RBM) is designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during high power level operation. Two channels are provided, and one of these may be bypassed from the console for maintenance and/or testing. Tripping of one of the channels will block erroneous rod withdrawal soon enough to prevent fuel damage. This system backs up the operator who withdraws control rods according to written sequences. The specified restrictions with one channel out of service conservatively assure that fuel damage will not occur due to rod withdrawal errors when this condition exists.

   During reactor operation with certain limiting control rod patterns, the withdrawal of a designated single control rod could result in one or more fuel rods with MCHFRs less than 1.0. During use of such patterns, it is judged that testing of the RBM system prior to withdrawal of such rods to assure its operability will assure that improper withdrawal does not occur. It is the responsibility of a Plant Engineer to identify these limiting patterns and the

BASES:

3.3.B.5 and 4.3.B.5 Control Rod Withdrawal (Cont'd)

designated rods either when the patterns are initially established or as they develop due to the occurrence of inoperable control rods in order than limiting patterns.

C. Scram Insertion Times

The control rod system is designated to bring the reactor subcritical at a rate fast enough to prevent fuel damage; i.e., to prevent the MCPR from becoming less than the safety limit. The limiting power transient that resulting from a turbine stop valve closure with failure of the turbine bypass system. Analysis of this transient shows that the negative reactivity rates resulting from the scram with the average response of all the drives as given in the above Specification, provide the required protection. The scram times for all control rods will be determined at the time of each refueling outage. The scram insertion times given in Specification 3.3.C for reactor pressures in excess of 950 psig, when met, insure that adequate insertion rates will result at all reactor pressures below 950 psig. The transient and accident analysis for the plant takes account of the slower scram insertion rates which are characteristic of the drives at certain reactor pressures below 950 psig. Change #5 Change #5

D. Control Rod Accumulators

At reactor pressures in excess of 950 psig, even those control rods with inoperable accumulators will be able to meet required scram insertion times due to the action of reactor pressure. Thus, above this pressure, a control rod drive is not designated as inoperable when the associated accumulator is unavailable. It should also be noted that control rods can be driven in under all operating conditions without the use of the accumulator.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.5.E Automatic Depressurization System (ADS)	4.5.E Automatic Depressurization System (ADS)
1. The automatic depressurization system shall be operable whenever there is irradiated fuel in the reactor vessel and the reactor pressure is greater than 113 psig and prior to a startup from a cold condition, except as specified in 3.5.E.2 below.	1. During each operating cycle the following test shall be performed on the ADS: A simulated automatic actuation test shall be performed prior to startup after each refueling outage.
2. From and after the date that one valve in the automatic depressurization system is made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding 7 days provided that during such 7 days the HPCI system is operable.	2. When one valve of the ADS is made or found to be inoperable, the HPCI system shall be demonstrated to be operable immediately and daily thereafter until the valve is returned to an operable condition. Change No. 5
3. If the requirements of 3.5.E.1 through 2 cannot be met, an orderly shutdown shall be initiated and the reactor pressure shall be reduced to at least 113 psig within 24 hours.	

Change No. 5

LIMITING CONDITIONS FOR OPERATIONS	SURVEILLANCE REQUIREMENTS
3.5.F Minimum Low Pressure Cooling and Diesel Generator Availability	4.5.F Minimum Low Pressure Cooling and Diesel Generator Availability
1. Four diesel generators are normally available for dual or single Unit operation. When one diesel generator is made or found to be inoperable, continued reactor operation is permissible only during the succeeding seven days provided that all of the low pressure core and containment cooling subsystems and the remaining diesel generators shall be operable.	1. During dual or single Unit operation, when one diesel generator is made or found to be inoperable, all low pressure core cooling subsystems shall be demonstrated to be operable within 24 hours. In addition, the operable diesel generators shall be demonstrated to be operable within 24 hours and daily thereafter until the diesel generator is returned to an operable condition.
2. If this requirement (3.5.F.1) cannot be met, an orderly shutdown of both units shall be initiated and the reactors shall be placed in the cold-shutdown condition within 24 hours.	
3. Any combination of inoperable components in the core and containment cooling systems shall not defeat the capability of the remaining operable components to fulfill the cooling functions.	
4. When irradiated fuel is in the reactor vessel and the reactor is in the cold shutdown condition, both core spray systems, the LPCI and containment cooling subsystems may be inoperable, provided no work is being done which has the potential for draining the reactor vessel.	
5. During a refueling outage, refueling operation may continue with one core spray system or the LPCI system inoperable for a period of 30 days.	

BASES:

3.5.E Automatic Depressurization System (Cont'd)

The core spray and/or LPCI provide sufficient flow of coolant to limit fuel clad temperatures to well below that at which fragmentation would occur and to assure that core geometry remains intact.

The ECCS calculations submitted in May 1975 to meet the FAC and conform to Section 50.46 of 10 CFR Part 50 were performed with all seven ADS valves operating for the small break spectrum with HPCI failed. Until analyses are completed to determine the margin provided by the ADS, a failure of one safety/relief valves ADS function will require the same allowable repair period (7 days) as the HPCI or other ECCS.

Change No. 5

F. Minimum Low Pressure Cooling and Diesel Generator Availablity

The purpose of Specification 3.5.F is to assure a minimum of core standby cooling equipment is available at all times. It is during a refueling outage that major maintenance is performed and during such time that all low pressure core cooling systems may be out-of-service. This specification provides that should this occur, no work will be performed on the reactor coolant system which could lead to draining the vessel. This work would include work on certain control rod drive components and reactor recirculation system. Thus, the specification precludes the events which could require core cooling. Specification 3.9 must also be consulted to determine other requirements for the emergency diesel generators.

G. Engineered Safeguards Compartments Cooling and Ventilation

One unit cooler in each pump compartment is capable of providing adequate ventilation flow and cooling. Engineering analyses indicate that the temperature rise in safeguards compartments without adequate ventilation flow or cooling is such that continued operation of the safeguards equipment or associated auxiliary equipment cannot be assured.

BASES:

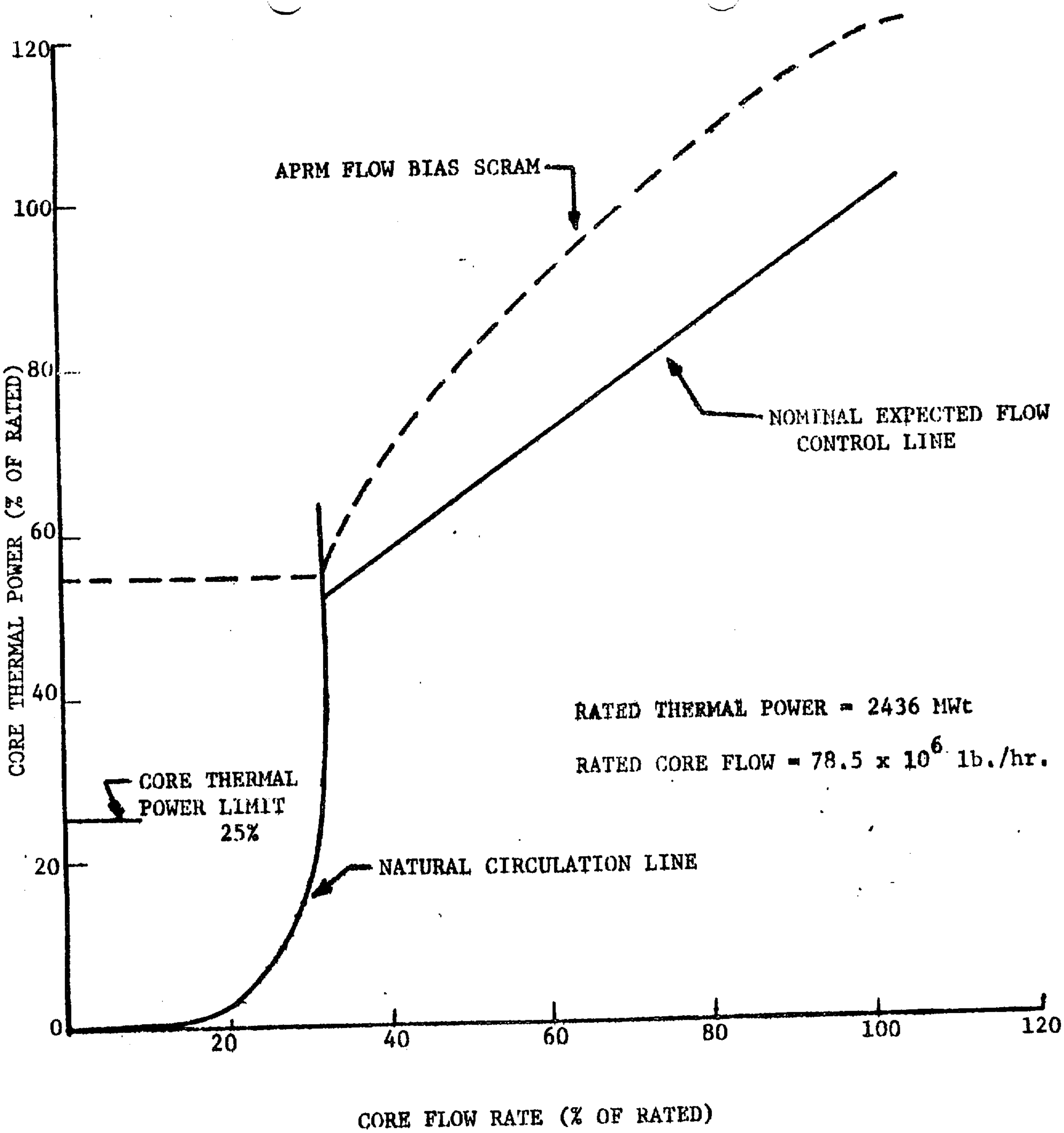
3.5.H Maintenance of Filled Discharge Pipe

If the discharge piping of the core spray, LPCI subsystem, HPCI, and RCIC are not filled, a water hammer can develop in this piping when the pump and/or pumps are started. An analysis has been done which shows that if a water hammer was to occur at the time the system was required, the system would still perform its design function. However, to minimize damage to the discharge piping and to ensure added margin in the operation of these systems, this Technical Specification requires the discharge lines to be filled whenever the system is in an operable condition. In addition, pressure switches indicate the loss of the automatic filling function on the core spray and LPCI subsystems.

3.5.I Control Room Air Treatment System

The control room air treatment system is designed to filter the control room atmosphere for intake air and/or for recirculation during control room isolation conditions. The control room air treatment system is designed to automatically start upon control room isolation and to maintain the control room pressure to the design positive pressure so that all leakage should be out leakage.

High efficiency particulate absolute (HEPA) filters are installed before the charcoal adsorbers to prevent clogging of the iodine adsorbers. The charcoal adsorbers are installed to reduce the potential intake of radioiodine to the control room. The in-place test results should indicate a system leak tightness of less than 1 percent bypass leakage for the charcoal adsorbers and a HEPA efficiency of at least 99 percent removal of DOP particulates. The laboratory carbon sample test results should indicate a radioactive methyl iodide removal efficiency of at least 90 percent for expected accident conditions. If the efficiencies of the HEPA filters and charcoal adsorbers are as specified, the resulting doses will be less than the allowable levels stated in Criterion 19 of the General Design Criteria for Nuclear Power Plants, Appendix A to 10 CFR Part 50.



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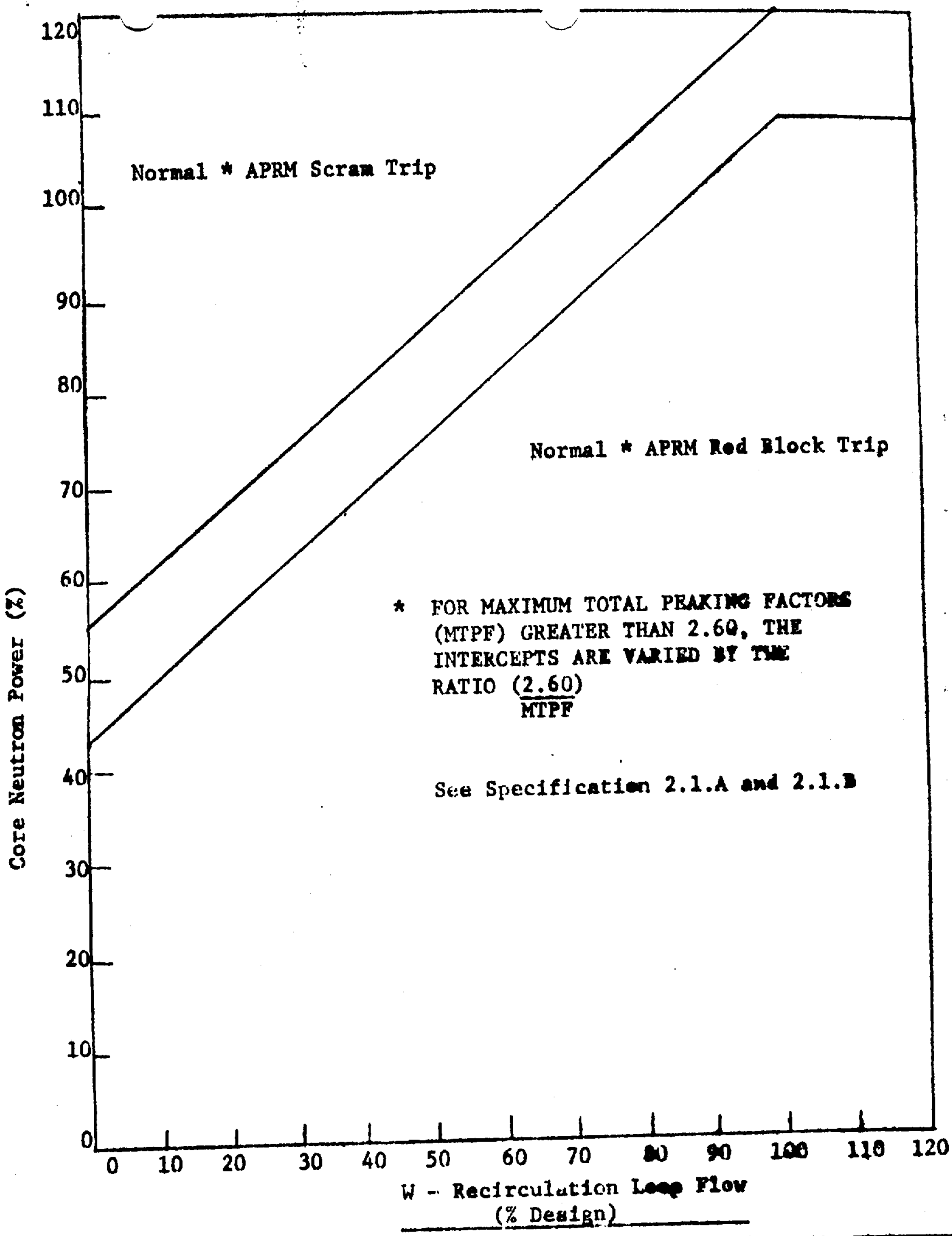
APRM FLOW BIAS SCRAM RELATIONSHIP TO NORMAL OPERATING CONDITIONS	
FIG. NO.	1.1-1

AUGUST 1975



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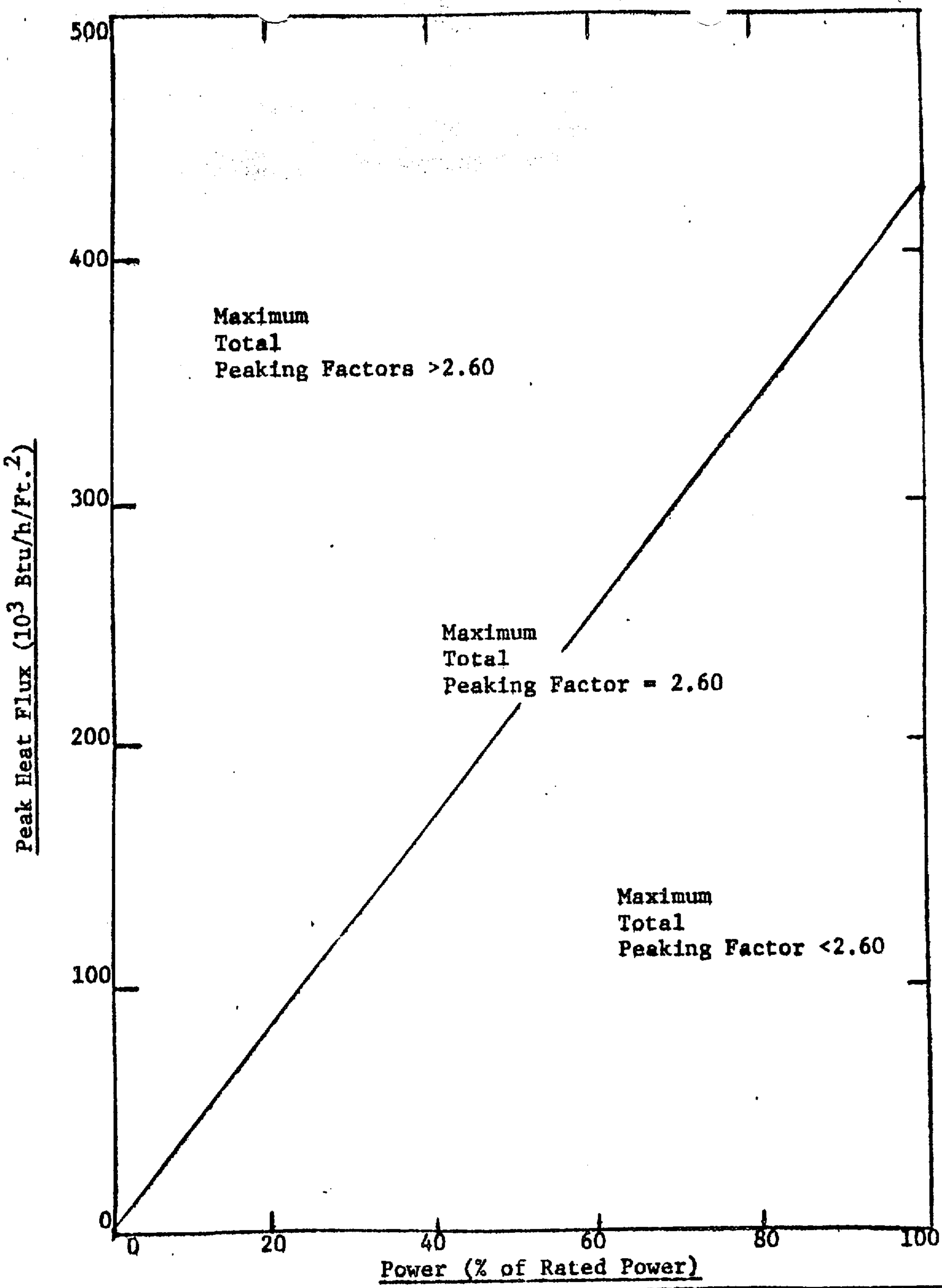
UNITS 1 & 2
Final Safety Analysis Report

APRM Scram and Rod Block
Trip Limiting Safety System Settings

AUGUST 1975

FIG. NO. 2.1-1



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Peak Heat Flux at Various Power Levels for a Peaking Factor of 2.60

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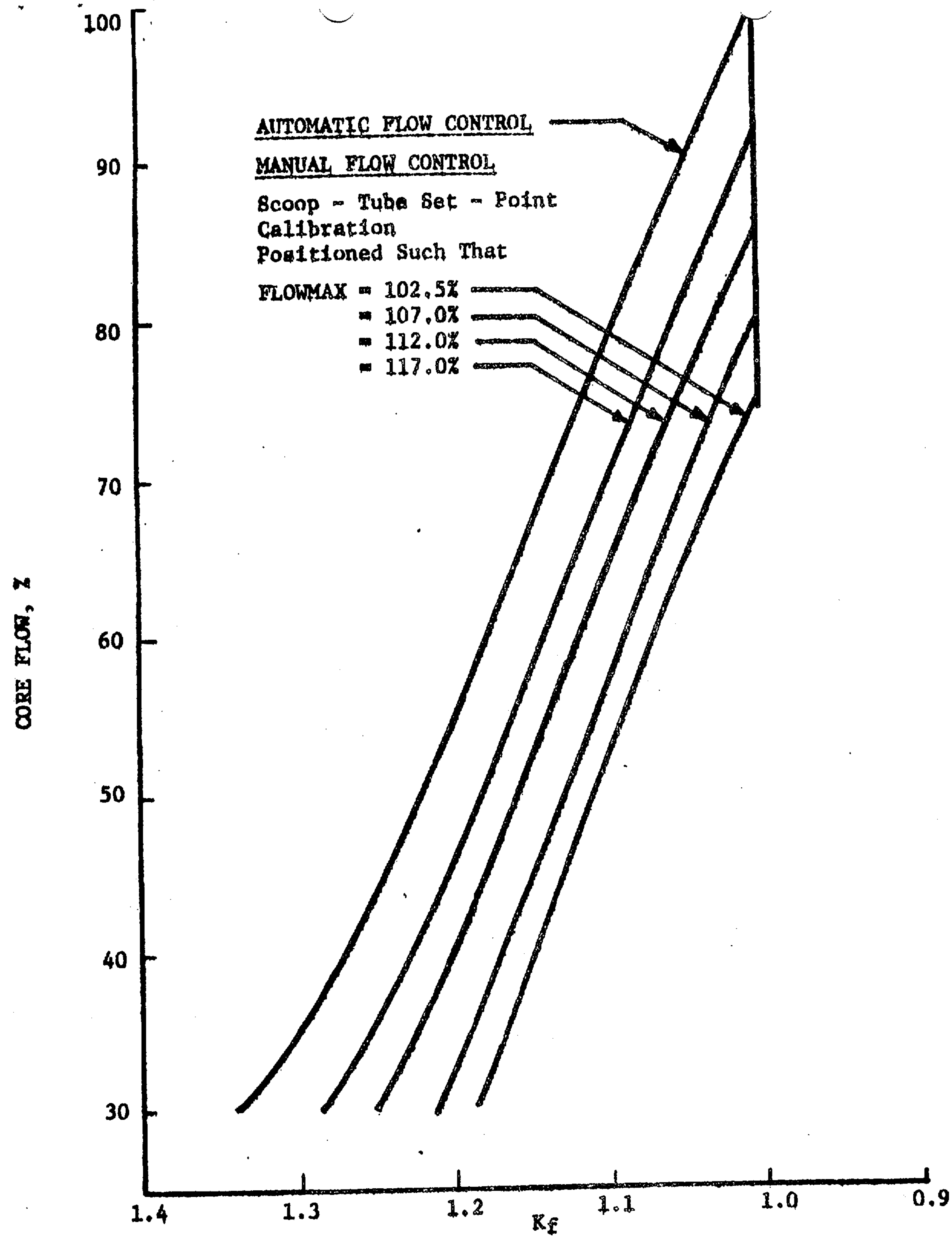
FIG. NO. 2.1-2



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K_f FACTOR

FIG. NO. 3.1-1

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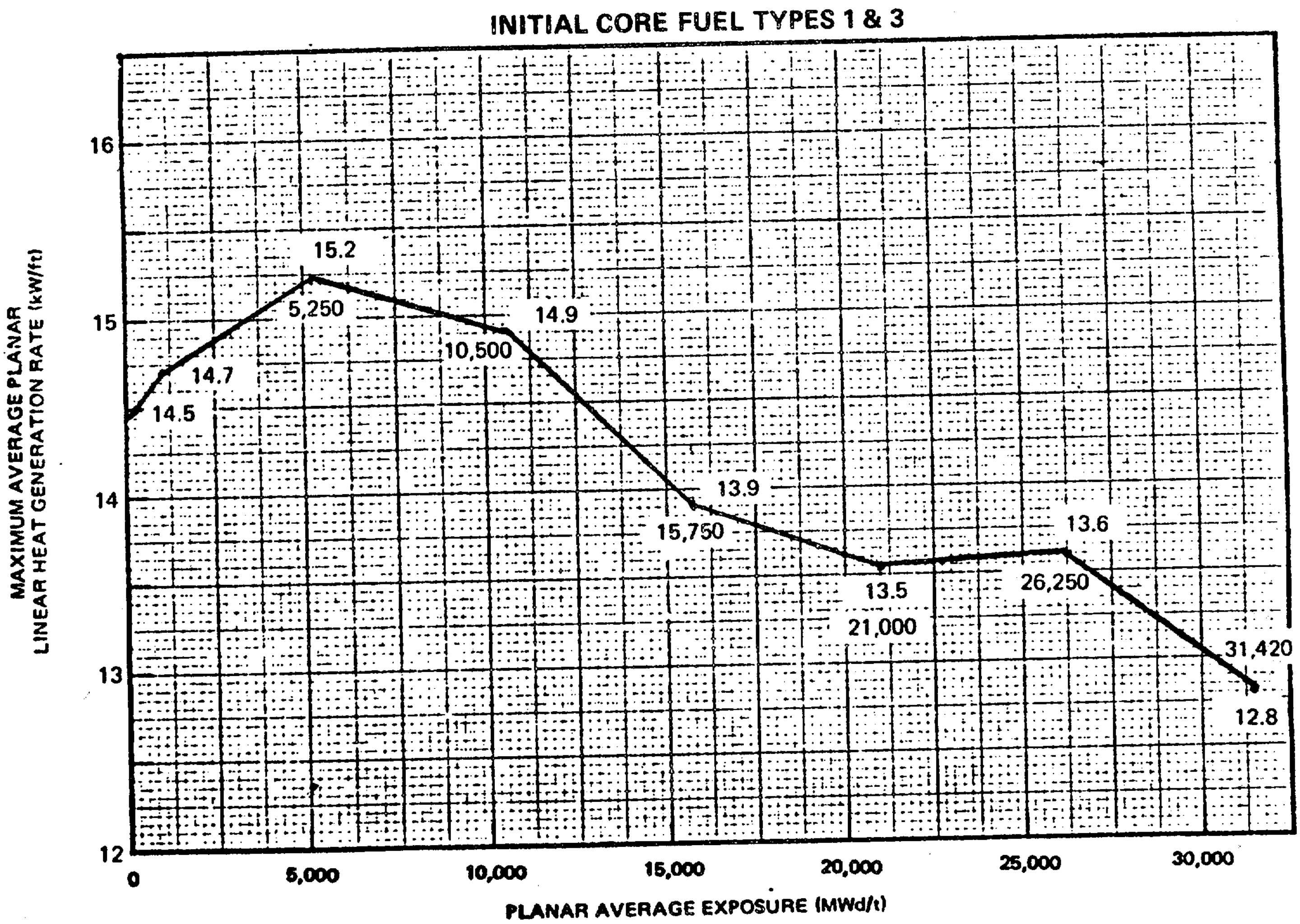

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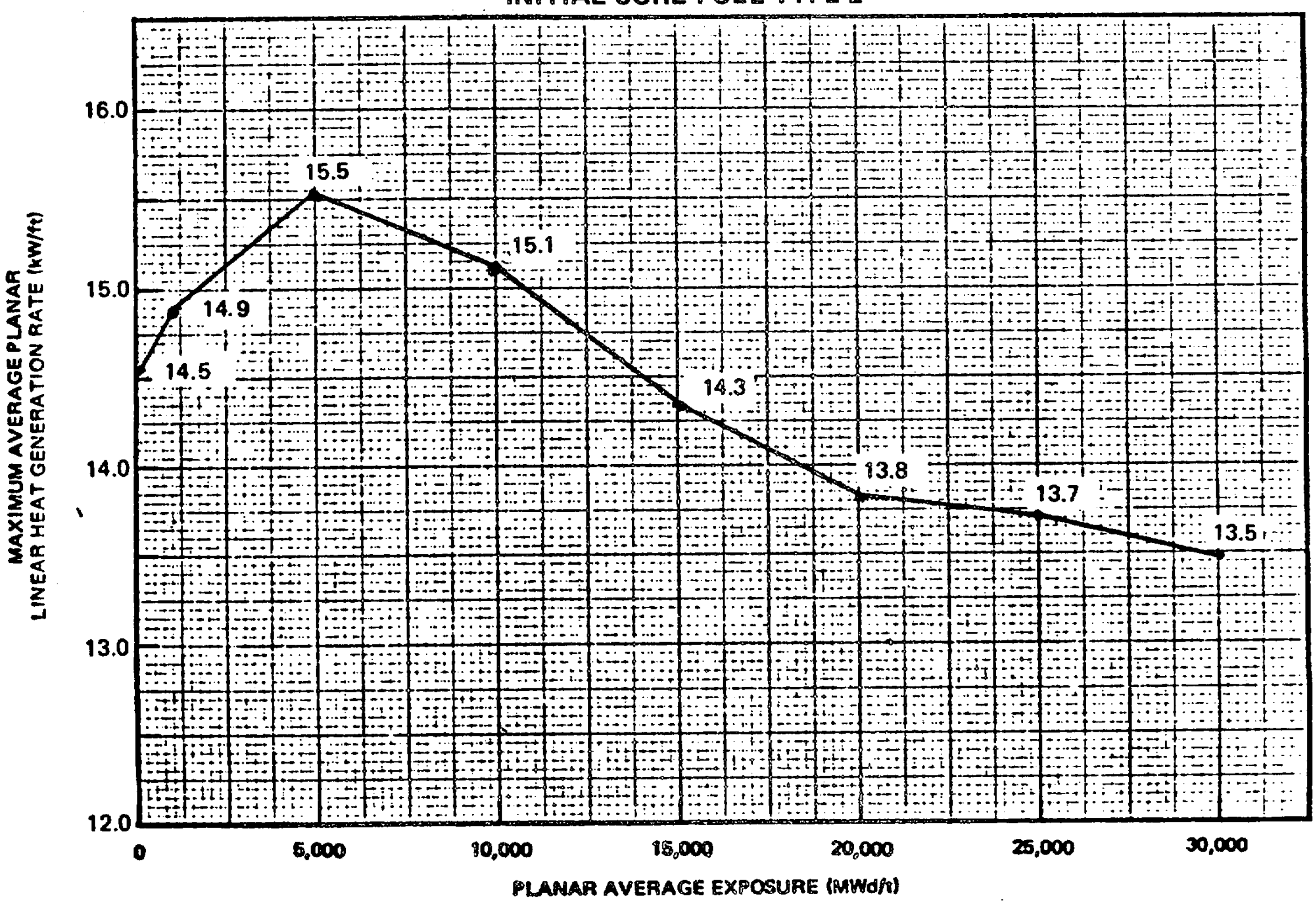
MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS PLANAR AVERAGE EXPOSURE

FIG. NO. 3.1–2A

AUGUST 1975

INITIAL CORE FUEL TYPE 2



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MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS PLANAR AVERAGE EXPOSURE	
FIG. NO.	3.1–2B