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VOLUME 4

EXXON NUCLEAR METHODOLOGY FOR BOILING WATER REACTORS: APPLICATION OF THE ENC METHODOLOGY TO BWR RELOADS

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EXXON NUCLEAR COMPANY, Inc.

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EXXON NUCLEAR METHODOLOGY FOR BOILING WATER REACTORS: APPLICATION OF THE ENC METHODOLOGY TO BWR RELOADS

This is the NRC approved version of Document XN-NF-80-19(NP) Volume 4, and has been prepared in accordance with NRC guidance.

EXXON NUCLEAR COMPANY, Inc.

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

AUG 3 1 1983

Mr. J. C. Chandler
Exxon Nuclear Company, Inc.
P. O. Box 130
Richland, Washington 99352

Dear Mr. Chandler:

Subject: Acceptance for Referencing of Licensing Topical Report XN-NF-80-19(P), Volume 4, "Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads"

We have completed our review of the subject topical report submitted October 13, 1982 by Exxon Nuclear Company, Inc. (ENC) letter JCC:098:82. We find this report is acceptable for referencing in license applications to the extent specified and under the limitations delineated in the report and the associated NRC evaluation which is enclosed. The evaluation defines the basis for acceptance of the report.

We do not intend to repeat our review of the matters described in the report and found acceptable when the report appears as a reference in license applications except to assure that the material presented is applicable to the specific plant involved. Our acceptance applies only to the matters described in the report.

In accordance with procedures established in NUREG-0390, it is requested that ENC publish accepted versions of this report, proprietary and non-proprietary, within three months of receipt of this letter. The accepted versions should incorporate this letter and the enclosed evaluation between the title page and the abstract. The accepted versions shall include an -A (designating accepted) following the report identification symbol.

Should our criteria or regulations change such that our conclusions as to the acceptability of the report are invalidated, ENC and/or the applicants referencing the topical report will be expected to revise and resubmit their respective documentation, or submit justification for the continued effective applicability of the topical report without revision of their respective documentation.

Sincerely,

Dino C. Scaletti for

Cecil O. Thomas, Chief
Standardization & Special
Projects Branch
Division of Licensing

Enclosure:
As stated

EVALUATION OF REPORT XN-NF-80-19(P), VOL. 4

Report Number:	XN-NF-80-19(P), Volume 4
Report Title:	Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads
Report Date:	October 1982
Originating Organization:	Exxon Nuclear Company, Inc.
Reviewed by:	Core Performance Branch, DSI

The Exxon Nuclear Company has submitted a technical report which presents the format and outlines the content of reports and analyses which establish the bases for acceptable plant operation. The report also describes the application of the various segments of Exxon Nuclear's BWR methodology in providing generic and plant specific analysis. The report is to be applied in the next few years to the reloads of a series of BWRs the first of which was Dresden Unit 2, Cycle 9. The report includes sections dealing with: fuel mechanical design analysis, thermal hydraulic design analysis, nuclear design analysis, evaluation of anticipated operational occurrences, analysis of postulated accidents and technical specifications.

Each of the sections conforms in its content with the corresponding topical reports describing the analyses and evaluation. A separate section references all of the BWR Exxon topical reports. Some of these reports have not yet been approved or are currently under staff review. The format used for the report includes all the quantities of interest to the technical reviewers. The establishment of this format will greatly facilitate future BWR reload reviews by standardizations of the reload report format.

Conclusion

The report presents the format and outlines the contents of BWR reload reports and the application of the Exxon BWR reload methodology. The outline is in agreement with the contents of the reload methodology and the format is acceptable, hence, the report is acceptable for reference for future reload submittals.

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EXXON NUCLEAR METHODOLOGY FOR

BOILING WATER REACTORS:

APPLICATION OF THE ENC METHODOLOGY TO BWR RELOADS

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PLEASE READ CAREFULLY

This technical report was derived through research and development programs sponsored by Exxon Nuclear Company, Inc. It is being submitted by Exxon Nuclear to the USNRC as part of a technical contribution to facilitate safety analyses by licensees of the USNRC which utilize Exxon Nuclear-fabricated reload fuel or other technical services provided by Exxon Nuclear for light water power reactors and it is true and correct to the best of Exxon Nuclear's knowledge, information, and belief. The information contained herein may be used by the USNRC in its review of this report, and by licensees or applicants before the USNRC which are customers of Exxon Nuclear in their demonstration of compliance with the USNRC's regulations.

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1.0 INTRODUCTION

The introduction of nuclear fuel fabricated by Exxon Nuclear Company (ENC) into the core of a boiling water reactor (BWR) requires assurance that the reactor will continue to meet accepted safety criteria during anticipated operation and accident conditions and that the ENC-fabricated fuel is compatible with existing fuel in the reactor core. In providing that assurance, ENC performs analyses in the areas of normal operation, anticipated operational occurrences, and postulated accidents which confirm or modify operating procedures, setpoints and limits.

The methodology used for these analyses is described in licensing topical reports issued by ENC. A complete bibliography of these reports is given in Section 8.0. This report presents the format and outlines the content of reports and analyses which establish the bases for acceptable plant operation. This report also describes the application of the various segments of Exxon Nuclear's BWR methodology in providing generic and plant specific analyses.

1.1 ANALYSES OF NORMAL OPERATION

Analyses for normal operation of the reactor include fuel evaluations in the areas of mechanical design, thermal hydraulic design, and nuclear design. To the maximum extent practical, ENC performs generic analyses of normal operation. Because of reactor, fuel design, and operating differences, much of the analyses supporting each part of the normal operation of the fuel is plant and cycle specific.

1.1.1 Mechanical Design Analysis

Mechanical design analysis of fuel fabricated by ENC is described in Section 2.0.

1.1.2 Thermal Hydraulic Design Analysis

Thermal hydraulic design analysis of fuel fabricated by ENC is described in Section 3.0.

1.1.3 Nuclear Design Analysis

Nuclear design analysis of fuel fabricated by ENC is described in Section 4.0.

1.2 ANALYSIS OF ANTICIPATED OPERATIONAL OCCURRENCES

Events of moderate and low frequency are analyzed to establish appropriate operating limits and to demonstrate that limits regarding fuel are not exceeded during such operation. Section 5.0 of this document defines several general classes of anticipated operational occurrences and identifies the specific methodology to be used in analyzing the limiting event (or events) in each classification.

Reactors are required to be operated such that events which are expected to occur with moderate frequency (i.e., expected to occur one or more times during the lifetime of the plant) will not result in exceeding the design limits for prevention of fuel failures. For infrequent events (i.e., events which are not expected to occur during the lifetime of the plant), the reactor must be operated such that although some fuel failures are possible during such an event, the radioactive release will be limited to a small fraction of the limits specified in 10 CFR 100. The treatment of infrequent and moderately frequent events is included in Section 5.0.

1.3 ANALYSIS OF POSTULATED ACCIDENTS

Analysis of postulated Loss of Coolant Accidents (LOCAs) is accomplished in accordance with Appendix K to 10 CFR 50 as required for compliance with 10 CFR 50.46. These analyses are undertaken to verify that operation with ENC-fabricated fuel satisfies the criteria of 10 CFR 50.46. Generic treatment of LOCA analyses is described in Section 6.0.

Analysis of the rapid withdrawal of a high-worth control element from the core is accomplished to assure that excessive energy will not be deposited in the fuel during the withdrawal. The Control Rod Drop Accident is described generically in Section 6.0.

1.4 SPECIFICATION OF OPERATING LIMITS

Addition of ENC-fabricated fuel and application of ENC analytical methods to a BWR core requires limited revision to the plant Technical Specifications. Operating limits are defined consistently for all BWRs, so the definition of operating limits allows generic treatment. Definition of operating limits is described in Section 7.0.

1.5 PLANT SPECIFIC SUBMITTALS

Plant and cycle specific reload analyses are reported in the format established by this document in a single summary report termed the Reload Analysis. Separate technical reports document detailed results of the analyses covering anticipated operational occurrences and postulated accidents. The organization of the Reload Analysis corresponds numerically to the format of this report. The Reload Analysis is described in Appendix A.

1.6 APPLICABILITY

The generic analyses reported in this document are applicable to jet pump BWR power plants utilizing Nuclear Steam Supply Systems designed and built by General Electric Company [

]

2.0 FUEL MECHANICAL DESIGN ANALYSIS

The mechanical design of BWR fuel fabricated by ENC is described on a generic basis in XN-NF-81-21 (Reference 9.1). This reference document addresses design bases, descriptions and design drawings, and plans for testing, surveillance and inspection. The design bases and evaluations establish criteria for the determination of fuel system damage and assure the following:

- Normal operation and anticipated operational occurrences do not result in the violation of established criteria; and
- ENC analyses of postulated accidents do not underestimate the number of fuel rod failures.

The cycle specific Reload Analysis verifies that the conditions of the generic mechanical design analysis are applicable to the core and cycle in question.

3.0 THERMAL HYDRAULIC DESIGN ANALYSIS

Thermal hydraulic analyses of the fuel and core are performed to verify that design criteria are satisfied and to establish an appropriate value for the MCPR Fuel Cladding Integrity Safety Limit. The analytical methods used by ENC for these analyses are described in Volume 3 of XN-NF-80-19 (Reference 8.6).

3.1 DESIGN CRITERIA

Primary thermal hydraulic design criteria of ENC reload fuel for BWR's are as follows:

3.1.1 Hydraulic Compatibility

The hydraulic flow resistance of the reload fuel assemblies shall be similar to existing fuel in the reactor so that there is no significant impact on total core flow or the flow distribution among assemblies in the core.

3.1.2 Thermal Margin Performance

The fuel design shall fall within the limits of applicability of the XN-3 Critical Power Correlation. Fuel assembly design shall minimize the likelihood of boiling transition during anticipated reactor operation.

3.1.3 Fuel Centerline Temperature

Fuel design and operation shall be such that fuel centerline melting is not expected for normal operation and anticipated operational occurrences.

3.1.4 Rod Bow

Anticipated magnitude of fuel rod bowing under irradiation shall be accounted for in establishing thermal margin requirements.

3.1.5 Bypass Flow

The bypass flow characteristics of the reload fuel assemblies shall not differ significantly from the existing fuel in order to provide adequate flow in the bypass region.

3.2 HYDRAULIC CHARACTERIZATION

This section describes the evaluations performed to verify that Exxon Nuclear's BWR reload fuel meets the stated thermal hydraulic design criteria.

3.2.1 Hydraulic Compatibility

Component hydraulic resistances for the ENC reload fuel design and representative General Electric (G.E.) design fuel have been determined in single phase flow tests of full scale assemblies.

Table 3.1 summarizes the component flow resistances for the two designs. The test results have been adjusted to account for the differences between the tests and actual reactor operating conditions. Figure 3.1 illustrates the relative hydraulic demand of the two bundle types when evaluated concurrently in a typical BWR core. The close similarity between the two fuel designs' performance characteristics indicate that they are sufficiently compatible for coresidence in a BWR core.

3.2.2 Thermal Margin Performance

Relative thermal margin performance has been evaluated by considering three configurations of a typical BWR core: a mixed core, a core composed of exclusively ENC fuel, and a core containing only G.E. fuel. All three configurations were evaluated for operating MCPR using the same core thermal hydraulic conditions and power distribution. The XN-3 critical power correlation was applied to all fuel bundles.

The resulting calculated values for operating MCPR are shown in Table 3.2. The close agreement between the fuel types in the different cases demonstrates the applicability of operating MCPR levels associated with the use of ENC fuel coresident with G.E. fuel.

3.2.3 Fuel Centerline Temperature

Fuel rod centerline temperatures are determined at steady state 120% overpower conditions as a check against the occurrence of calculated centerline melting during anticipated operational occurrences. This analysis is performed with RODEX2 (Reference 8.13) as part of the fuel mechanical design analysis. The operating temperature, the overpower temperature, and the fuel melting temperature are calculated for the expected operating lifetime of the fuel. If the operating power history is determined to differ from that established in the generic mechanical design report, the fuel centerline temperature analysis is performed on a plant specific basis.

3.2.4 Rod Bow

Post-irradiation examination of BWR fuel fabricated by ENC has shown that the magnitude of fuel rod bowing is very small. No impact on thermal margins is expected from these small dimensional changes.

3.2.5 Bypass Flow

The bypass flow fraction is calculated on a plant specific basis.

3.3 MCPR FUEL CLADDING INTEGRITY SAFETY LIMIT

The MCPR Fuel Cladding Integrity Safety Limit is calculated for each cycle using the methodology described in XN-NF-524 (Reference 8.10). The uncertainties identified in Table 3.3 are applied generically to all BWR safety limit analyses. These generic uncertainties are described in the documents referenced in the table.

Plant specific conditions which contribute to the MCPR safety limit include the initial thermodynamic state of the coolant, the design basis radial power distribution, and the design basis local power distribution.
[

]

3.3.1 Coolant Thermodynamic Condition

The thermal hydraulic response of the core is based on analyses with ENC's multi-channel BWR thermal hydraulic code XCOBRA (Reference 8.6). The absolute values of core thermal power, core inlet flow rate, and reference pressure are determined from reactor operating characteristics.

3.3.2 Design Basis Radial Power Distribution

Nuclear fuel management analyses provide expected limiting radial power distributions for the operating cycle. [

]

[ ] The resulting radial power histogram is used in the safety limit analysis.

3.3.3 Design Basis Local Power Distribution

[ ]

3.3.4 Treatment of Uncertainties

In establishing the MCPR safety and operating limits, ENC applies the criterion that during normal operation and anticipated operational occurrences at least 99.9% of the fuel rods in the core would be expected to avoid boiling transition. Operation above the MCPR Fuel Cladding Integrity Safety Limit assures that the criterion is met for normal operation. Operation at or above the MCPR operating limit assures that the criterion is met for anticipated operational occurrences. The methodology used for evaluation of the MCPR safety limit is described in XN-NF-524 (Reference 8.10).

The methodology used for determination of the change in MCPR associated with the limiting transient event is described in XN-NF-79-71 (Reference 8.8). After the conservative statistical characteristics of the change in MCPR are determined, the Limiting Transient ΔCPR is defined such that the occurrence of the limiting transient would not result in a greater decrease in MCPR in at least 95% of the random statistical combinations of uncertainties (Reference 8.12). The sum of the MCPR safety limit and the Limiting Transient ΔCPR defines the MCPR operating limit.

Table 3.1 Hydraulic Characterization Comparison Between ENC 8x8 and GE 8x8 Fuel

	ENC	GE 8x8R
Lower Tie Plate Loss Coefficient (K_{LTP})	[ ]	[ ]
Upper Tie Plate Loss Coefficient	[ ]	[ ]
Spacer Loss Coefficient	[ ]	[ ]
Bare Rod Friction Factor	[ ]	[ ]

---

* This is at Reynolds number of 200,000. More generally, the difference between GE and ENC lower tie plate pressure losses as referenced to the GE 8x8R bare rod flow area is given by:

$$\Delta K_{LTP} = K_{LTP_{GE}} - \frac{A_{BR_{GE}}}{A_{BR_{ENC}}} K_{LTP_{ENC}}$$

$$= [\quad]$$

where: K_{LTP} = Lower tie plate pressure loss coefficient

Δ_{BR} = Bare rod flow area

Table 3.2 Critical Power Ratio Results for Different Core Configurations

		ENC 8x8	GE 8x8
Case 1	Mixed Core	1.536	1.486
Case 2	All G.E. Core	--	1.495
Case 3	All ENC Core	1.509	--

Table 3.3 Uncertainties Considered in the MCPR Safety Limit

Parameter	Standard Deviation*	Reference
Feedwater Flow Rate	0.0176	8.10
Feedwater Temperature	0.0076	8.10
Core Pressure	0.0050	8.10
Total Core Flow Rate	0.0250	8.10
Core Inlet Enthalpy	0.0024	8.10
XN-3 Critical Power Correlation	0.0411	8.9
Assembly Flow Rate	0.0280	8.10
Power Distribution		
Radial Peaking Factor	0.0528	8.1
Local Peaking Factor	0.0246	8.1

*Fraction of nominal value

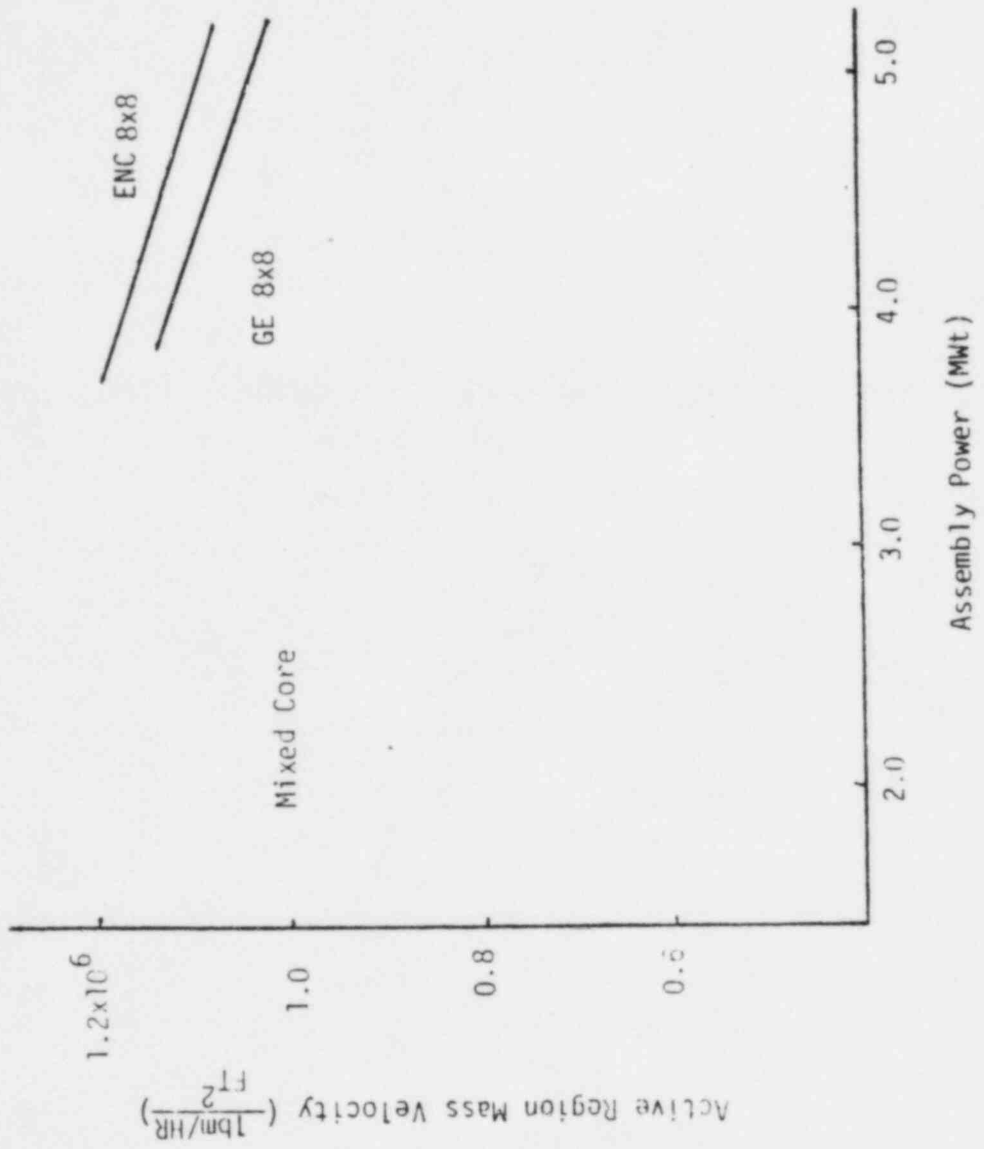

Figure 3.1 Typical Hydraulic Demand Curves

4.0 NUCLEAR DESIGN ANALYSIS

The nuclear design analyses are subdivided into two parts: a fuel bundle nuclear design analysis and a core nuclear design analysis. The bundle nuclear design analysis is specific to the bundle and does not vary unless the fuel design changes. The core nuclear design analysis is specific to the core configuration and may change if the fuel design changes or if the core configuration is changed by such mechanisms as a change in the relative fuel loading pattern or the removal of an existing fuel type from the core inventory. The methodology used by ENC for the nuclear design analyses is described in XN-NF-80-19, Volume 1 (Reference 8.1).

4.1 FUEL BUNDLE NUCLEAR DESIGN ANALYSIS

The fuel bundle nuclear design characteristics are considered for each ENC fuel bundle design added to the core. The key characteristics to the nuclear design analysis include the following items:

- Assembly average enrichment;
- Radial and axial enrichment distribution;
- Burnable poison content and distribution;
- Nature and location of non-fueled rods; and
- Neutronic design parameters.

The neutronic design parameters include descriptions of the fuel pellet, fuel rod, and fuel assembly assumptions used in the design analysis regarding core configuration, operating parameters, and control rods.

4.2 CORE NUCLEAR DESIGN ANALYSIS

The core nuclear characteristics are calculated for the reference core configuration for each operating cycle.

4.2.1 Core Configuration

For purposes of nuclear design analyses, the core configuration is assumed as a reference fuel loading pattern and fuel bundle inventory. Analyses for mixed core configurations explicitly consider all fuel types resident in the core. Core average exposure values corresponding to the end of the previous cycle and the beginning and end of the present cycle are calculated based on the assumed core configuration. The specific core loading pattern is established during the reactor refueling period, and supplemental nuclear design analyses are performed if the actual core configuration differs from the assumed configuration.

4.2.2 Core Reactivity Characteristics

Core reactivity characteristics are calculated for the reference core configuration. These characteristics include the following items:

- Cold shutdown margin at beginning of cycle;
- Cold excess reactivity at beginning of cycle;
- Cold shutdown margin with highest-worth rod withdrawn from the core;
- Reactivity defect (R-value); and
- Standby liquid control system shutdown margin.

4.2.3 Control Rod Patterns

Representative operating control rod patterns for the cycle are determined in the nuclear fuel management analysis.

4.2.4 Core Hydrodynamic Stability

The stability of the reactor core is verified through the decay ratio, which is calculated using COTRAN as a function of core power and recirculation flow state. The evaluation of core stability includes decay ratio values for natural circulation and operation along the nominal flow control line. Where reactor operation is expected above the nominal flow control line, the analysis also includes the highest allowed operating power-flow line. Acceptable core stability is demonstrated if the highest calculated value of the decay ratio is less than 1.00, as exemplified in Figure 4.3 of Appendix A.

5.0 EVALUATION OF ANTICIPATED OPERATIONAL OCCURRENCES

Analyses are performed to demonstrate that the fuel performs within design criteria for boiling transition during infrequent and moderately frequent anticipated operational occurrences and to establish appropriate operating limits for the reactor. The methodology used for the analysis of these anticipated events has been reported in References 8.1, 8.6, 8.8 and 8.12. The purpose of this section is to identify the potentially limiting events which require evaluation for each operating cycle.

To prevent or minimize boiling transition, the operating limits established by the evaluation of anticipated operational occurrences consist of a limiting transient ΔCPR, which in turn defines the MCPR operating limit, and a reduced flow MCPR limit function which adjusts the MCPR operating limit at reduced flow settings to allow for the consequences of events which are more severe at reduced flow. These analyses may also require a reduced power MCPR limit function which protects the core from the consequences of a control rod withdrawal error from less than full power conditions.

5.1 ANALYSIS OF PLANT TRANSIENTS AT RATED CONDITIONS

Anticipated operational occurrences involving the entire core and the recirculation system are evaluated at full power and flow conditions to determine the nominal MCPR operating limit. The limiting transient event (or events) is (are) evaluated using the plant transient methodology described in XN-NF-79-71 (Reference 8.8).

The evaluation of anticipated operational occurrences at rated conditions considers events in the following classifications:

- Rapid vessel pressurization;
- Decrease in recirculation flow rate;
- Increase in recirculation flow rate;
- Decrease in core inlet subcooling;
- Increase in core inlet subcooling;
- Decrease in vessel coolant inventory;
- Increase in vessel coolant inventory; and
- Combination events.

[

] Representative analyses of potentially limiting events in the above classifications for BWR/3 plants are contained in Reference 8.8.

5.2 ANALYSES FOR REDUCED FLOW OPERATION

The transient events described in the preceding section are most severe at full power conditions except [

]

[ ] Protection of the MCPR Fuel Cladding Integrity Safety Limit is assured during reduced flow operation through application of a flow-dependent MCPR operating limit which is established independently of the full flow MCPR limits through analyses of the flow-dependent transients from reduced power and flow settings.

The reduced flow MCPR limit is established [ ] to perform two protective functions. During operation in the Automatic Flow Control (AFC) mode, the limit assures that MCPR will not be below the MCPR operating limit if the flow control system demands an increase to full power and full flow. During operation in the Manual Flow Control (MFC) mode, the limit assures that MCPR will not be below the MCPR Fuel Cladding Integrity Safety Limit if the recirculation flow is inadvertently increased to the maximum allowed by the physical settings of the equipment.

Transient analyses are performed from various points on the power-flow operating map to demonstrate the adequacy of the flow-dependent MCPR limit to provide the desired degree of protection of the MCPR limits.

A special case of operation at less than rated power and flow is operation with a single recirculation loop out of service. It may be desirable to operate the reactor on a single loop if one component should require extensive maintenance. Analysis of single loop operation is performed on a plant specific basis, where needed.

5.3 ASME OVERPRESSURIZATION ANALYSIS

An overpressurization analysis is performed to assure that the vessel pressure requirements of the ASME Code are satisfied. This analysis, which presumes failure of the most critical active component and all non-safety grade components, does not contribute to the determination of thermal margin requirements.

[

.] The ASME overpressurization event is analyzed with COTRANSA.

5.4 CONTROL ROD WITHDRAWAL ERROR

Normal withdrawal of the highest worth control rod in the core until its movement is blocked by the control system is evaluated with XTGBWR, which is described in XN-NF-80-19, Volume 1 (Reference 8.1). The results are determined parametrically with rod block monitor setting. The setting which allows the greatest operational flexibility without restricting thermal margins is selected for implementation on a cycle specific basis.

Results for the control rod withdrawal error analysis include maximum control rod withdrawal distance, change in thermal margin, and the limiting control rod pattern used for the analysis. For reactors utilizing reduced power augmentation to MCPR limits, the existing reduced power limit functions are revised as necessary and verified for operation with ENC-fabricated fuel.

5.5 FUEL LOADING ERROR

The erroneous loading of a fuel bundle into the core and subsequent failure to detect the error is classified as an infrequent event.

5.5.1 Misloaded Fuel Bundle

The inadvertent misloading of a fuel bundle into an incorrect core location is analyzed with the XTGBWR methodology described in XN-NF-80-19, Volume 1 (Reference 8.1). The analysis identifies a maximum MCPR penalty and a maximum LHGR associated with the loading error.

5.5.2 Misoriented Fuel Bundle

The inadvertent rotation of a fuel bundle away from its intended orientation is evaluated with the XFYRE methodology described in XN-NF-80-19, Volume 1 (Reference 8.1). The analysis identifies a maximum MCPR penalty and a maximum LHGR associated with the orientation error.

5.6 DETERMINATION OF THERMAL MARGINS

The results of the anticipated operational occurrences evaluated under this chapter are compared for the greatest change in MCPR for full power operation. [

]

The Limiting Transient ΔCPR which is used to define the MCPR operating limit is used to select the rod block monitor setting from the tabulated results of the control rod withdrawal error analysis. Observance of the operating MCPR limit and rod block monitor settings determined in this fashion provides protection of the MCPR Fuel Cladding Integrity Safety Limit during operation at rated conditions.

The results of the reduced flow and reduced power analyses are used to establish proper values for the MCPR limit functions required for operation at lower than rated power and flow conditions. Reactor operation within the power- and flow-dependent limits defined in this fashion assures adequate protection of MCPR limits throughout the power-flow operating map.

5.7 SUPPORTING DOCUMENTATION

Results of the transient analyses at rated conditions are documented in the plant transient analysis report, which also reports the calculation of the MCPR Fuel Cladding Integrity Safety Limit.

[.] Control rod withdrawal error analyses at reduced power conditions are performed and reported on a generic basis for the classifications of BWR plants utilizing reduced power augmentation to MCPR limits.

Results of the cycle analyses described in this section are reported in the Reload Analysis.

6.0 ANALYSIS OF POSTULATED ACCIDENTS

Hypothetical Loss of Coolant Accidents (LOCA's) are analyzed in accordance with Appendix K modeling requirements using the ECCS models described in XN-NF-80-19, Volumes 2, 2A, and 2B (References 8.2, 8.3, and 8.4), XN-CC-33 (Reference 8.7), XN-NF-81-58 (Reference 8.13) and XN-NF-82-07 (Reference 8.14). Postulated Control Rod Drop Accidents are analyzed using the COTRAN methodology described in XN-NF-80-19, Volume 1 (Reference 8.1).

6.1 LOSS OF COOLANT ACCIDENT ANALYSIS

The ECCS analyses provide peak cladding temperature (PCT) and peak local metal-water reaction (MWR) values and are used to define MAPLHGR limits in accordance with 10 CFR 50.46. For each ENC fuel type, limiting break calculations are undertaken to determine the MAPLHGR, PCT, and MWR values over the expected exposure lifetime of the fuel. The limiting break is determined generically for each BWR type by evaluating a spectrum of potential break locations and sizes.

6.1.1 Break Location Spectrum

Representative LOCA analyses for piping breaks in the recirculation system piping form the basis for the location spectrum evaluation, which is accomplished on a generic basis for each major class of BWR plants. A figure of merit is drawn from MAPLHGR, PCT, and MWR values calculated for consistent exposure conditions in the fuel at each of the break spectrum locations. Analyses performed by the NSSS supplier are used as guidelines to narrow the scope of the analyses.

6.1.2 Break Size Spectrum

Once the location of the limiting break has been established, representative analyses are undertaken to establish the size of the limiting break. These analyses are performed on a generic basis for each major class of BWR plants.

Hypothetical piping system breaks are evaluated up to and including those with a break area equal to twice the cross sectional area of the largest pipe in the limiting break area. Due to physical phenomena observed during the blowdown phase of the LOCA analysis, the difference in results between guillotine pipe breaks and split pipe breaks is not significant for total break areas greater than 40% of the maximum guillotine break area. Smaller breaks are evaluated to assure that the largest break is also the most severe. As with the location spectrum, the determination of the limiting break size is based on a comparison of MAPLHGR, PCT, and MWR values for consistent exposure conditions in the fuel.

6.1.3 MAPLHGR Analyses

After the location and size of the limiting break have been determined, analyses are undertaken to characterize the maximum power at which the fuel may be operated without exceeding the ECCS limits specified in 10 CFR 50.46.

[

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[

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The blowdown phase is evaluated with RELAX (Reference 8.3). Refill and reflood are evaluated with FLEX (Reference 8.4). Fuel heatup is analyzed with HUXY (Reference 8.7). Stored energy and fuel response characteristics are determined with RODEX2 (Reference 8.13).

[

]

6.2 CONTROL ROD DROP ACCIDENT ANALYSIS

Analysis of the postulated Control Rod Drop Accident (RDA) is performed on a generic basis in XN-NF-80-19, Volume 1 (Reference 8.1). Because the behavior of the fuel and the core during such an event is not dependent upon system response, a single generic RDA analysis can be applied to all BWR types. The applicability of the generic RDA analysis is verified for each application.

The results of the generic RDA analysis consist of deposited fuel enthalpy values parameterized as a function of [

] Each application of the generic analysis includes the values for each of the parameters and the resulting deposited fuel enthalpy.

6.3 SUPPORTING DOCUMENTATION

The break spectrum analyses are performed and reported generically for each major classification of BWR types. The cycle specific Reload Analysis references the generic break spectrum analyses in specifying the limiting break location and size for the reactor in question. Detailed MAPLHGR analyses are reported separately and referenced in the Reload Analysis.

7.0 TECHNICAL SPECIFICATIONS

Technical Specifications are amended to assure that operation of the reactor is within safety criteria. Margins are established by the analyses of anticipated operational occurrences and postulated accidents described earlier in this document and in the plant and cycle specific supporting documentation. Technical Specification parameters are established in the following categories:

- Limiting safety system settings; and
- Limiting conditions for operation.

7.1 LIMITING SAFETY SYSTEM SETTINGS

Limiting Safety System Settings, or safety limits, are limits upon important control variables which are found to be necessary to reasonably protect the integrity of certain of the physical barriers which guard against the uncontrolled release of radioactivity.

7.1.1 MCPR Fuel Cladding Integrity Safety Limit

A minimum value for MCPR is established such that during sustained operation at the MCPR safety limit, at least 99.9% of the fuel rods in the core would be expected to avoid boiling transition. This limit is established as a protection against cladding failure.

7.1.2 Steam Dome Pressure Safety Limit

A maximum value for sensed pressure in the reactor vessel steam dome is established such that during the unlikely occurrence of the overpressurization accident as defined in the ASME Code the maximum pressure in the reactor vessel would not exceed 110% of the vessel design pressure.

7.2 LIMITING CONDITIONS FOR OPERATION

Limiting Conditions for Operation, or operating limits, are the lowest functional capability or performance levels of equipment required for safe operation of the facility.

7.2.1 Average Planar Linear Heat Generation Rate

Appendix K to 10 CFR 50 requires that the ECCS analysis be performed with the maximum peaking factor allowed by the Technical Specifications. Specifying a maximum value for average planar linear heat generation rate, or MAPLHGR, assures that the operating peaking factor remains within the limits of the ECCS analysis.

MAPLHGR is specified as a function of assembly average exposure over the expected lifetime of the fuel.

7.2.2 Minimum Critical Power Ratio

The thermal margin requirement established by the analyses of anticipated operational occurrences is added to the MCPR safety limit to determine a minimum operating value for MCPR. Observance of the MCPR operating limit assures that the occurrence of the limiting transient will not result in violation of the MCPR safety limit in at least 95% of the random statistical combinations of uncertainties. MCPR operating limits are established for each fuel type in the core.

MCPR operating limits which are based on rapidly developing transient events terminated by a scram trip are dependent on measured performance of the control rod drives. Procedures are provided for surveillance of scram insertion times and determination of the need to reevaluate thermal margin requirements if observed scram times are less favorable than those used in the transient analysis.

8.0 REFERENCES FOR EXXON NUCLEAR METHODOLOGY FOR BOILING WATER REACTORS

The following referenced reports describe the ENC methodology for the analysis of jet-pump boiling water reactors. They are incorporated into this submittal by reference.

8.1 XN-NF-80-19(A), Volume 1, May 1980
Exxon Nuclear Methodology for Boiling Water Reactors
Neutronics Methods for Design and Analysis

8.2 XN-NF-80-19(A), Volume 2, Revision 1, June 1981
Exxon Nuclear Methodology for Boiling Water Reactors
EXEM: ECCS Evaluation Model, Summary Description

8.3 XN-NF-80-19(A), Volume 2A, Revision 1, June 1981
Exxon Nuclear Methodology for Boiling Water Reactors
RELAX: A RELAP4 Based Computer Code for Calculating Blowdown Phenomena

8.4 XN-NF-80-19(A), Volume 2B, Revision 1, June 1981
Exxon Nuclear Methodology for Boiling Water Reactors
FLEX: A Computer Code for Jet Pump BWR Refill and Reflood Analysis

8.5 XN-NF-80-19(A), Volume 2C, June 1981
Exxon Nuclear Methodology for Boiling Water Reactors
Verification and Qualification of EXEM

8.6 XN-NF-80-19(P), Volume 3, Revision 1, April 1981
Exxon Nuclear Methodology for Boiling Water Reactors
THERMEX: Thermal Limits Methodology, Summary Description

8.7 XN-CC-33(A), Revision 1, November 1975
HUXY: A Generalized Multirod Heatup Code with 10CFR50
Appendix K Heatup Option

8.8 XN-NF-79-71(P), Revision 2, November 1981
Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors

8.9 XN-NF-512(A), Revision 1, March 1981
The XN-3 Critical Power Correlation

8.10 XN-NF-524(P), November 1979
Exxon Nuclear Critical Power Methodology for Boiling Water Reactors

8.11 XN-NF-79-59(P), October 1979
Methodology for Calculation of Pressure Drop in BWR Fuel Assemblies

8.12 XN-NF-81-22(P), September 1981
Generic Statistical Uncertainty Analysis Methodology

8.13 XN-NF-81-58(P), August 1981
RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model

8.14 XN-NF-82-07(P), January 1982
Exxon Nuclear Company ECCS Cladding Swelling and Rupture Model

9.0 ADDITIONAL REFERENCES

Although not specifically identified as part of the Exxon Nuclear Methodology for Boiling Water Reactors listed in Section 8.0, the following referenced documents provided generic analyses or other pertinent information.

9.1 S. F. Gaines, "Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel," XN-NF-81-21(A), Revision 1, January 1982.

9.2 J. E. Krajicek, "Generic Jet Pump BWR 3 LOCA Analysis Using the ENC EXEM Evaluation Model," XN-NF-81-71(A), October 1981.

9.3 U. S. Nuclear Regulatory Commission, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," NUREG-0800, July 1981.

9.4 General Electric Company, "General Electric Boiling Water Reactor Generic Reload Fuel Application," NEDO-24011-A, July 1979.

APPENDIX A

FORMAT OF THE PLANT-SPECIFIC RELOAD ANALYSIS

1.0 INTRODUCTION

This section provides a brief narrative describing the plant and cycle of interest and any additional items that may distinguish the submittal.

2.0 FUEL MECHANICAL DESIGN ANALYSIS

Applicable Fuel Design Report: Reference 9.1

Fuel Centerline Temperature

Exposure at Minimum Margin Point	________
Centerline Temperature at 120% Overpower	________
Melting Point of Fuel	________
Margin to Centerline Melting	________

3.0 THERMAL HYDRAULIC DESIGN ANALYSIS

3.2 HYDRAULIC CHARACTERIZATION

3.2.2 Thermal Margin Performance

Core Configuration	ENC Fuel MCPR	Existing Fuel MCPR
All Existing Fuel Core	NA	________
All ENC Fuel Core	________	NA
Mixed Core	________	________

3.2.5 Bypass Flow

Calculated Bypass Flow Fraction	________
Previous Cycle Bypass Flow Fraction	________

3.3 MCPR FUEL CLADDING INTEGRITY SAFETY LIMIT:	Reference 9.3
3.3.1 Coolant Thermodynamic Condition	
Core Power	________
Core Inlet Flow Rate	________
Steam Dome Pressure	________
Feedwater Enthalpy	________
3.3.2 Design Basis Radial Power Distribution	Figure 3.1
3.3.3 Design Basis Local Power Distribution	Figure 3.2

4.0 <u>NUCLEAR DESIGN ANALYSIS</u>

4.1 FUEL BUNDLE NUCLEAR DESIGN ANALYSIS	
Assembly Average Enrichment	________
Radial Enrichment Distribution	Figure 4.1
Axial Enrichment Distribution	________
Burnable Poisons	Figure 4.1
Non-Fueled Rods	Figure 4.1
Neutronic Design Parameters	Table 4.1
4.2 CORE NUCLEAR DESIGN ANALYSIS	
4.2.1 Core Configuration	Figure 4.2
Core Exposure at EOC__	____/____*
Core Exposure at BOC__	________
Core Exposure at EOC__	________
4.2.2 Core Reactivity Characteristics	
BOC Cold K-effective, All Rods Out	________
BOC Cold K-effective, All Rods In	________
BOC** Cold K-effective, Strongest Rod Out	________

* Nominal value/Value used in Shutdown Reactivity Calculations.
** Or worst exposure condition for cycle.

Reactivity Defect (R-value) ____________

SBLC Reactivity, Cold conditions, ___ ppm ____________

4.2.4 Core Hydrodynamic Stability — Figure 4.3

Maximum Decay Ratio Values

100% Flow Control Line ____________

Rod Block Line ____________

5.0 ANTICIPATED OPERATIONAL OCCURRENCES

Applicable Generic Transient Analysis Report — Reference 9.2

5.1 ANALYSIS OF PLANT TRANSIENTS AT RATED CONDITIONS — Reference 9.3

Limiting Transient(s): ____________________________

____

5.2 ANALYSES FOR REDUCED FLOW OPERATION — Reference 9.4

Limiting Transient(s): ____________________________

____

5.3 ASME OVERPRESSURIZATION ANALYSIS — Reference 9.3

Event ____________

Single Failure Assumed ____________

Maximum Pressure ____________

Maximum Sensed Pressure ____________

5.4 CONTROL ROD WITHDRAWAL ERROR

Starting Control Rod Pattern for Analysis — Figure 5.1

Rod Block Reading	Distance Withdrawn	ΔCPR
105	________	______
106	________	______
107	________	______
108	________	______

5.5 FUEL LOADING ERROR

ΔCPR ____________________

Max. LHGR ____________________

5.6 DETERMINATION OF THERMAL MARGINS

Event	Model	Exposure	Power	Flow	Maximum Heat Flux	Maximum Power	Maximum Pressure	Indicated MCPR Limit

MCPR Operating Limits at Rated Conditions

Fuel Type	MCPR Operating Limit

MCPR Operating Limits at Off-Rated Conditions: Figure 5.3*

6.0 POSTULATED ACCIDENTS

6.1 LOSS-OF-COOLANT ACCIDENT

6.1.1 Break Location Spectrum — Reference 9.5

6.1.2 Break Size Spectrum — Reference 9.5

6.1.3 MAPLHGR Analyses — Reference 9.6

Limiting Break: ____________________

*Format of figure is plant specific.

Bundle Average Exposure	MAPLHGR	Peak Clad Temperature	Peak Local MWR
____	____	____	____
____	____	____	____

6.2 CONTROL ROD DROP ACCIDENT — See XN-NF-80-19, Vol. 1

Dropped Control Rod Worth ____________ mk

Doppler Coefficient ____________ 1/K dk/dT

Effective Delayed Neutron Fraction ____________

Four-Bundle Local Peaking Factor ____________

Maximum Deposited Fuel Rod Enthalpy ____________ cal/gm

7.0 TECHNICAL SPECIFICATIONS

7.1 LIMITING SAFETY SYSTEM SETTINGS

7.1.1 MCPR Fuel Cladding Integrity Safety Limit

MCPR Safety Limit ____________

7.1.2 Steam Dome Pressure Safety Limit

Pressure Safety Limit ____________

7.2 LIMITING CONDITIONS FOR OPERATION

7.2.1 Average Planar Linear Heat Generation Rate

Bundle Average Exposure	MAPLHGR
____	____
____	____
____	____

7.2.2 Minimum Critical Power Ratio

Fuel Type	MCPR
__________	________
__________	________
__________	________

Reduced Flow MCPR Limits — Figure 7.1

Reduced Power MCPR Limits — Figure 7.2

7.2.3 Surveillance Requirements

Addressed on a plant- and cycle-specific basis.

9.0 ADDITIONAL REFERENCES

9.1 Applicable fuel design report.

9.2 Applicable generic transient analysis.

9.3 Plant transient analysis report.

9.4 Reduced flow transient analysis report.

9.5 Applicable generic LOCA analysis.

9.6 LOCA analysis report.

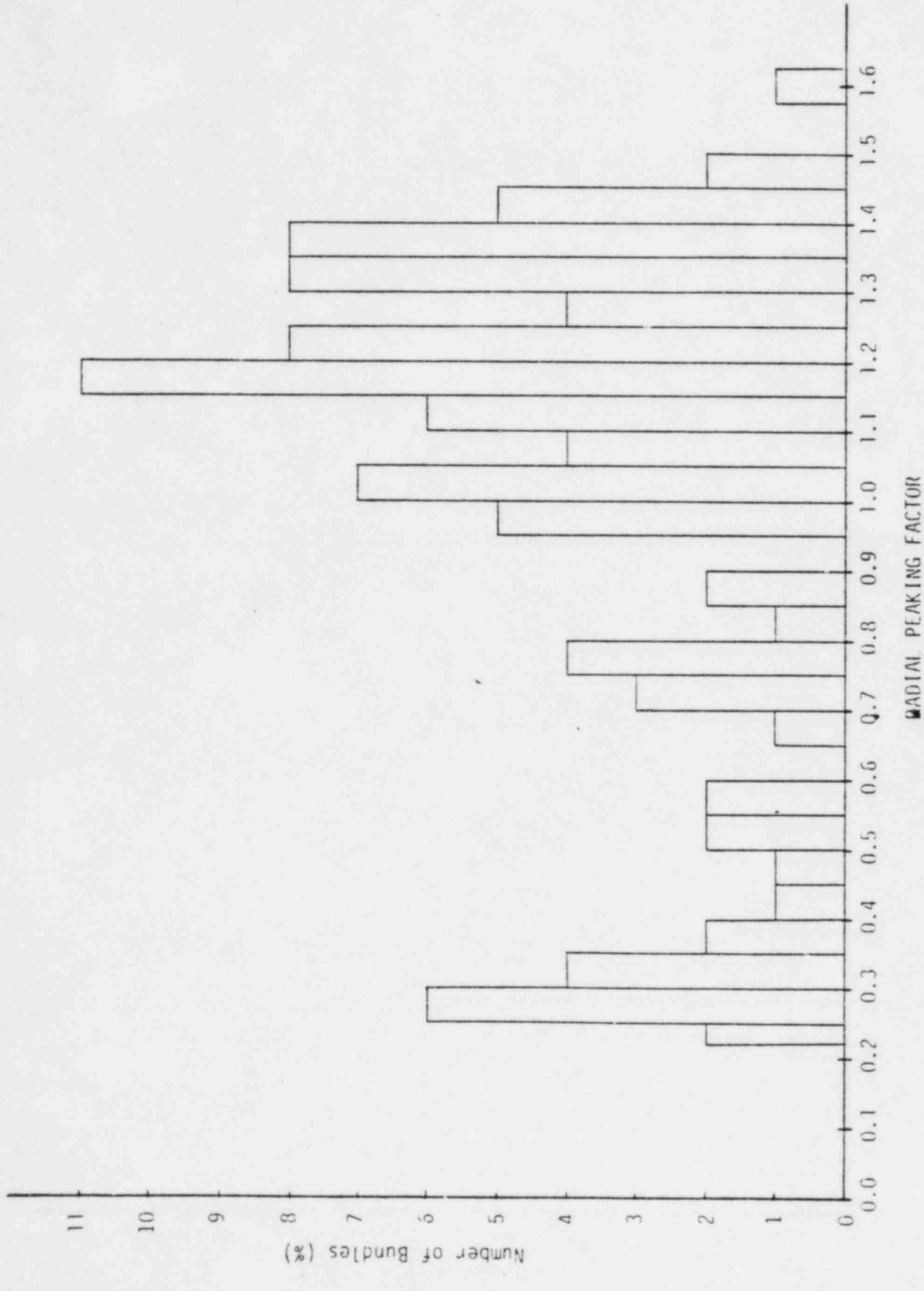

Figure 3.1 Typical Safety Limit Radial Power Histogram

L 1.06	ML 1.02	ML 0.92	M 1.08	M 1.06	M 1.08	ML 0.92	ML 1.01
ML 1.08	ML* 0.90	M 1.03	H 1.00	H 0.98	ML* 0.73	M 1.03	ML 0.92
ML 1.03	M 1.09	H 1.00	H 0.94	H 0.95	H 0.96	ML* 0.73	M 1.08
ML 1.01	M 1.06	H 0.97	H 0.94	W 0.00	H 0.95	H 0.95	M 1.06
ML [illegible]	M 1.08	H 0.99	H 0.94	H 0.94	H 0.94	H 1.00	M 1.08
ML 1.08	ML 0.93	M 1.03	H 0.99	H 0.97	H 1.00	M 1.03	ML 0.92
L 1.07	ML* 1.01	ML 0.95	M 1.08	M 1.06	M 1.09	ML* 0.90	ML 1.02
LL 1.01	L 1.07	ML 1.08	ML 1.03	ML 1.01	ML 1.03	ML 1.08	L 1.06

WIDE

WIDE

Figure 3.2 Typical Safety Limit Local Peaking

L	ML	ML	M	M	M	ML	ML
ML	ML*	M	H	H	ML*	M	ML
ML	M	H	H	H	H	ML*	M
ML	M	H	H	W	H	H	M
ML	M	H	H	H	H	H	M
ML	ML	M	H	H	H	M	ML
L	ML*	ML	M	M	M	ML*	ML
LL	L	ML	ML	ML	ML	ML	L

WIDE

WIDE

LL --- ______ W/O U235
L --- ______ W/O U235
ML --- ______ W/O U235
M --- ______ W/O U235
H --- ______ W/O U235
ML* --- ______ W/O U235 + ______ W/O GD2O3
W --- INERT WATER ROD

Figure 4.1 Typical Enrichment Distribution

C1	A2	C1	A2	C1	B2	C1	C1	B2	C1	C1	C1	C1	B4	A4
A2	C1	D0	C1	D0	C1	D0	B2	D0	B2	D0	B2	D0	B2	A4
C1	D0	A2	D0	B2	D0	B2	D0	C1	D0	C1	D0	C1	B2	A4
A2	C1	D0	C1	D0	B2	D0	B2	D0	B2	D0	C1	D0	B2	A3
C1	D0	B2	D0	B2	D0	C1	D0	C1	D0	C1	D0	C1	B3	A4
B2	C1	D0	B2	D0	B3	D0	B3	D0	B2	D0	C1	B2	B3	
C1	D0	B2	D0	C1	D0	C1	D0	C1	D0	C1	B3	B3		
C1	B2	D0	B2	D0	B3	D0	B3	D0	C1	D0	B2	A3		
B2	D0	C1	D0	C1	D0	C1	D0	C1	D0	B2	B2	B4		
C1	B2	D0	B2	D0	B2	D0	C1	D0	C1	B2	B4			
C1	D0	C1	D0	C1	D0	C1	D0	B2	B2	A3				
C1	B2	D0	C1	D0	C1	B3	B2	B2	B4					
C1	D0	C1	D0	C1	B2	B3	A3	B4						
B4	B2	B2	B2	B3	B3									
A4	A4	A4	A3	A4										

XY

X = Fuel Type
Y = Cycles Irradiated

Fuel Type	Number of Assemblies	Description

Figure 4.2 Typical Core Configuration

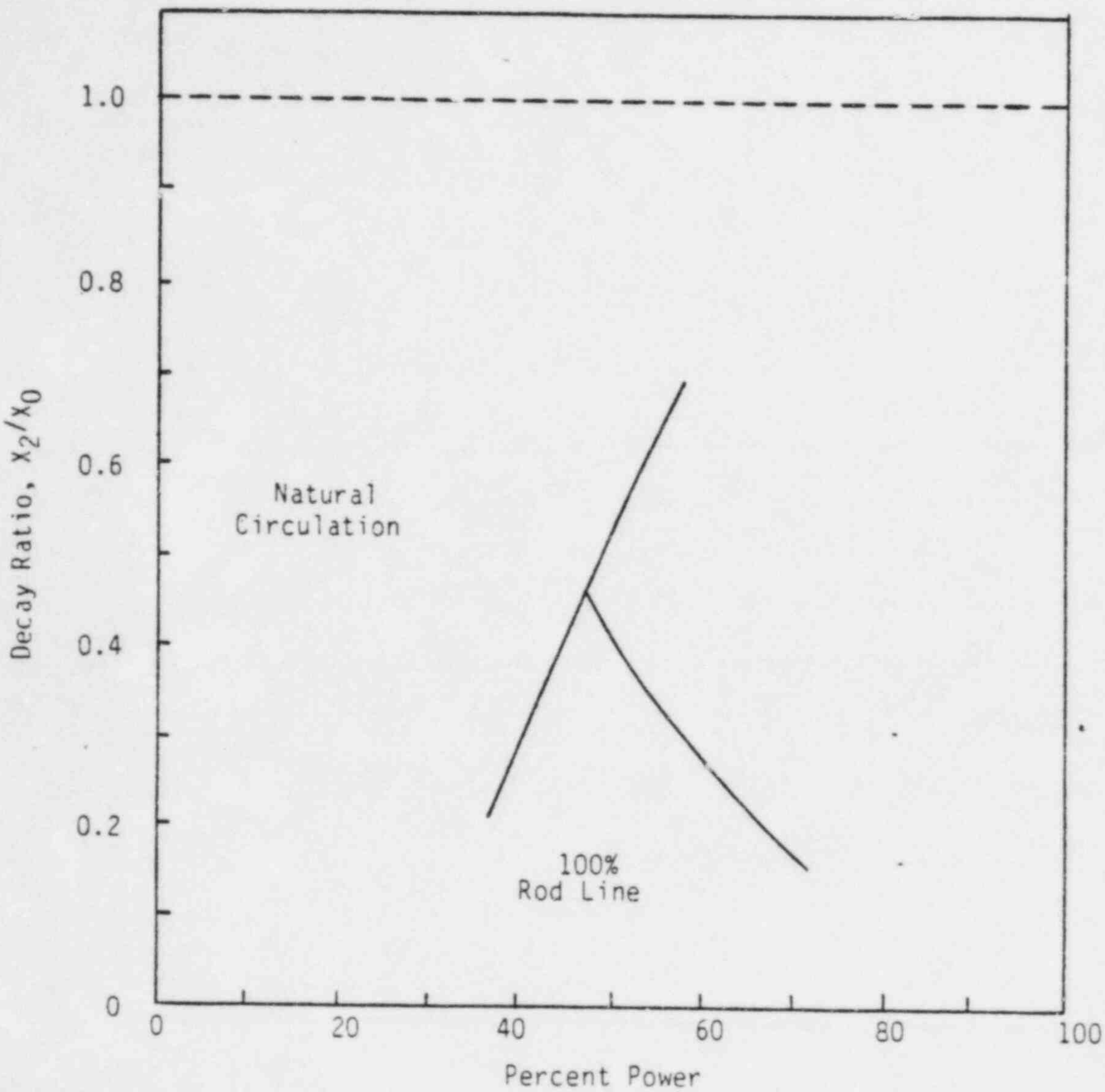

Figure 4.3 Decay Ratio vs. Reactor Power

Table 4.1 Neutronic Design Values

Fuel Pellet	Reference 9.1
Fuel Rod	Reference 9.1
Fuel Assembly	Reference 9.1
Core Data	
Number of fuel assemblies	
Rated thermal power, MW	
Rated core flow, 10^6 lbm/hr	
Core inlet subcooling, Btu/lbm	
Moderator temperature, ^oF	
Channel thickness, inch	
Channel inside face-to-face dimension, inch	
Fuel assembly pitch, inch	
Wide water gap thickness, inch	
Narrow water gap thickness, inch	
Control Rod Data	
Absorber material	
Total blade span, inch	
Total blade support span, inch	
Blade thickness, inch	
Blade face-to-face internal dimension, inch	

Table 4.1 Neutronic Design Values (Cont.)

Control Rod Data (Cont.)	
Absorber rods per blade	
Absorber rod outside diameter, inch	
Absorber rod inside diameter, inch	
Absorber density	

31								
27								
23								
19			0*					
15								
11								
07								
03								
	30	34	38	42	46	50	54	58

Note: *Control Rod Being Withdrawn, Rod Positions in Notches, Full In = 0, Full Out = Blank or 48

Figure 5.1 Starting Control Rod Pattern for Control Rod Withdrawal Analysis

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