Appendix B

Responses to NRC Questions

APPENDIX B

Table of Contents

REQUEST: Table of Contents

Complete the Table of Contents by adding all of the missing subsection headings (e.g., Subsection 2.4.1.2, **Safety Evaluation Results**). This will enable the user to more clearly understand the contents of the major sections and their interrelationships.

RESPONSE:

The Table of Contents is specifically limited to three digits so that it remains concise while directing the unfamiliar reader to the general area of interest. By adding detail, the Table of Contents would tend to become unwieldy to use and lose its effectiveness. Therefore, General Electric prefers to retain the current philosophy.

REQUEST: Section 1 – Introduction

The introduction is inadequate as presently written. It does not provide a clear presentation of the purpose, scope, philosophy, format, etc. of the Topical. The Topical will serve as a reference document for a broad range of reload reviewers. The introduction should be written to introduce the aforementioned aspects of the Topical. Specifically, it should briefly but clearly:

1. Discuss the Purpose of the Topical.

2. Discuss the Generic and Bounding analysis concepts.

3. Explain generic and bounding analysis Limitations and define those situations where Plant Specific analyses will be needed.

4. Discuss the relationship between the Topical and the Supplemental Reload Licensing Submittal.

5. Discuss the Scope (with regard to the plants involved).

6. Explain the purpose and applicability of the Thermal Margin Improvement Options Section relative to the other sections.

7. Explain the Applicability of the Physics Startup Test Section.

8. Explain when and why changes, addendums, revisions or supplements will be made and how they will be noted in the text.

RESPONSE:

The following information has been added to the applicable paragraphs in Section 1:

This report is intended to be a comprehensive reference document for plants adding General Electric reload fuel, and application of the report is limited to those plants.

The information contained in this report represents information that is independent of a plant specific reload application. Thus, efficiency is gained by not repeating information in individual reload submittals which are subjected to individual review.

The purpose and applicability of each section is contained in the initial paragraphs of that section.

Bounding analyses are performed where sufficient margins exist to allow analyses to be performed using parameters that are not expected to be exceeded in a group of reloads. Bounding analyses are not performed where they would result in a group of reloads. Bounding analyses are not performed where they would result in restricted operation. If any parameter in a given reload exceeds a parameter used in a bounding analysis, a new evaluation is required. This may take the form of a new analysis or sensitivity study which demonstrates that the amount that a parameter exceeds the bounding parameter is compensated for by another parameter being more conservative than the bounding value. Where a bounding analysis is not applicable, a plant specific analysis will be documented in the Plant Supplemental Submittal.

Until NRC approval, changes to this report will be indicated by brackets in the margins with the number of the amendment following the report number at the top of each page. When NRC approval is granted, the letter A will be added to the report identification number and a copy of the Safety Evaluation Report will be added as Appendix C. Appendix D will then be started and contain all requested changes noted in the above manner. As they are approved, they will be incorporated in the body of the approved report and the revised report distributed.

REQUEST: Subsection 2.1 Fuel Assembly Description

Discuss the safety significance of the changes made to the 8x8R fuel rod mechanical design as compared to the present 8x8 fuel rod mechanical design. That is, this section should include a qualitative discussion of the changes and the basis for the changes made to the radial and axial characteristics of the fuel rod. Comparison to the 7x7 characteristics should also be included. The effect of these changes on the fuel performance during normal (e.g., PCI, fuel duty) operation, anticipated transients (e.g., margin to 1% strain limit) and accident (e.g., LOCA PCT) should be briefly discussed. Include a discussion of the basis for the axial enrichment zoning of the UO_2 and UO_2 Gd_2O_3.

RESPONSE:

The 8x8R fuel design was introduced to realize an improvement in operating margins and uranium utilization. It should be recognized that 8x8R fuel represents only a very small change in fuel design (much less significant than previous changes). Both 8x8 and 8x8R fuel are analyzed using the same general models and methods and satisfy the same criteria. In each plant specific reload submittal, the safety analyses are performed using the same methods and satisfy the same criteria. Therefore, the introduction of 8x8R is not considered to have any significance with respect to plant safety. The effect of the fuel changes is described below.

The active fuel length is increased to 150 inches from 144 or 146 inches for BWR/4 reactors and to 145.2 inches from 144 inches for BWR/2 and 3 reactors. This is made possible by the lower fission gas release in the 8x8R design relative to the 7x7 fuel design, which allows a reduction in the fission gas plenum length originally sized for the 7x7 fuel. From this increase in active fuel length, the core average power density is slightly reduced. Natural uranium is incorporated into the top or the top and bottom 6 inches of some fuel rods. This change reduces the core average enrichment for a fixed energy production.

The number of water tubes was increased from one to two and their outside diameter was enlarged from 0.493 to 0.591 inch. The larger water rods tend to reduce the maximum local power factor, decrease the amount of fissile inventory required to achieve a fixed energy production, and reduce the magnitude of the void coefficient of reactivity. The increase in the ratio of nonboiling to boiling water results in an additional reduction in the magnitude of the void coefficient. The changes to the void coefficient, due to the two water tubes, flatten the axial power distribution, which more than offsets the effect of the natural uranium on the axial power peaking.

The fuel rod diameter is reduced from 0.493 to 0.483 inches. The 10 mil reduction in fuel rod diameter was accomplished by reducing the pellet diameter by 6 mils and decreasing the cladding thickness by 2 mils. The maximum linear heat generation rate is preserved at 13.4 kW/ft as a design basis; therefore, the maximum fuel centerline temperature at full power remains very nearly the same.

Information on 7x7 fuel was provided in the individual plant initial core or reload submittal in which it was introduced.

REQUEST: Subsection 2.1.2 Water Rods

Briefly describe the safety basis for introducing a second water rod into the fuel assembly design. Briefly discuss the thermal hydraulic performance improvements (if any) expected from the two water rod design, as compared to a single water rod design.

RESPONSE

See Response to request on Subsection 2.1.

REQUEST: Subsection 2.1.3 – Other Fuel Assembly Components

This Section should contain subsections referring to the:

- Fuel Spacer Grid
- Finger Springs
- Upper and Lower Tieplates
- Channel Box Fastener
- Channel Box

Each subsection should contain a brief discussion of the design and purpose of the component.

RESPONSE:

Dividing Subsection 2.1.3 into smaller subsections does not serve to clarify the information within this subsection. A discussion of the design and purpose of the fuel spacers, finger springs, and upper and lower tieplates is already given in this subsection. As stated in Subsection 2.1, channels are addressed separately in Reference 2 1.

REQUEST: Subsection 2.2 – Functional Requirements

Provide a table which shows the normal, abnormal and accident conditions which were analyzed to demonstrate that the safety related functional requirements are met for each component listed in Subsection 2.1.3. Include in the table footnotes which reference the (sub)sections in the Topical which discuss, in detail, the analyses, tests or evaluations performed to show that the safety-related functional requirements are satisfied.

RESPONSE:

Capability to withstand normal, abnormal, and handling loads is addressed in the appropriate subsection of the mechanical evaluation Subsection 2.5. Handling loads are documented in Subsection 2.5.3. Seismic considerations are bounded by the evaluation of LOCA and seismic loads in Subsection 2.5.4. Individual accident analyses are documented, as indicated, in Subsection 5.5.

REQUEST: Subsection 2.3.1 – Cladding

Provide a reference for the plastic (modulus) behavior of Zircaloy 2. Table 2-2, used in the safety analyses, should reference the constitutive equations for burnup dependent material properties utilized in the thermal-mechanical evaluations (e.g., Zr 2 creep equation, irradiated growth equation). Table 2-2 indicates a total irradiation elongation of greater than 1%. Is this (>) a typographical error? Give the fluence corresponding to the percent total irradiated elongation. Sections 2.6 and 2.7 should also be briefly summarized and referenced. References 2-4 and 2-21 discuss the 7x7 fuel design. Provide a more recent reference to the 8x8 and 8x8R fuel performance in this section. The last sentence referring to UO_2 properties appears to have been incorrectly placed in this subsection. Equation 2-1 implies that the 1% plastic strain limit is for a uniaxial strain (stress) field. Describe or provide a reference as to how the 1% plastic strain criterion for the uniaxial stress is related to the experimental results from cylindrical rods, which are subject to a triaxial state of stress during irradiation.

RESPONSE:

A plastic modulus or behavior of the fuel cladding is not used in the analyses performed. With respect to interference between the pellet and the cladding, the interference is presumed to be accommodated solely by an increase in cladding strain. This is equivalent to specifying an infinite pellet modulus, in which case the actual cladding modulus (elastic or plastic) is unimportant with respect to determining the resultant cladding system.

Cladding creep and irradiation growth in the radial direction are not explicitly treated in fuel rod thermal or mechanical evaluations but are implicitly considered through data comparisons as a part of the integral fuel rod model verification. This model has been described in detail in Reference 1, and model predictions compared to data in Reference 2-4. The relationship for cladding creep used in the cladding creep collapse analysis has been reported in Reference 2-16.

Credit is not taken for fuel rod irradiation growth in fuel rod analyses to determine fuel rod internal pressure; however, the differential axial growth among fuel rods is considered in the fuel assembly design. The relationships used for this axial irradiation growth plus plastic strain and creep of the fuel rod cladding have been added to Table 2-2. The equations are based on measurements of fuel rods operated in commercial BWR's and predict the upper and lower 95/95 tolerance limits of that data. More recent measurements on 8x8 fuel with pre irradiation measurements available, have indicated that these expressions are very conservative with respect to the 8x8 fuel design. The axial irradiation growth assumption used in densification analyses is defined in Reference 2-8.

The "total irradiated elongation" referred to in Table 2-2 was meant to describe the plastic strain capability of the cladding. The title used in Table 2-2 is being changed to avoid misunderstanding. The value provided is considered applicable over the range of peak pellet exposures expected during fuel lifetime (0–40,000 MWd/t).

A reference to Subsection 2.6 has been added to Subsection 2.3.1; Subsection 2.7 does not include material properties. Reference 2 4 was referred to in Subsection 2.3.1 because it provides supporting information in regards to the cladding plastic strain capability. Reference 2-21 did not provide any additional information with respect to cladding material properties. In Subsection 2.6.1, where fuel operating experience is discussed with respect to hydriding, a new reference (Reference 2) is being added which contains the most recent information on operating experience of BWR fuel from General Electric.

Equation 2-1 limits the diametral expansion of the fuel cladding by limiting the calculated plastic hoop strain to 1%. The calculated strain is based upon a one-dimensional (radial) model, which predicts only hoop strain. Reference 2-4 established a cladding damage limit of 1% plastic strain from data obtained from tests of irradiated zircaloy cladding. In addition to uniaxial tests, the test geometries included closed end burst specimens, which subject the tubing to a triaxial stress state. All of the burst specimens had a circumferential fracture elongation >1%, which supports a hoop strain limit of 1% in a multiaxial loading case.

REFERENCES:

1. "General Electric Standard Safety Analysis Report" Proprietary Supplement to Amendment 14, Docket No. STN-50-447, May 1974.

2. R. B. Elkins, "Experience with BWR Fuel through December 1976", NEDE-21660P (Proprietary) and NEDO-21660, July 1977.

REQUEST:

Subsection 2.3.2 – This section is missing. Please provide.

RESPONSE:

The subsection title was inadvertently omitted in the original submittal. This was corrected in Amendment 1.

REQUEST: Subsection 2.3.3 – Urania Gadolinia Fuel Pellets

Section 2.7.3 should be cross referenced for continuity.

RESPONSE:

Subsection 2.7.3 documents the manufacturing quality control of gadolinia and urania gadolinia fuel rods, while Subsection 2.3.3 provides the urania gadolinia fuel rod material properties. Thus, a cross reference is not provided.

REQUEST: Subsection 2.4 – Fuel Rod Thermal Analysis

State, in this paragraph, all the analyses performed under the heading "Fuel Rod Thermal Analysis" and which appear in the section (e.g., creep collapse, center melting). Provide a summary discussion of the distinction between the "Safety Evaluation" and the "Design Evaluations". Summarize the plant specific reload parameter review discussed in Subsection 2.4.3. State the fuel types analyzed in the Safety Evaluation (include the 7x7 type, i.e., for kW/ft for 1% strain).

RESPONSE:

Given the character of Subsection 2.4, adding summaries for each thermal evaluation is redundant. However, this subsection has been revised in response to the requests on Subsections 2.4.1 and 2.4.2 to clarify the generic thermal analyses performed and their relationship to other analyses. Documentation of specific analyses and reload parameters remains in the appropriate subsection.

REQUEST: Subsection 2.4.1 – Safety Evaluation

Subsections 2.4.1.1 and 2.4.1.2 do not by themselves provide a complete "safety evaluation" of the fuel rod thermal performance. Subsection 2.4.1 simply establishes a correspondence between the 1% plastic strain limit and the local LHGR limit. The "safety evaluation results" and conclusions depend upon the localized anticipated transients described in Section 5 and reported in the reload supplement. Therefore, provide new headings for these sections to reflect the limits, analyses and results appearing in this section. Provide a discussion of the connection between the kW/ft values reported in the tables and the transient analyses described in Section 5. Discuss the application of these kW/ft limits to the transient analysis results that are to be reported in the plant specific reload supplements.

RESPONSE:

This subsection provides the basis and the results of a generic safety evaluation of linear heat generation rate (LHRG) associated with the 1% plastic strain safety limit. Results provided in the subsection are used in conjunction with results from analyses of plant transients which are described in Section 5.0. The peak LHGR calculated for normal and abnormal operating transients must be less than or equal to the LHGR at which 1% plastic strain is calculated to occur.

An introduction has been added to Subsection 2.4.1 which discusses the safety analysis and application of the results.

REQUEST: Subsection 2.4.1.1 – Basis for Safety Evaluation

See comments on Subsection 2.4.1. Provide a reference or summary description of the code used for this evaluation. Discuss the important assumptions incorporated in the analysis; e.g., consideration of tolerances (see comments on Subsection 2.7.1), fuel swelling rate, densification, etc. Discuss how local effects such as pellet hour glassing and radial cracking are considered in the evaluation.

RESPONSE:

The model used in the evaluation of the 1% plastic strain limit is described in detail in Reference 1. Dimensions used in conjunction with this evaluation are the most limiting combination of tolerances. A discussion of the model and associated assumptions has been added to Subsection 2.4.1.1.

REFERENCES:

1. "General Electric Standard Safety Analysis Report", Proprietary Supplement to Amendment 14, Docket No. STN-50-447, May 1974.

REQUEST: Subsection 2.4.1.2 Safety Evaluation Results

See comments on Subsection 2.4.1. Discuss the application of the tabular LHGR's vs Exposure to the transient analysis results appearing in the reload supplement to the applicable transients discussed in Section 5. Provide in Table 2-3 LHGR limits for the 7x7 and 8x8 fuel designs. Discuss the differences between the fuel types. Discuss the reasons for the burnup effect.

RESPONSE:

Differences between beginning of life and end of life are primarily due to changes in: [[

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The values calculated as resulting in 1% plastic strain in the cladding are used during specific plant evaluations of transients due to single operator error or equipment malfunctions to ensure that the safety limit is not exceeded. (See Section 5.)

This discussion has been added to Subsection 2.4.1.2. In Table 2-3, the title has been changed to indicate that the results are applicable to both the 8x8 and 8x8R fuel designs. Results for 7x7 fuel are given is noted in the response to the request on Subsection 2.1.

REQUEST: Subsection 2.4.2 – Design Evaluations

Discuss the purpose of the "Design Evaluations" as compared to the "Safety Evaluation" appearing in the previous section. Provide a substantially expanded summary description of the integral design model. Give a reference for the code. Provide a comparison of the models and assumptions used in the Safety Evaluation and the Design Evaluations. Discuss the conservatisms in the assumptions models and methods (e.g., operating conditions, consideration of tolerances, etc). Explain why the resulting design evaluations are considered generic (bounding for all plants appearing in Table 1-1). Discuss the effects of varying plant to plant operating conditions. List in tabular form which design analyses will be reviewed on a plant specific reload application and which will not. Give the basis for omitting any design evaluations on a plant specific review. Provide tables which list each of the key fuel and plant operating parameter values used in each of the generic design evaluations.

Provide justification for the values selected relative to plant specific conditions for the plants listed in Table 1-1. Provide a complete list of the parameters which will be reviewed for each specific reload to assure that the generic evaluations are applicable. State the acceptance criteria for each of the parameters. What procedure will be followed if a particular reload application is not bounded by the generic evaluation? State what information will be provided in the reload submittal in these circumstances. Provide a general comparison of the past and projected thermal performance improvement among the 7x7, 8x8 and 8x8R fuel designs during normal operation.

RESPONSE:

Safety evaluations are performed and measured against established safety criteria. The consequence of calculating values which exceed such criteria is that fuel failure must be assumed to occur. For plant normal and abnormal operation, this is not permissible. There are two such established safety criteria for the fuel: (1) the 1% plastic strain safety limit and (2) the fuel cladding integrity safety limit.

Design evaluations are also performed and measured against established criteria. These nonsafety related evaluations are included in the report in response to previous specific questions from the NRC. The criteria are not intended to predict failure of the fuel if they are exceeded, but are considered prudent design practice. These design evaluations are included in Subsection 2.5. Thermal analyses performed to provide input into these design analyses are given in Subsection 2.4.2.

The models used for fuel rod thermal analyses have been described in References 1 and 2. The latter model is used only for the evaluation of fuel rod initial conditions at the initiation of a loss of coolant accident.

Design and safety evaluations performed using the model described in Reference 1 assume the most limiting combination of tolerances for all critical dimensions. Assumptions used in conjunction with the model described in Reference 2 are stated in the reference.

The above information has been included in Subsection 2.4.

Conditions which vary from one design to another, or from plant to plant, were taken into account in design evaluations. All dimensional, enrichment, or gadolinia variations among the designs were either analyzed separately or the most limiting case was analyzed. Operating conditions which vary from plant to plant are parameters such as core pressure and the expected maximum power vs exposure for the peak duty fuel rod. In the generic analysis, the highest core operating pressure was used which results in the highest coolant temperatures. With respect to power vs exposure, the limiting fuel rod is assumed to operate at its maximum permitted power over its entire lifetime, based on thermal mechanical limitations such as the stress limits presented in Table 2-6.

The power level and exposure range considered in the generic analyses, therefore, provides a basis for review of these same parameters on individual projects. For each reload fuel project, the following evaluations are performed to ensure the applicability of the generic analyses:

1. The performance of the reload fuel batch and other fuel already present in the core is projected for the specific plant.

2. The combinations of power, exposure, and residence time from this projected performance are compared with those used in the respective generic analyses. The application is acceptable only if the power levels and exposures analyzed in the generic analyses equal or exceed those protected for the project.

In addition, other plant dependent operating conditions such as core pressure are reviewed and compared with those used in the generic analyses to ensure that limiting conditions have not changed. Adherence to these procedures ensures the applicability of the generic design to individual reload fuel protects.

The above application procedure has been combined with the information in Subsection 2.4.3 and relocated in a new Subsection 2.7. The previous Subsection 2.7 now becomes Subsection 2.8.

A recent report on the performance of General Electric BWR fuel can be found in Reference 3. This reference has been included in Subsection 2.8.

REFERENCES:

1. “General Electric Standard Safety Analysis Report”, Proprietary Supplement to Amendment 14, Docket No. STN-50-447, May 1974.

2. “GEGAP III, A Model for the Prediction of Pellet Cladding Thermal Conductance in BWR Fuel Rods”, NEDC-20181, Revision 1, November 1973.

3. R.B. Elkins, “Experience with BWR Fuel through December 1976”, NEDE-21660P (Proprietary) and NEDO-21660, July 1977.

REQUEST: Subsection 2.4.2.1.1 – Power Spiking

Discuss the application of densification power spiking to Design and Safety Evaluation considerations (See Attachment 2).

RESPONSE:

See the response to the request on Attachment 2, Subsection 2.4.2.1.1.

REQUEST: Subsection 2.4.2.1.3 – Cladding Creep Collapse

It appears that the reference to Subsection 2.5.3.1.2 should be to Subsection 2.5.3.1.1. Make the correction. Reference 2-16 in (cross referenced) Subsection 2.5.3.1.1 refers to a generic pressure increase of 160 psi due to pressurization events, which is subsequently utilized for the collapse evaluation. Many turbine trip without bypass events previously analyzed for reloads report more than a 160 psi pressure increase. Justify the use of this magnitude of pressure increase for generic application to all reload cores. Provide a figure which gives the assumed DP and fast flux over the fuel lifetime for the collapse analysis. Discuss the treatment of the initial ovality, diameter and wall thickness tolerances used in the collapse analysis. Discuss the effects of a reactor vessel cold hydrostatic test on clad buckling potential. State the design criteria for fuel clad collapse during normal and abnormal operating conditions.

RESPONSE:

Analyses of the majority of abnormal operational transients will result in a pressure increase of 160 psi or less. There are a few infrequent events such as turbine trip without bypass or generator load rejection without bypass which may result in pressure increases greater than 160 psi on some operating plants. Studies indicate that the fuel rod is capable of withstanding transient differential pressure increases above rated conditions in excess of 250 psi without experiencing instantaneous collapse or subsequent creep collapse.

The model used for cladding creep collapse analysis utilizes maximum initial ovality, minimum wall thickness and nominal average internal diameter. The design criterion for fuel cladding collapse during normal and abnormal operating conditions is that collapse is not permitted to occur.

Cold hydrostatic test pressure is 1045 psia, which is below the 1065 psia design pressure.

REQUEST: Subsection 2.4.2.1.4 – Stored Energy

This section consists of one sentence and is inadequate. Provide a brief summary discussion in this section of the effect of densification on stored energy for LOCA analysis. (Subsection 5 5, relating to LOCA is Subsection 5.5.2 and had not yet been provided.) Provide or reference a discussion of the effect of the increase in stored energy on the consequences of a control rod drop accident from hot full power. Discuss the radial contraction model (for predicting stored energy) for the control rod drop accident form hot full power.

RESPONSE:

The effects of densification on stored energy are considered in the LOCA evaluation. Stored energy in the fuel pellet at the initiation of the LOCA is calculated using the model and assumptions described in Reference 1. Analysis of the LOCA is presented in Subsection 5.5.

Section 3.3 of Reference 5-17 discusses the control rod drop accident in the power range. Reference 1 in the response to the request on Subsection 5.5.1 describes the rod drop accident at rated power. Both of these analyses show the rod drop in the power range to be much less severe than the rod drop in the startup range.

REFERENCES:

1. “GEGAP-III, A Model for the Prediction of Pellet Cladding Thermal Conductance in BWR Fuel Rods”, NEDC-20181, Revision 1, November 1973.

REQUEST: Subsection 2.4.2.2 – Fuel Cladding Temperature

Provide the basis (e.g., crud and oxide resistance equations) for the results shown in the figures. State the design criteria for the cladding temperature during normal operation.

RESPONSE:

A direct limit on fuel cladding temperature is not used in design evaluations. However, the impact of high cladding temperature, such as decreased yield strength and reduced cladding thickness due to oxidation, is considered in the design evaluation. [[

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The model used to calculate the fuel cladding temperature is documented in Reference 1. This reference has been included into Subsection 2.4.2.2.

REFERENCES:

1. “General Electric Standard Safety Analysis Report”, Proprietary Supplement to Amendment 14, Docket No. STN-50-447, May 1974.

REQUEST: Subsection 2.4.2.3 – Fission Gas Release

State the design criteria for internal gas pressure. Discuss/reference the analysis procedure (models, methods, assumptions) used to calculate the internal gas temperature and pressure. Discuss the consideration of stable isotopes and high burnup dependence. State the limiting rod type (fuel length) considered in the calculation. Compare this to Hatch Reload 2. Discuss the peaking factors and axial profile used in the analysis. Discuss the effect of radial pellet densification on calculated temperatures, gas release, volumes and pressures. State the maximum calculated gap pressure at end of life for normal and transient conditions.

RESPONSE:

No direct criteria are employed for the internal gas pressure. (See response to request on Subsection 2.4.) The internal pressure is used in conjunction with other loads on the fuel rod cladding when calculating cladding stresses and comparing these stresses to the design criteria. Details of this evaluation are described in Subsection 2.5.3.1.2.

The fuel rod internal pressure is calculated using the perfect gas law and the assumptions detailed in Subsection 2.4.2.3. [[

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REFERENCES:

1. *General Electric Standard Safety Analysis Report, Proprietary Supplement to Amendment 14*, Docket No. STN-50-447, May 1974.

REQUEST: Subsection 2.4.2.4 – Fuel and Cladding Expansion

What is meant by the "cladding average temperature" (radial, axial or volumetric)? What is meant by the average fuel temperature? Give the equation and reference for the fuel thermal expansion coefficient vs. temperature.

RESPONSE:

The cladding average temperature is calculated as the sum of inside and outside surface temperature divided by two at an axial elevation. Based on this cladding average temperature, the cladding axial thermal expansions for the separate axial nodes are summed to determine the total fuel rod cladding axial thermal expansion.

The average fuel temperature in this context is the volumetric average temperature of the fuel pellet cross section at an axial elevation. Again, this calculation is repeated for several axial nodes and the fuel thermal expansions are summed.

The above information has been included into Subsection 2.4.2.4.

The expression used for thermal expansion coefficient of the fuel is provided in Subsection 2.3.2.

REQUEST: Subsection 2.4.2.5 – Incipient Center Melting

Relate the significance of incipient center melting to the design evaluation. Explicitly state a design criteria (e.g., no center melting during normal operation). Relate the calculated results to the criteria. Discuss (or reference previous descriptions of) the code used to calculate the fuel temperatures. The reference to fuel material melting temperature should be to Subsections 2.3.2. and 2.3.3.

RESPONSE:

The model used to calculate fuel temperature is described in Reference l. The fuel is designed so that fuel melting is not expected to occur during normal steady state full power operation. The above information has been incorporated into Subsection 2.4.2.5.

REFERENCES:

1. "General Electric Standard Safety Analysis Report:, Proprietary Supplement to Amendment 14", Docket No. STN-50-447, May 1974.

REQUEST: Subsection 2.4.2.6 – Pellet to Cladding Radial Differential Expansion

This section consists of a single sentence and is inadequate. State the purpose or criteria used in this section relative to thermal design (evaluation) considerations. Discuss the differences in the code modeling and analytical methods between the thermal (fuel temperature) evaluations and the mechanical (fuel stress and strain) evaluations. Relate the purpose of this evaluation to fuel rod thermal analyses and the fuel rod mechanical (stress/strain) analysis. Discuss how fuel rod tolerances (e.g., wall thickness) are considered in the design evaluations. Discuss how pellet hour glassing and radial cracking is accounted for in the fuel rod thermal analysis. Discuss the treatment of these phenomena relative to the 1% plastic strain, fatigue and stress rupture limits. Discuss the handling of radial fuel densification effects. Discuss the effect of gap size changes, due to power/exposure effects, on the value of the gap conductance and resulting fuel temperature and pellet expansion.

RESPONSE:

The purpose of the evaluation described in this subsection is to provide inputs to the mechanical evaluation discussed in Subsection 2.5.3.1.2. No direct criteria are applied to the thermal evaluations. Indirectly, the resulting loads, combined with other loads, must not exceed stress criteria when the mechanical evaluations are performed (see also responses to requests on Subsections 2.4, 2.4.2.2, and 2.4.2.3).

A discussion of the models used in fuel rod thermal analyses has been added to Subsection 2.4 in response to request on Subsection 2.4.2. The same response also discusses assumptions made for the evaluations and the referenced model description discusses the details of the model. This same model and assumptions are used for analysis of 1% plastic strain and for inputs to fatigue analysis (see also responses to requests on Subsections 2.4.1.1 and 2.4.2. Since the present fuel rod analysis methods do not account explicitly for the effect of pellet hourglassing, pellet cracking, or creep relaxation of clad stresses, stress rupture is not explicitly addressed for the fuel rod. This approach is supported by experience in that large cladding plastic strains indicative of creep rupture have not been observed. The handling of radial fuel densification effects is described in responses to requests on Subsections 2.4.1.1, 2.4.2, and 2.4.2.1.4.

REQUEST: Subsection 2.4.3 – Plant Reload Parameters

See comments relating to Subsection 2.4.2.

RESPONSE:

See response to Request on Subsection 2.4.2.

REQUEST:

Subsection 2.5.1 – Analytical Criteria for Assurance of Mechanical Integrity

Identify the critical instability loads considered. State the references and/or test data which support them. Reference Topicals and/or tests used to support the adequacy of the 0.060 inch deflection limit. Your recent tests and references that indicate a lower limit should be provided. Provide the basis for the stress limits described in this section and presented in Table 2-6. Reference previous submittals on these specific limits and/or provide a direct comparison with the ASME Boiler and Pressure Vessel Code requirements. Where differences exist between these sources and the limits stated in the Topical, discuss the difference. Specify, if applicable, stress indices used for normal, upset, emergency and faulted conditions. An expansion of Tables 2-5 and 2-6 to the extent of the analogous diagrams appearing in the ASME Boiler Code Sections would be adequate for documenting the off normal reactor conditions. Define and clarify what deflection limits could result in more serious consequences, as stated in paragraph 5 of Subsection 2.5.1. Describe these consequences and reference any supporting data.

Briefly describe and reference the scaling laws related to the dynamic similitude instability loads. Provide brief discussions and references of the comparison between analyses and loading test results. Discuss the "goodness-of-fit" between the results predicted by the codes used in the analysis and measured test data. Describe which components are subject to shakedown phenomena and how shakedown is handled in the stress criteria. Compare your methods of shakedown analysis with those of ASME Boiler and Pressure Vessel Code. Describe how the combined seismic/LOCA analysis is handled on a generic basis. If both are performed on a "plant specific" analysis, clearly specify this. If a bounding analysis, or "lead plant", approach is to be taken, provide and justify each of the conservatisms used in the analysis.

RESPONSE:

There are two instability (buckling) loads considered: cladding creep collapse and seismic. These loads are evaluated in Subsections 2.5.3.1.1 and 2.5.4, respectively. Both of these subsections reference topical reports for details of the analyses. The seismic LOCA evaluation given in Subsection 2.5.4 is considered to be a bounding evaluation (see the response to the request on Subsection 2.5.4).

A discussion of the tests performed to support the deflection limit is given in References 1 through 3. These references have been included in Subsection 2.5.1. The statement on deflection limits was meant to imply that the stated limits precluded serious consequences and has been clarified.

Stress limits were established as the material yield strength for normal and abnormal events and its ultimate strength for emergency and faulted events. The yield strength was used as the limit for normal and abnormal events so that the reactor may restart after experiencing these events. Use of the ultimate strength limit for emergency and faulted events ensures that the material will not fracture during the single application of loads resulting from these events. For additional information, see the response to the Request on Subsection 2.5.1.1.

With respect to shakedown, because unirradiated Zircaloy exhibits cyclic strain hardening and fully reversed load induced stress amplitudes are less than the monotonic unirradiated material yield strength, the strain response behavior of the material due to cyclic effects remains linear.

REFERENCES:

1. "BWR/6 Fuel Design Amendment No. 1," NEDE-20948-1P (Proprietary) and NEDO-20948-1, November 1976.

2. "BWR/4 and BWR/5 Fuel Design Amendment 1," NEDE-20944-1P (Proprietary) and NEDO-20944-1, January 1977.

3. Attachment to Letter MFN 114-77-050, G.G. Sherwood (GE) to Office of Nuclear Reactor Regulation (D.G. Eisenhut), "NRC Questions on Rod Bowing," March 29, l977

REQUEST: Subsection 2.5.1.1 – Stress Limits for Fuel Rod Analysis

See comments on Stress limits in Subsection 2.5.1. The reference to Subsection 2.5.3.1.3 should be to Subsection 2.5.3.1.2. Please correct.

RESPONSE:

GE fuel rods are designed to assure that the stress intensity limits of Table 2-6 are met for normal and abnormal operation. Fuel integrity during accidents is separately addressed in the evaluation of each accident. The use of the maximum shear stress theory for combined stresses and the stress intensity limits of Table 2-6 was adopted by General Electric during the mid 1960's and provides a very conservative design basis. Essentially all of the GE BWR fuel currently in operation was designed in compliance with these criteria. The resultant fuel rod designs have exhibited satisfactory dimensional stability, with no significant dimensional changes having been observed. Documentation of these limits was previously provided in References 1 through 3. The reference to 2.5.3.1.3 was corrected to 2.5.3.1.2 in Amendment 1.

REFERENCES:

1. "BWR/4 and BWR/5 Fuel Design", NEDE-20944 P (Proprietary) and NEDO-20944, October 1976.

2. "General Electric Boiling Water Reactor Generic Reload Application for 8x8 Fuel," NEDE-20360-1P, Revision 4 (Proprietary) and NEDO-20360, Revision 1, Supplement 4, March 1976 and November 1974.

3. "General Electric Standard Safety Analysis Report," NEDO-10741, April 1973.

REQUEST: Subsection 2.5.3 – Fuel Assembly Normalized and Transient Load Evaluations

The heading for this section is misleading relative to what is presented. Provide a concise table which indicates the loads or load combinations assumed to be applied to each of the fuel assembly components analyzed for each event category (i.e., normal operation, anticipated transients and accidents). Discuss the conditions considered, and cross reference these conditions with the functional requirements. Provide an additional table which indicates the limiting design parameters and each of the functional requirements.

RESPONSE:

This subsection evaluates the effect of normal and abnormal loads on the fuel assembly (functional requirements a and b). Loadings on the fuel assembly during handling are documented in Reference 2-14 of the report. Loadings applied to the fuel rods are given in Subsection 2.5.3.1.2. These loads result in the stresses given in Table 2.5.3-1. Stress combinations for each region of cladding and stress category are given in Table 2.5.3-2. This information has been included in Subsection 2.5.3.1.2. Loadings on other components are given in the appropriate subsection.

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REQUEST: Subsection 2.5.3.1.1 – Cladding Creep Collapse

See comments on Subsection 2.4.2.1.3. Consolidate this subsection with Subsection 2.4.2.1.3.

RESPONSE:

Cladding creep collapse is a densification consideration and is, as such included, briefly, in Subsection 2.4.2.1 with other densification considerations. The analysis itself, however, is a mechanical analysis which is properly presented in Subsection 2.5.3.1.1.

REQUEST: Subsection 2.5.3.1.2 – Stress Evaluation

Define "Stress Category" appearing in paragraph five.

RESPONSE:

The Stress categories are given in Table 2-6. This reference has been included in Subsection 2.5.3.1.2.

REQUEST: Subsection 2.5.3.1.4 – Fatigue Evaluation

The reference to Subsection 2.5.3.1.3 should be to Subsection 2.5.3.1.2. Compare the fatigue damage calculated for the 8x8R fuel design to the 8x8 and 7x7 fuel designs. Reference the model used for the analysis. Justify the K f(= 2) factor used for the fatigue cycling evaluation. Provide the source of the value selected. Relate the analyses performed in this section to PCI fuel failures. Relate the PCIOMR's utilized by a plant to any of the assumptions or inputs in the fatigue evaluation. Relate the mechanical design (analysis) evaluation to the PCIOMR's to be recommended for the 8x8R fuel design. Discuss the preconditioning program which will be recommended for the 8x8R fuel design. Compare these recommendations to those for the 7x7 and 8x8 fuel designs.

RESPONSE:

The intersection of a fuel rod tube and an end plug at the weld is a circumferential notch.
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The above information has been included in Subsection 2.5.3.1.4. Reference 2.5.3.1.3 was corrected to 2.5.3.1.2 in Amendment 1.

Preconditioning interim operating management recommendations (PCIOMR's) are recommendations made by General Electric (GE) which have been demonstrated to improve fuel performance during normal operation. Because they are operational recommendations (not requirements), no assumption regarding their application is made in any design application. Thus, PCIOMR's are not related to any mechanical design evaluations.

With regards to pellet cladding interaction (PCI) failures, Reference 3 documents the bases for GE's conclusion that there is no significant safety concern relating to PCI failures. PCI failures are, therefore, considered an operational inconvenience to be avoided to the maximum extent practical, since they can result in plant limitations and increased maintenance problems. Thus, many plants follow PCIOMR's to reduce the risk of PCI failures.

REFERENCES:

1. D.H. Winne and B.M. Wundt, "Application of the Griffith Irwin Theory of Crack Propagation to the Bursting Behavior of Disks, Including Analytical and Experimental Studies", Transactions of the ASME, Vol. 80, November 1958.

2. W.J. O'Donnell and C.M. Purdy, "The Fatigue Strength of Members Containing Cracks", Journal of Engineering for Industry, May 1964.

3. Letter, G.G. Sherwood to Victor Stello, Jr., "Information Concerning Feedwater Nozzles and Pellet Clad Interaction", November 10, 1976.

REQUEST: Subsection 2.5.3.1.6 – Flow Induced Vibrations

Include in the discussion the impact of the new fuel design on the vibration of the instrument tubes and the resulting potential for channel box wear. Provide a general description of the problem and the "fix" taken relative to the fuel assembly lower tieplate flow holes. Reference and discuss operating experience which supports the adequacy of the fix. Cross reference Section 4.0 for analytical modeling and analysis of the fix.

RESPONSE:

Significant channel wear due to in core vibration was experienced in boiling water reactor (BWR's) which incorporated bypass flow holes in the core support plate. A description of the cause of the wear, the development of a plant modification to eliminate significant levels of in core vibration and the testing performed to demonstrate the efficacy of the modifications are given in Reference 1. The modifications consisted of eliminating the bypass flow path in the core support plate and creating an alternate flow path by introducing two holes in the fuel assembly lower tieplates. Data on plants operating with the modifications implemented are given in Reference 2.

All 8x8R fuel for BWR/4 plants incorporate finger springs in the channel to lower tieplate flow path and two alternate flow path holes in the lower tieplate. BWR/2 and 3 plants may incorporate finger springs and alternate flow path holes for 8x8R fuel. The hydraulic characteristics of 8x8R fuel are very similar to 8x8 fuel. They are within the testing and operation experience base in References 1 and 2. Some fuel assemblies may be fabricated with no finger spring or alternate flow path holes. Plant operation with this configuration and fuel similar to 8x8R is documented in Reference 1. Therefore, the introduction of 8x8R does not introduce any new features which could induce significant in core vibrations.

REFERENCES:

1. “Supplemental Information for Plant Modifications to Eliminate Significant In Core Vibrations” (NEDE-21156).

2. Letter, R.E. Engel (GE) to Robert L. Baer (NRC), “Results of Channel Wear Inspection”, October 18, 1977.

REQUEST: Subsection 2.5.4 – Combined LOCA and Seismic Loads

Define, describe and give the magnitude of each of the loads applied to the 8x8R reload fuel assemblies. Discuss the conservatism of the generic load values of the reactors listed in Table 1-1. Provide a reference for the models methods and assumptions used to calculate the loads.

RESPONSE:

As discussed in Subsection 2.5.4, the BWR/6 LOCA and seismic evaluation documented in Reference 2-5 is conservative for operating plants. The magnitude of peak horizontal acceleration used in the BWR/6 analysis was 3.897 g. Maximum peak acceleration at operating plants listed in Table 1-1 is 3.0 g. The loads applied to the fuel assembly and models methods and assumptions are all documented in Reference 2-5.

REQUEST: Subsection 2.7.1 – Fuel Manufacturing

Discuss the general procedures and acceptance criteria used to evaluate fuel assemblies, subassemblies, components, parts and materials which deviate from the applicable manufacturing drawings or specifications. Discuss the consideration of geometrical tolerances within the fuel rod design and safety evaluations (e.g., kW/ft for 1% strain analyses).

RESPONSE:

The general procedures and acceptance criteria used to evaluate fuel assembly, subassembly, component parts and materials which deviate from applicable manufacturing drawings and specifications are defined in Reference 1. Methods defined in this document have previously been accepted by the NRC (Reference 2). This information has been incorporated in Subsection 2.8.1 (previously 2.7.1).

For a discussion of the consideration of geometrical tolerances within the fuel rod design and safety evaluations, see the responses to the requests on Subsections 2.4.1.1 and 2.4.2.

REFERENCES:

1. “Nuclear Energy Divisions BWR Quality Assurance Program Description”, NEDO-11209-03A, pg. 69, November 1976.

2. Letter, C.J. Heltemes (NRC) to J.F. Quirk (GE), “NRC Acceptance of General Electric QA Topical Report”, October 27, 1976.

REQUEST: Subsection 2.7.2 – Enrichment Control Program

Discuss or reference the analytical methods which are used to evaluate the effects of enrichment manufacturing tolerances on each of the power peaking factors. Tabulate these effects for each fuel type.

RESPONSE:

The effect of the enrichment tolerance on local power peaking factors has been investigated for several types of fuel bundles by intercomparison studies between calculated values and experimental measurements documented in Reference 3-4. The standard analytical method to evaluate the effect of enrichment tolerance on the local power peaking factor is the two dimensional lattice physics code (Reference 3-1). It has been found that the standard deviation of the manufactured fuel pellets is about 0.015 wt % U-235. The corresponding effect on the local power peaking factor is given in Reference 5-1 as 0.7%.

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This response has been incorporated into Subsection 3.2.2.

REQUEST: Subsection 2.7.4 – Surveillance Inspection and Testing of Individual Fuel Rods

Discuss the channel change out criteria and inspection procedures.

RESPONSE:

Channel inspection procedures and deformation limits are detailed in Reference 2-1. This reference, which provides all information on fuel channels, is given in Subsection 2.1.

REQUEST: Table 2-1

Include grid to grid spacing.

RESPONSE:

Spacer Pitch (Grid to grid) spacing for BWR/2,3 is 19.55 inches; for BWR/4, this spacing is 20.15 inches. This information has been included in Table 2-1.

REQUEST:

Table No. 2-8

Provide references for each of the formulas presented.

RESPONSE:

Stress	Reference
Membrane	Simple Force Balance
Vibration	1 and 2
Discontinuity Membrane	Simple Force Balance
Initial Ovality	3
Spacer Contact	4
Discontinuity Bending	5
Thermal Mismatch	5
Radial Thermal Gradient	2
Thermal Bow	2
End Plug Angularity	2
Pellet Cladding Interaction	5

These references have been included in Table 2-8.

REFERENCES:

1. E.P. Quinn, “Vibration of Fuel Rods in Parallel Flow”, GEAP-4059, July 1962.

2. R.J. Roark, “Formulas for Stress and Strain”, Fourth Edition, McGraw-Hill Book Company, 1965.

3. S. Timoshenko, “Strength of Materials, Part II”, Third Edition, D. Van Nostrand Company, 1956.

4. R.J. Roark, “Stresses and Deflections in Thin Shells and Curved Plates Due to Concentrated and Variously Distributed Loading,” NACA-TN-806, May 1941.

5. S. Timoshenko, and Woinowsky-Krieger, S., “Theory of Plates and Shell”, Second Edition, McGraw-Hill Book Company, 1959.

REQUEST: Table 2-10

Footnote the procedure used to obtain the design ratios.

RESPONSE:

The method of determining the design ratio is given in Subsection 2.5.3.1.2 in which Table 2-10 is referenced. A separate footnote for Table 2-10 is, therefore, redundant.

REQUEST: Table 2-11

Same as for Table 2-8.

RESPONSE:

Deflection	Reference
End Plug Angularity	1
Tube Induced Vibration	1 and 2
Thermal Bow	1
Tube Bow	Circular Arc
Axial Load Non tie Fuel Rods Tie Rods	 3 3

These references have been included in Table 2-11.

REFERENCES:

1. R.J. Roark, "Formulas for Stress and Strain", Fourth Edition, McGraw-Hill Book Company, 1965.

2. E.P. Quinn, "Vibration of Fuel Rods in Parallel Flow", GEAP-4059, July 1962.

3. S. Timoshenko, and Gere, J.M., "Theory of Elastic Stability", Second Edition, McGraw-Hill Book Company, 1961.

REQUEST: Table 2-13

Identify limiting component and location for this table. Provide the basis for $K_f = 2$ in the table (see Subsection 2.5.1).

RESPONSE:

See the response to the request on Subsection 2.5.3.1.4.

REQUEST: Figure 2-11

Provide the basis for correcting the maximum mean stress.

RESPONSE:

The basis for correcting the maximum mean stress is given in Reference 2-5 as documented in Subsection 2.3.1.

REQUEST: Figure 2-16

Discuss in the text the basis for the step change shown.

RESPONSE:

The temperature vs. time shown in the figure reflects the assumptions that at a location of the fuel rod, the fuel is operating at 13.4 kW ft/plus the calculated power spiking penalty, up to an exposure of 4000 MWd/t. At that point in time, a gap is assumed to form as a result of densification, and the cladding temperature therefore decreases.

This discussion has been added to Subsection 2.5.3.1.1.

REQUEST: Section 3 – Nuclear Evaluation

The nuclear evaluation section, as written, does not mention anywhere in the text, that there are significant differences between the 8x8 and 8x8R fuel designs. Except for one reference on page 3-3, there is nothing which states that 8x8, rather than 7x7, fuel is under discussion. Therefore, in order to reduce the general obscurity of Section 3, add a new subsection, after 3.1, which discusses the significant differences between the 8x8 and 8x8R fuel designs (water rods, natural uranium, rod diameter, and increase in length). The effects on average power density, local peaking, void coefficient, shutdown margin, axial power distribution, net energy production and all other important nuclear effects should be discussed.

RESPONSE:

This section discusses the methods used to perform nuclear evaluations. These methods are generic and not dependent upon fuel type. The generic analytical results presented are for 8x8 and 8x8R fuel. The introduction to this report (Section 1.0) limits the scope of the report to these two fuel types. As discussed in the response to the request on Subsection 2.1, the 8x8R fuel was not introduced because of safety considerations.

REQUEST: Subsection 3.1 – Introduction

For each reload cycle there will be a different set of four bundle arrays, with a Potential for wide differences in exposure among the bundles within an array.

(a) Is a new library of cross sections, lattice reactivities, and relative rod power generated for each cycle?

(b) Are differences in bundle exposure taken into account by performing four bundle GEBLA calculations or are single bundle calculations performed?

(c) Are the libraries of lattice parameters ever employed over a range of exposure such that extrapolations are required? What are the assurances that conservatism is attained?

RESPONSE:

The exposure used in the bundle design process covers the range of expected in core exposure.

The lattice nuclear libraries are generated by the two dimensional finite difference diffusion depletion lattice physics code (Reference 3-1). It is a single lattice infinite medium calculation. The purpose of generating exposure dependent nuclear libraries it to prepare input for the three dimensional BWR simulator code (Reference 3-2). It has been shown (References 3-3 and 3-4) that these lattice nuclear libraries yield good agreements with experimental measurements for different reactors at various cycles and exposures.

In the process of bundle design, the lattices are depleted ("burned") for the exposure range of 0–35 GWd/t. The nuclear libraries of cross sections, isotopic compositions, atom densities, lattice reactivities and relative rod powers are generated as a function of this exposure range. This exposure range is sufficient for most applications. In the few cases where nodal exposure of the bundle may exceed the nuclear library data points, a curve fitted extrapolation is made. These nodes would be low power, low reactivity nodes of the bundle.

Typically, a bundle would stay in core for four cycles. The accumulated bundle exposure would be in the range of 20–30 GWd/t. Accordingly, the lattice nuclear libraries cover the exposure range for all four of these cycles.

REQUEST: Subsection 3.2 – Bundle Nuclear Characteristics

Summarize the experimental data which justify application of the GEBLA lattice physics code to 8x8R fuel with two water rods.

(a) The discussion should include consideration of the data included in the document, Lattice Physics Methods Verification, and any new data which may be available.

(b) Comparisons between experiment and calculation should be provided for lattice reactivity, relative rod powers, and isotope buildup and burnout.

(c) Those instances where four bundle calculations are necessary to provide good agreement between calculations and experiment should be identified.

(d) RMS differences between experiment and calculation are not sufficient for this discussion. The maximum extent to which the calculations underpredict reactivity and rod powers should be included.

The discussions in Section 2, which are referenced here, state only that Gd_2O_3 is uniformly distributed in the UO_2 pellets. Provide more descriptive text on the Gd_2O_3 loading, and in particular, for each fuel type:

(a) Describe any axial variation in the Gd_2O_3 loading.

(b) Describe the pellet loading. Do all pellets contain gadolina or only selected pellets in the stack?

(c) Discuss the effect of granularity, failure to form a true homogeneous solution and inhomogeneities on the nuclear characteristics of the 8x8R fuel.

RESPONSE:

The qualification of the lattice physics code is based on thorough analyses of critical experiments and benchmark calculations using the Monte Carlo method. Listed below are descriptions of major experiments and benchmark calculations for this purpose:

1. **High Conversion Experiment** – This experiment allows for the examination of the lattice physics code reactivity calculation for a wide range of H/U ratios (0.95–4.165). Comparisons of eigenvalues between the design method and experiment are discussed in Reference 3-2.

2. **Gd Critical Experiment –** The assembly contained 8x8 bundles with Gd_2O_3 and B_4C rods of advanced BWR design. The gamma-scan rod fission rate data are compared with lattice physics code calculations in Reference 3-2.

3. **Thermal Critical Assembly** – The critical assembly consisted of 8x8 bundles under voided and unvoided conditions. The lattice physics code evaluated eigenvalue, U-238 Cd ratio, δ^{28} δ^{49} and Dy reactivity are compared to measurements. For detailed comparison, see Reference 3-2.

4. **Jersey Central Gamma Scan Experiment** – The assembly consisted of 16 7x7 array bundles. Experiments were performed with and without control curtains; this allowed testing of the control blade model in the code. In Reference 3-2, rod-to-rod gamma-scan data are compared with calculation.

5. **KRB Gamma Scan Experiment** – The gamma-scan measurement was made for an exposed bundle of the KRB reactor. The bundle was selected such that to minimize uncertainty in void and neutron leakage. Comparisons of rod-to-rod fission rates are discussed in Reference 3-2.

6. **Tsuruga Gamma Scan Experiment** – Rod-to-rod gamma-scan measurements were made for a bundle at different axial elevations, and lattice physics code calculations were performed for each level with varying void and control conditions. Reference 3-2 presents detailed comparisons.

7. **KKM Gamma Scan Experiment** – This experiment is similar to the Tsuruga experiment. Multi-bundle calculations were made to incorporate the radial leakage effect. For detailed comparisons, see Reference 3-2.

8. **Dresden-l Isotopic Measurement** – Isotopic measurement for the Dresden-1 reactor was analyzed to verify the burnup calculation of the lattice physics code. The analyses included burnup calculations for corner rods as well as interior rod. For details, see Reference 3-2.

9. **Yankee-Rowe Isotopic Measurement** – The Yankee-Rowe isotopic data for the first cycle was analyzed using the lattice physics code.

10. **High Temperature Critical Experiment** – Criticality and gamma-scan measurements were performed for 8x8 MO_2 reload fuels. The experiments included two substitution measurements by UO_2 bundle, which has two water rods. A selected number of experiments have been analyzed using the lattice physics code. Tables 3.2-1 and 3.2-2 compare rod-to-rod fission rates for bundles with two water rods. The leakage effect was small in these central test bundles and the agreement between the two data is generally good, including rods near the water rods. This fuel contained gadolinia rods which have axially gadolinia zoning. The fuel measured in these experiments has subsequently operated successfully for more than two years in an overseas plant.

11. **Monte Carlo Studies** – Monte Carlo calculations were performed frequently to qualify the code for specific problems. For example, Reference 3-2 contains comparisons between the design method and Monte Carlo calculations for 7x7 and 8x8 bundles. Additional comparisons are shown in Tables 3.2-3 and 3.2-4 for the 8x8 bundles with water rods. The agreement is shown to be generally good as before; there is no observable difference in rod power accuracy between the regular rod and the rod near the water rod.

For the 8x8 reload fuel bundles, there is no axial variation in the Gd_2O_3 loading (i.e., gadolinia distribution is axially uniform). For the 8x8R reload fuel bundles, there are some rods with 6 inches of natural U ends on top and on the bottom of part of these rods. These natural U ends contain no Gd_2O_3 (see description of the lattice 8DRL07l). Within the enriched fuel length (which is 138 in. long for a 150-in. active fuel length and 133.24-in. long for a 145.24-in. active fuel length), there are several Gd_2O_3 rods.

For a reload bundle, there may be three to seven gadolinia rods. In each gadolinia rod, all the pellets within the middle 138 in. or 133.2-in. section contain Gd_2O_3.

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These manufacturing specifications were established based on neutronic and material considerations. The specifications are controlling for the manufacturing of the 8x8R reload fuel. Therefore, the reload fuel bundle would have up-to-specification pellets. As a result, their neutronic and thermal behavior are acceptable.

Finally, as an auxiliary check of the core composed of numerous as-built fuel pellets, the local power peaking and reactivity have been monitored and compared with calculations by the three-dimensional BWR simulator. The calculations are based on nominal homogenized fuel rods. Intercomparisons between calculations and measurements have shown close agreements (Reference 3-4). These close agreements point to the conclusion that the manufacturing specifications provide an appropriate acceptance criteria.

3.2-1. Percent Differences in Fission Density
High Temperature Critical A

$\frac{\text{LATTICE PHYSICS CODE-EXP}}{\text{EXP}} \times 100$	1.061	-2.073		2.218	-1.820		-1.351	-0.285
		-5.534	2.018	2.056	2.146	2.670	-6.641	-0.986
			0.351	0.512	1.585	0.692	2.202	
					H_2O	0.679	2.540	-2.808
				H_2O	-0.541	0.280	1.026	-3.100
			-0.683			0.610	1.190	
							-3.629	-3.027
	0.096							-1.412

3.2-2. Percent Differences in Fission Density High Temperature Critical B

$$\frac{\text{LATTICE PHYSICS CODE-EXP}}{\text{EXP}} \times 100$$

3.301	1.967		1.367	0.718		0.215	1.451
	-1.027	-0.212	-1.370	-1.073	0.638	-0.687	1.523
		-0.642	-2.577	-2.767	-0.832	0.0	
				H_2O	-2.301	-0.197	1.856
			H_2O	-2.373	-1.700	-.099	-1.098
	1.667				-1.695	0.967	
						-0.687	-0.851
4.119							1.452

3.2-3. Percent Differences in Fission Density
(8x8, 4Gd, 2 H_2O rods)
C Lattice

$\frac{\text{C-LATTICE PHYSICS CODE}}{\text{LATTICE PHYSICS CODE}} \times 100$

-3.33	0.54	-0.91	-2.33	-3.23	0.74	-1.62	-2.74
	2.98	0.65	2.22	0.84	-0.27	1.84	1.53
		-5.06 Gd	5.07	0.22	-4.45 Gd	-2.24	-1.38
			2.43	H_2O	4.50	0.93	-2.79
				0.82	-0.81	2.53	-4.14
					-5.63 Gd	0.45	-3.05
						1.95	3.25
							3.33

$$\% \text{ Re}\,activity\ Difference\left(\frac{Lattice\ Physics\ Code - MC}{MC}\right) = 0.2$$

3.2-4. Percent Differences in Fission Density
(8x8, 7Gd, 2 H_2O rods)
D Lattice

$\frac{\text{C-LATTICE PHYSICS CODE}}{\text{LATTICE PHYSICS CODE}} \times 100$

-0.43	1.38	-3.08	-2.81	3.10	3.63	2.78	1.84
	-5.38 Gd	-3.25	1.20	2.23	2.03	1.17	1.16
		-4.29 Gd	-2.74	-0.83	-0.35	-1.37 Gd	0.08
			-1.35	H_2O	0.31	-1.25	-1.02
				-3.43 Gd	-0.98	1.44	-2.12
					-4.87	1.66 Gd	-2.74
						-1.90	-2.08
							-4.24

$$\%\ \text{Re}\,activity\ Difference\left(\frac{Lattice\ Physics\ Code - MC}{MC}\right) = 1.1$$

REFERENCES:

1. D.M. Rooney, "Ceramographic technique for revealing inhomogeneity in UO_2 specimens with small additions of selected oxides", NEDO-12024, General Electric Co., Pleasanton, CA (1969).

REQUEST: Subsection 3.2.1 – Reactivity

Provide explanatory text on the dependence of bundle reactivity and reactivity swing to average enrichment, gadolinia loading, void fraction, and exposure. The one sentence reference to Figures 3.1-1 through 3.1-24 is particularly inadequate and should be substantially expanded:

Explain the void history and why the curves cross.

Do the figures include equilibrium xenon? If so, at what flux were the curves calculated?

Some of the even numbered figures in the 3-1 series (e.g., Figures 3.1- 6 and 3.1-10) are not labeled with an exposure. Are all of these k_∞ vs. in channel void fraction plots done at zero exposure? Provide clarification.

RESPONSE:

The following information has been added to Subsection 3.2.1:

3.2.1.1 Factors Affecting Lattice Reactivity

The lattice reactivity is a function of lattice average enrichment, gadolinia loading, void fraction, hydrogen-to-U ratio and exposure. To delineate all these functional dependences, it is necessary to note the following.

For a given lattice, it is observed that:

(a) At zero exposure, the reactivity is highest at 0.0 void fraction (VF), followed by 0.4 VF and has least reactivity at 0.7 VF. This is because enriched U-235 fuel lattice is of higher reactivity for a softer neutron spectrum. The softer neutron spectrum is most abundant in the highly thermalized medium of 0.0 VF.

(b) The term "void history" refers to the fact that the k_∞ curves were obtained for the respective lattices based on a nuclear depletion (exposure) history in the water density of 0.0, 0.4 or 0.7 VF. In other words, assuming a 0.0, 0.4 or 0.7 VF water density for the whole exposure range, the illustrated k_∞ curves were obtained.

(c) [[

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(d) The in situ production of the Pu nuclides increases with increasing void fraction. For example, 0.7 VF is better for Pu production than 0.4 and 0.0 VF. This is because the first has a harder (less thermalized) neutron flux spectrum than the last two. As the lattice attains higher exposures, the U-235 is increasingly depleted and the worth of the bred fissile Pu nuclides becomes more significant. At about 15 GWd/t and thereafter, the fractional fission of the bred Pu nuclides (Pu-239 and Pu-241) in the 0.7 VF case is a few percent greater than those of the 0.0 and 0.4 VF cases. The net result is that the 0.7 VF case is of higher reactivity than the low void fraction cases. This is the reason that the 0.7 k_∞ curves cross over and become of higher reactivity at about GWd/t than the lower void cases.

An exception to these k_∞ behavior and crossover phenomena at high exposure is the natural U lattice. Here, it is observed that, for all exposures, the higher the void fraction, the greater the reactivity. This is due to the fact that the natural U lattice, having no enriched U-235 and gadolinia, is primarily a U-238 system. It is most favored for Pu fissile nuclides generation under a hard neutron flux spectrum; namely, that of the 0.7 VF.

It is also observed that the maximum reactivity for the natural U lattice occurs earlier than the other lattices with enriched U and gadolinia. It peaks at about 2.0 GWd/t for the former and 6.0 GWd/t for the latter. This is because the enriched lattice is designed such that the gadolinia occurs around the end of the first cycle, usually about 6.0 GWd/t.

The primary driving force in fission for the natural U lattices is from the bred fissile Pu-nuclides. Since the Pu production is neutronically favored for a hard spectrum, the high void case always displays a higher reactivity. This reactivity difference increases at the higher exposure range with the increasing fractional fission by Pu-nuclides.

(e) [[

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REQUEST: Subsection 3.2.2 – Local Peaking Factors

This paragraph is very brief and extremely difficult to relate to actual technical specifications. Provide an expanded text which precisely defines all peaking factors (gross, local, radial, axial, total) and how the elevation is chosen, etc. Explain how these peaking factors are related to LHGR, APLHG, TPF, and MCPR, and how these quantities are limited by technical specifications. Describe how the peak local power migrates vertically and from pin to pin during exposure and how this causes “breaks” in Figures 3.2-1 through 3.2-12. Explain why infinite lattice local peaking factors are appropriate.

The maximum credible exposure which could be encountered for any assembly at any axial location within the assembly should be indicated. The possibility that peaking factors have not been calculated over a sufficiently wide range of exposure should be discussed.

The dependence of the maximum local peaking factor, P_L, on the local average void fraction should be included by either expanding Figures 3.2-1 through 3.2-12 to include void fractions ranging from zero to the maximum expected, or by justifying that P_L is not sensitive to void fraction.

Are the curves of local peaking factor in Figures 3.2-1 through 3.2-12 a worst-case (rather than a best estimate) curve as the word "maximum" in the legend implies? If not, what is the calculational uncertainty? Identify which pin is limiting on each portion of each curve.

Provide (explicitly or by reference) local power distribution for each fuel type at zero MWd/t and at a high exposure (e.g., 30,000 MWd/t).

RESPONSE:

The local peaking, gross radial, axial and total peaking factors are design parameters related to reload core analysis. Their respective definitions are shown in Table 3.2.2-1. These peaking factors determine, directly or indirectly, the thermal performance parameters such as maximum linear heat generation rate (MLHGR), maximum average planar linear heat generation rate (MAPLHGR), and minimum critical power ratio (MCPR). The relations between the various peaking factors and the core thermal performance parameters are detailed in Table 3.2.2-2. The peaking factor, by itself, does not constitute a limiting condition. The thermal performance parameters such as MLHGR, MAPLHGR, and MCPR do limit unacceptable combinations of these peaking factors.

For a given lattice at a given void fraction, the maximum local peaking factor will occur at different fuel rods as the exposure increases. This is due to the different depletion and generation rate of the various fissile nuclides in each fuel rod. Previous Figures 3.2-1 through 3.2-12 (now Figure 3-4) show the maximum local peaking factor for the respective lattices at different exposures. These curves are calculated at the typical void fraction of 0.4. [[

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It has been observed that local peaking factors from infinite lattice calculations exhibit close agreement with experimental results (Reference 3-2); hence, their use is appropriate.

The axial power shapes and axial peaking factors are dependent on the fuel bundle types and exposures, the in-core locations, the control rod pattern, the specific reload cycle and the plant. These axial power shapes and axial peaking factors are calculated by the three-dimensional BWR simulators which takes all of the effects into account. Therefore, while it is possible to provide curves showing the variation of the local peaking factor as a function of exposure, it is not possible to do so for the axial peaking factor because it depends on plant unique features and operating characteristics.

The local peaking factors are generated for the exposure range of 0–35 GWd/t. This covers the expected range of exposures for any fuel bundle at any axial location. If a particular location should exceed 35 GWd/t, the power at that location will be sufficiently low that there is no danger of exceeding any performance limits because of its low reactivity at high exposures.

The above information has replaced the documentation previously in Subsection 3.2.2.

The local peaking factor does vary with void fraction, and this dependence is taken into account in the calculations used to assign local peaking factors to each axial segment of the fuel. Figure 3-4 shows the volume at 0.40 voids, as this is the typical average bundle void fraction.

The curves of local peaking factor in Figures 3.2-1 through 3.2-12 are the nominally calculated values. The word “maximum” is used to denote the fact that the maximum pin power at each exposure is used to construct the curves. As stated earlier, these nominally calculated values are appropriate, as they agree well with experimental results.

Figures showing the local pin power distributions as a function of void fraction and exposure for typical fuel lattices are provided in Reference 1.

3.2.2-1. Definitions of Some Peaking Factors

Local Peaking Factor – Local peaking factor is the ratio of the heat flux in the highest powered rod at a given plane to that of the average rod in the plane.
Radial Peaking Factor (RPF) – The ratio of the fuel assembly power in a particular assembly to the power of the average fuel assembly.
Axial Peaking Factor (APF) – The ratio of the heat flux at the axial plane of interest to the heat flux averaged over the active length of the fuel (assembly or rod) of interest.
Gross Peaking Factor (GPF) – The product of the radial peaking factor for a fuel assembly and the maximum axial peaking factor for the same fuel assembly.
Total Peaking Factor (TPF) – The total peaking factor is that peaking factor which, when multiplied by the average linear heat generation rate of a specific bundle type, yields the technical specification limit on MLHGR for that bundle type. This definition of the total peaking factor is presented as an equation in the response to the request on Subsection 5.2.1.5. This response also discusses the relationship of the total peaking factor to the APRM rod block and scram setpoints.

3.2.2-2. Relationship of Technical Specifications and Peaking Factors

Maximum Linear Heat Generation Rate (MLHGR) The MLHGR is the maximum linear heat generation rate (expressed in kW/ft) in any fuel rod allowed by the technical specifications (tech specs) for a given fuel type. The MLHGR is attained when the product of the local, radial and axial peaking factors in an axial segment of a fuel bundle equals the total peaking factor for that fuel type.
Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) The MAPLHGR is the maximum average linear heat generation rate (expressed in kW/ft) in any plane of a fuel bundle allowed by the tech specs for that fuel type. This parameter is obtained by averaging the LHGR over each fuel rod in the plane and its limiting value is selected such that: (a) the peak clad temperature during the design basis loss of coolant accident will not exceed 2200°F, and (b) the LHGR will not exceed the MLHGR in the plane of interest. The MAPLHGR is attained when the product of the gross peaking factor and the average rod peaking factor (1.0) equals the tech spec value.
Minimum Critical Power Ratio (MCPR) The MCPR is the minimum CPR allowed by the tech specs for a given bundle type. The CPR is a function of several parameters, the most important of which are bundle power, bundle flow, and bundle R factor. The R factor is dependent upon the local power distribution, but is only indirectly related to the local peaking factor. The limiting value of CPR is selected for each bundle type such that during the most limiting event of moderate frequency, the CPR in that bundle will not be less than the safety limit CPR. The MCPR is attained when the bundle power, R factor, flow, and other relevant parameters combine to yield the tech spec value. Therefore, MCPR is not directly related to any of the peaking factors described in Table 3.2.2 1.

REFERENCES:

1. "BWR/4 and BWR/5 Fuel Design," NEDE-20944-P (proprietary) and NEDO-20944, October 1976.

REQUEST: Subsection 3.2.3 – Doppler Reactivity

Provide estimates of the uncertainties in the nominal Doppler coefficients related to microscopic cross sections used and resonance capture calculation methods.

Include a discussion of the differences between the nominal or point kinetic doppler coefficient and the Doppler coefficient as used for a 3D transient or accident calculation. In particular, discuss the magnitude of and the basis for any "conservatism" factors which have been applied.

Explain how the Doppler coefficient becomes more negative as exposure and Pu-240 inventory increase and as void fraction increases. Explain further, preferably by including graphical illustrations, how the Doppler coefficient range stabilizes as the equilibrium cycle is approached.

Are the curves in Figures 3.3-1 through 3.3-24 best estimate or worst case values? That "design conservatism factor" is used?

Discuss the upturn in the void coefficient at 4000°F at 0.4 voids for 8DRO.711-G080M fuel at 200 MWd/t (Figure 3.3-9).

RESPONSE:

The subject of the Doppler reactivity coefficient model is described in detail in Reference 3-5 of the report and subsequent responses to NRC questions documented in Reference 3-7. Although no detailed uncertainty evaluation exists, Subsection 2.3.3 of Reference 3-5 discussed the comparison of the Doppler model to experimental data. These comparisons show good agreement between the calculated and experimental results.

Subsections 2.3.2, 2.3.5, and 4.l.2.2 of Reference 3 5 discuss the application of the point kinetic Doppler coefficient to 3D plant transient analyses. The response to Request 17 of Reference 3-7 discusses the application of the Doppler feedback to the continuous rod withdrawal error events and the control rod drop accident.

Subsection 2.3.1 of Reference 3-5 describes the methods used in calculating the Doppler reactivity including Pu-240. Subsequent NRC questions (e.g., Request 1, 2, 8c, etc. of Reference 3-5) and their response have previously addressed the subject of Pu-240 and fuel burnup effects on Doppler coefficients. The sensitivity of the Doppler coefficient to various lattice designs, voids exposure, and control are discussed in Subsection 2.3.4 of Reference 3-5.

Figure 2.3-3 of Reference 3-5 and subsequent figures demonstrate that the Doppler coefficient is not a parameter which varies significantly other than for moderator states and exposure. Therefore, the Doppler coefficient is stabilized and its range of variation will be essentially the same for all reload cores.

The above information has been incorporated into Subsection 3.2.3. The curves in Figures 3-5 and 3-6 are best estimate or nominal. Other than the “design conservatism factor” discussed in Reference 3-5, the nominal values for Doppler coefficients are used in design calculations.

The upturn in the Doppler coefficient in previous Figure 3.3-9 was due to a graphical error. This figure as been replaced.

REQUEST: Subsection 3.2.4 – Void Effect

Do Figures 3.4-1 through 3.4-12 refer to a complete bundle depleted at an average void fraction of 40% or do they refer to one axial height of a bundle, with a void fraction of exactly 40%? Please clarify.

Are Figures 3.4-1 through 3.4-12 best estimate or worst case? What “design conservatism factors” are used?

RESPONSE:

Figure 3-7 refers to the instantaneous void Δk_∞ of the various central lattices for the exposure range shown. Specifically, for a given lattice the change of infinite multiplication factor, Δk_∞, due to the change of the void content (from 40% to 0% and 70%) is plotted as a function of a hot operating BWR fuel bundle. The Δk_∞ shown was computed at hot operating conditions also. It is shown that, in all cases for enriched lattices, a loss of reactivity (i.e., a negative Δk_∞) occurs from increasing the void content from 40% to 70%. On the other hand, a gain of reactivity (a positive Δk_∞) occurs from decreasing the void content from 40% to 0%. This observation confirms the fact that BWR has a negative void coefficient.

The determination of the void coefficient and the design conservatism factors used in this application are discussed in References 3-5 and 3-7.

REQUEST: Subsection 3.3 – Reference Loading Pattern

This section does not discuss control rod patterns and worths. Provide a section which discusses how rod worths are restricted during startup, why the worths are reduced during power operation, and why no restriction on rod worths and pattern symmetry are needed during power operation.

A slight reversal of differential rod worth (i.e., a slight increase in core power following a notch insertion) when a control blade is almost completely withdrawn has been observed in BWRs. As part of the rod worth discussion requested above, provide a paragraph which discusses this phenomenon and the effect of the new fuel design on the power distribution and differential rod worth in the subcooled region at the bottom of the core.

Some parameters become limiting at the end of cycle. How is end of cycle defined (e.g., achievement of a particular exposure, achievement of ARO, coastdown by some percentage of rated power, etc.) and how are EOC limited parameters (e.g., delayed neutron fraction) assured to remain within bounds?

RESPONSE:

Rod worths during startups are restricted to meet the requirements imposed by the control rod drop accident. This is discussed in detail in Subsection 5.5.1 and in References 5-16 through 5-19. Rod worths as a function of moderator density and power level are discussed in the FSARs. Pattern symmetry during power operation is desirable but not necessary. Symmetry greatly simplifies the core limits calculations; however, the core instrumentation is adequate for limits monitoring during asymmetric operation.

The new fuel design is not expected to have a significant effect on reverse power response. The effect of any design change on differential rod worth in any region of the core is inherently accounted for in the design process.

End of cycle (EOC) is defined as the exposure at which the reactor no longer has the reactivity to sustain rated power at the all rods out (ARO) condition. In the case where the cycles include coastdown to a specific power level, EOC is defined as the exposure at which the reactor no longer has reactivity to sustain that specific power at the ARP condition. Operation beyond this exposure requires either: (a) verification that the validity of the operating limits will be maintained, or (b) calculation of a new set of operating limits.

REQUEST: Subsection 3.3.2.1.1 – Core Effective Multiplication and Control System Worth

What is the calculational uncertainty in the three eigenvalues given in the reload supplement table? How much of this uncertainty is due to possible deviation in the exposure at the end of the previous cycle? Do these values represent xenon free (and, for R, equilibrium Sm) conditions?

Describe how the core reactivity peaks as the burnable poison depletes and how the value of R quantifies this effect. Explain why the worth of the strongest control rod at BOC remains appropriate at the point of maximum reactivity.

RESPONSE:

These three eigenvalues (effective multiplication of the core, uncontrolled, fully controlled and with the strongest rod out) are calculated at an exposure corresponding to the minimum expected exposure of the previous cycle. The core is assumed to be in a xenon free condition. This procedure insures that the calculated values are conservative. The value of R includes equilibrium Sm. Further discussion of the uncertainty of these calculations is in Reference 3-4 and Reference l.

As exposure accumulates and burnable poison depletes in the least burned fuel bundles, an increase in core reactivity may occur. The nature of this increase depends on specifics of fuel loading and control state. For example, if one control cell is loaded with two fresh bundles, then the core reactivity calculated with that single control rod removed will probably increase as exposure accumulates. However, the core reactivity calculated with a different rod out may decrease with increased exposure.

Cold k_{eff} is calculated with the strongest control rod out at various exposures through the cycle. At each exposure point a search is made for the single strongest rod, the location of which may change with exposure. The value R is the difference of the strongest rod out k_{eff} at BOC and the maximum calculated strongest rod out k_{eff} at any exposure point. The strongest rod out k_{eff} at any exposure point is equal to or less than the fully controlled k_{eff} at BOC plus the strong rod worth at BOC plus R.

REFERENCES:

"BWR 4/5/6 Standard Safety Analysis Report Rev. 2," Chapter 4, 1977.

REQUEST: Subsection 3.3.2.1.2 – Reactor Shutdown Margin

List any plant specific effects (e.g., operation with inverted absorber tubes) which may effect the reload evaluation of the shutdown margin.

This section addresses shutdown margin, but scram reactivity is not addressed. Explain and quantitatively justify why the scram curve (negative reactivity vs time) for all possible reloads will be bounded by the negative reactivity insertion rate assumed in the accident and transient analyses. Some reactors may be operated in "coastdown." The effect of operation beyond the ARO exposure point should be discussed.

RESPONSE:

Items which may affect the evaluation of reactor shutdown margin for a given cycle are:

1. an early termination or an extension of the previous cycle, and

2. the ability to insert a control rod to the fully inserted position.

The above information has been incorporated into Subsection 3.3.2.1.2. The topics of negative reactivity insertion rate during transient and accident conditions are addressed in Subsections 5.2 and 5.5, respectively, of the report. Also, as documented in Appendix A, the scram curve used for transient analyses is given for each reload. The scram curve for group notch plants is calculated and compared to the bounding curves for the control rod drop accident. Bank position withdrawal sequence plants are covered by Subsection 5.5.1.3 of Amendment 1.

The deviation of control rod worth due to inverted absorber tubes was examined by generic analyses. Acceptance criteria were established to limit the number of inverted tubes and the control worth deviation. These acceptance criteria have proven to be satisfactory in assuring that there always exist sufficient control rod worth.

Accordingly, the acceptance criteria of the control blade have preempted the inverted tube problem. As a result, a plant-by-plant inverted tube analysis is not necessary and therefore not part of the standard reload analyses.

Plants with shutdown margin adjustment for inverted B_4C tubes are:

Dresden 2

Dresden 3

Quad Cities l

Quad Cities 2

Millstone

Monticello

REQUEST: Subsection 3.3.2.1.3 – Standby Liquid Control System

According to the text, the shutdown margin for the SLCS (to be given in the reload supplement) is calculated at the "minimum control rod position," which is not necessarily all rods out. Explain how the exposure corresponding to the maximum net defect is chosen. If cases exist where this is EOC but not ARO, explain how exposure and/or control fraction will be limited to assure SLCS shutdown capability. Justify the use of "minimum control rod position" rather than the all rods out configuration.

What is the uncertainty in the ppm and shutdown margin figures on page A 3 of the reload supplement? Will the values given in an actual reload be best estimate or will conservatism factors be included?

RESPONSE:

The SLCS must have "the capability of bringing . . . from the minimum control rod position." However, the SLCS shutdown margin is calculated for a fully uncontrolled condition at the most reactive point in the cycle which can never be achieved by an operating reactor. This calculated value conservatively verifies that the system requirement is met.

The plant specific ppm is given in the Plant Supplemental Submittal and is the value in that plant's technical specification basis. The shutdown margin is calculated consistent with the technical specification ppm value.

REQUEST: Subsection 3.3.2.1.4 – Reactivity of Fuel in Storage

Verify that all operating BWR's are equipped with spent fuel storage racks with the two rack spacings given. If storage racks other than types A and B exist explain why this analysis is bounding.

Provide a third column in the table of page 3-7 which gives the exposure corresponding to the maximum k.

RESPONSE:

The two rack spacings apply for two storage racks designed by the General Electric Company. For other storage rack designs, specific information must be provided by the utilities.

The table in Subsection 3.3.2.1.4 has been revised to appear as follows:

Table

Bundle Type	Maximum k_∞	Exposure (GWd/t)
8D250	1.236	5.0
8D262	1.241	5.0
8D274L	1.238	5.0
8D274H	1.216	7.0
8D219L	1.159	0.0
8D219H	1.119	8.0
8DRL303	1.210	9.0
8DRL301	1.228	7.0
8DRL282L	1.239	5.0
8DRL282H	1.218	7.0
8DNL282L	1.226	8.0
8DNL282H	1.212	8.0
8DRL254	1.220	8.0

REQUEST: Subsection 3.3.2.1.5 – Reactivity Coefficients

Although the Doppler and void coefficients dominate reactor behavior during power operation, the moderator temperature coefficient is significant during reactor startup. Discuss the importance of the moderator temperature coefficient during plant heatup. Relate the moderator temperature coefficient(s) calculated in this section to the moderator temperature effect discussed in Section 3.5.3.

Equation 3.3-1 appears to have an incorrect "•" between k_∞^{uc} and (E_i , V). Please correct.

Provide estimates of the uncertainties in the nominal void coefficient related to microscopic cross sections used, resonance capture calculation methods, etc.

Discuss the differences between the nominal or point kinetics void coefficient and the void coefficient as used for a 3D transient or accident calculation. In particular, discuss the magnitude of and the basis for any “conservatism” factors which have been applied.

RESPONSE:

The moderator temperature coefficient is not a significant reactivity coefficient because its effect is limited to primarily the reactor startup range. Once the reactor reaches the power producing range, boiling begins and the moderator temperature remains essentially constant. As with the void coefficient, the moderator temperature coefficient is associated with a change in the moderating power of the water. The temperature coefficient is negative during power operation.

The range of values of moderator temperature coefficients encountered in current BWR lattices does not include any that are significant from the safety point of view. Typically, the temperature coefficient may range from $+4 \times 10^{-5}$ $\Delta k/k°F$ to -14×10^{-5} $\Delta k°F$, depending on base temperature and core exposure. The small magnitude of this coefficient (relative to that associated with steam voids) and combined with the long time constant associated with transfer of heat from the fuel to the coolant, modes the reactivity contribution of moderator temperature of small importance.

Because of its relative insignificance, current core design criteria do not impose limits on the value of the temperature coefficient. A measure of design control over the temperature coefficient is exercised by applying a design limit to the void coefficient. This constraint implies control over the water to fuel ratio of the lattice; this, in turn, controls the temperature coefficient. Thus, imposing a quantitative limit on the void coefficient effectively limits the temperature coefficient.

The above information has been incorporated into a new Subsection 3.3.2.1.8.

The void coefficient model mused for plant transient analyses is discussed in detail in References 3-5 and 3-7. No detailed evaluation of uncertainties has been made; however, results of extensive sensitivity studies (Subsection 3.4), and comparisons of the point model void coefficients to more detailed three-dimensional void coefficients and one dimensional transient model are presented in Subsections 3.4.7, 3.4.8 and 3.5.2. The allowance for nuclear input uncertainties is discussed in Subsection 4.1.2 of the subject report.

Equation 3.3-1 was corrected in Amendment l.

REQUEST: Subsection 3.3.2.1.7 – Doppler Coefficient

Provide explicit definitions for “E” and “UH” in Equation 3.3 4.

RESPONSE:

Nomenclature for “E” and “UH” were provided in Amendment 1.

REQUEST: Subsection 3.4.2 – Acceptable Deviation From Reference

Include the results of the sensitivity studies performed to indicate how the validity of the licensing analysis is affected by differences between the reference design core and the actual design core.

RESPONSE:

The sensitivity studies identified in Subsection 3.4.2 are continuing. To date, these studies have identified items to be reviewed (Subsection 3.4.2) and the basis of any re examination (Subsection 3.4.3).

REQUEST: Subsection 3.4.2.5 (and 6) – Location of Reload (and Exposed) Bundles

What is meant by “regions of least importance”?

RESPONSE:

When a small change from the reference design core is necessary, the location of one or more fresh (or exposed) bundles may need to be modified also. Under such circumstances, the primary concern is to ensure that the core still meets reference licensing requirements such as cold shutdown margin and transient behavior. It should also approach a core incremental exposure close to the original reference exposure. To this end, the change of these bundle locations is made in “regions of least importance,” which is usually in the periphery. This refers to the fact that these regions (bundle locations) do not contribute much in core exposure capabilities and are of low power. Therefore, the modifications of these loading regions do not result in power mismatch problems, shutdown degradations and/or adverse transient behavior.

REQUEST: Subsection 3.4.2.8 – Symmetry

Provide more text which explains how the process computer programs use assumptions of symmetry and how deviations from symmetry in fuel loading and rod patterns are allowed for. What portions of the safety analysis assume core quadrant symmetry?

RESPONSE:

Calculation of the safety limit MCPR by the GETAB analysis assumes core quadrant fuel bundle type symmetry. No such assumption is made in the other areas of safety analysis. It should be noted that the safety limit MCPR was derived for the most conservative power distribution and should also apply for the case of asymmetric reactor power. This is discussed further in Reference 1.

When the reactor core is being operated with a mirror or rotationally symmetric control rod pattern, the neutron flux at similarly symmetric narrow narrow gap locations in the four quadrants is considered to be equal. This fact is used to reflect the readings of the real strings into their symmetric counterpart locations where no real strings exist. This reflection is done prior to the commencement of the power distribution calculations.

In the few incidences where fuel bundles near the edge are quadrant loaded asymmetrically, the error induced by reflecting real readings is partially negated by the fuel type dependent correlations. Any remaining error is considered to be second order. Further, because such bundles are in low power regions, the chance that one of them is a limiting bundle is minimal.

In the rare case of the reactor being operated with an asymmetric control rod pattern, the reflection of real string readings is not utilized. In this instance, readings at locations without strings are inferred by interpolation of the real string values in the immediate vicinity.

The above information replaces the information previously given in Subsection 3.4.2.8. It should be noted that operating GETAB calculations do not require the process computer.

REFERENCES:

1. "Process Computer Performance Evaluation Accuracy Amendment l," NEDO-20340-l, December 1974.

REQUEST: Subsection 3.4.3

Discuss the reasons why the six listed parameters were chosen and why others (e.g., delayed neutron fraction, Doppler coefficient, SLCS effectiveness) can be excluded.

RESPONSE:

The listed parameters were chosen by one of the following two criteria:

1. It is a parameter whose magnitude or behavior is explicitly reported in the reload licensing analysis.

   Examples: Cold Shutdown Margin, Peak Fuel Enthalpy in Rod Drop Accident, Change in CPR Due to a Misloaded Assembly, and Rod Block Monitor Response.

2. It is a parameter important to the quantification of an operating limit.

   Examples: Scram Reactivity Insertion and Dynamic Void Coefficient Affect MCPR limit.

   Other parameters were excluded for the following reasons:

   (a) SLCS effectiveness with no significant change in cold shutdown margin, SLCS effectiveness will not change.

   (b) Doppler Coefficient and Delayed Neutron Fraction – These are slowly varying functions of exposure which do not change significantly over the expected range of exposure deviations.

The above information has been included in Subsection 3.4.3.

REQUEST: Subsection 3.5 – Startup Physics Test Program

The startup physics test program as outlined in this section lacks the necessary depth of discussion. The discussion will require a significant amount of additional detail in order to make clear the acceptability of the methods, procedures and acceptance criteria used for the various tests in the program. Specifically, the following comments are submitted on each subsection:

RESPONSE:

The startup physics program was originally included in the report in response to numerous NRC questions on recent reload submittals. This subsection was intended to represent the basis for a suitable restart test program. However, this program is the responsibility of the plant operator. Therefore, a detailed program is considered outside the scope of this report, and this subsection has been deleted.

REQUEST: Subsection 3.5.1 – Core Verification

Discuss or reference the procedures used to check the location and orientation of each fuel assembly during and after core loading. What changes are made to the plant process computer (settings, constants, coefficients, etc.) whenever the core must be reloaded differently from the reference core (e.g., due to excessive fuel failures) appearing in the reload submittal. Discuss how such differences may impact upon any of the other startup physics tests. Discuss the criteria used to select substitute fuel assemblies for failed or damaged assemblies.

RESPONSE:

See the response to the request on Subsection 3.5.

REQUEST: Subsection 3.5.2 – Cold Shutdown Margin

Discuss in detail the procedures used for the verification of cold shutdown margin with the highest worth control rod withdrawn. Provide the details of the measurement as well as the methods used to verify the Technical Specification requirements for shutdown margin. Discuss the acceptance criteria and the procedures followed if the acceptance criteria are exceeded.

RESPONSE:

See the response to the request on Subsection 3.5.

REQUEST: Subsection 3.5.3 – Moderator Temperature Coefficient

Describe in detail the procedures used for the moderator temperature coefficient measurement. This description should include the acceptance criteria and the procedures to be followed if the acceptance criteria is not met.

RESPONSE:

See the response to the request on Subsection 3.5.

REQUEST: Subsection 3.5.4 – SRM Performance

Describe in detail the procedures and methods used for the SRM Performance Test. This description should include the acceptance criteria and the procedures to be followed if the acceptance criteria is not met.

RESPONSE:

See the response to the request on Subsection 3.5.

REQUEST: Subsection 3.5.5 – Core Power Distribution

A power distribution comparison at a given control rod pattern and power level (>50% rated power with equilibrium xenon) must be done for each cycle. Discuss the details of this measurement and the methods used to compare the results with the predictions. The acceptance criteria and the procedures to be followed if the acceptance criteria are exceeded should also be discussed.

RESPONSE:

See the response to the request on Subsection 3.5.

REQUEST: Subsection 3.5.8 – Control Rod Drive Testing

Describe in detail the procedures used for the drive tests and the scram time tests. This description should include the acceptance criteria for these tests and the procedures to be followed if the acceptance criteria is not met.

RESPONSE:

See the response to the request on Subsection 3.5.

REQUEST: Additional Tests Required

Core Power Symmetry Test

A core power symmetry test above 75% rated power must be done with the other BOC physics tests. Discuss the details of this test and the methods used to analyze the results. The acceptance criteria and the procedures to be followed if the acceptance criteria are exceeded should be discussed.

Critical Eigenvalue Comparison

A critical eigenvalue comparison (nonvoided moderator condition) for a fixed control rod pattern must be done for each cycle. Discuss the details of the measurement and the methods used to compare the results with the prediction. The acceptance criteria and the procedures followed if the acceptance criteria are exceeded should also be discussed.

TIP Reproducibility Test

A TIP reproducibility test must be done at BOC at a power >75% rated power. Discuss the details of this test and how the results will be used to validate the reproducibility assumed in analysis.

RESPONSE:

See the response to the request on Subsection 3.5.

REQUEST: Subsection 3.X – Process Computer

Provide a section which addresses the changes made to the plant's process computer prior to each cycle of operation. The section should describe: (l) what elements of the process computer change from cycle to cycle; coefficients, constants, correlations, etc. and why they change; (2) what codes and methods are used to establish the new values; and (3) what quality assurance procedures (testing) are used at the site to verify that the process computer has been correctly reprogrammed. Discuss the impact of the new 8x8R fuel design relative to (1). Reference the topical report which discusses the process computer.

RESPONSE:

Before discussing the process computer update procedure, it should be emphasized that the process computer hardware is not a safety system and that the process computer software is not a part of the license basis. The process computer system is considered a tool which the reactor operator may use to more efficiently perform the limits monitoring function. The reload licensing process is valid irrespective of the availability of the process computer. Past experience has demonstrated that the BWR can operate for extended periods without the process computer.

During each refueling, the process computer is "updated" to reflect changes made to the core. The bundle identification (ID) array is rearranged in accordance with the shuffled core; ID's of discharged bundles are replaced with ID's of the reload bundles. Simultaneously, exposure and void history arrays are rearranged and the indices which assign bundle types to core locations are shuffled. If a new bundle is introduced, nuclear coefficients (power to TIP coefficients, local peaking factors, MAPLHGR limits, etc.) may be required. If a new mechanical design is introduced, such as 8x8R, new thermal hydraulic coefficients will be required. The "update" is accomplished by generating a data bank in San Jose which is transmitted to the utility. The generation of all data and of the data bank are performed in strict accordance to General Electric Quality Assurance Procedures. This includes verification and sign off of all data and independent checking of the accuracy of the data bank.

REQUEST: Section 4 – Steady State Hydraulic Analyses

Provide a subsection heading for the introductory paragraphs (to be consistent with the format of other sections). Clarify the modeling with a modeling diagram of the fuel assembly features which contribute significantly to the hydraulics within a single bundle at various axial elevations. Provide an additional modeling diagram of the model which will be used to determine the steady state hot (and average) bundle flows for each fuel type in a mixed core loading configuration containing up to three fuel types. Discuss the adequacy of the model in view of the various orificing bundle type combinations which can occur. Support the adequacy of the "coarseness" of the licensing hydraulics model with the results of model sensitivity studies which involved "finer" (additional channels) modeling. Discuss the anticipated fuel assembly type/orifice type patterns which will be employed on reloads. Provide a summary description (and reference for the code and solution methods for the hydraulic analyses. Describe modeling considerations related to other primary system components (e.g., steam separator, jet pump, etc.). Discuss the modeling of the bypass flow through the two water rods in the 8x8R fuel assembly. Describe the applicability of the tests, assumptions and modeling of References 4-4, 4-5 and 4-6 for this application.

RESPONSE:

Fuel Assembly Hydraulic Modeling

The fuel bundle and fuel support assembly consists of three basic regions as illustrated in Figure 4A. Figure 4-1 of the report is a schematic showing the features of the lower nonfuel region. During normal operation, this region is characterized by single phase flow. A pressure drop is calculated across the orifice and entrance to the lower tieplate, which includes friction, an irreversible loss due to area change, and a reversible pressure drop due to an area change. Flow through the lower tieplate holes and through the lower tieplate fuel support paths is calculated and subtracted from the flow through the fuel support orifice. The remaining flow passes through the lower tieplate grid, experiencing additional reversible and irreversible pressure changes. The flow through the channel lower tieplate path is calculated and subtracted from the lower tieplate grid flow. Flow through the water rods is also subtracted, as will be discussed separately.

The fueled region is divided into 24 axial segments or nodes over which the heat flux is assumed constant and coolant thermal properties are assumed to vary linearly. Fluid properties and pressure changes (using Equations 4-l, 4-2, 4-3, 4-7 and 4-8) are calculated across each node.

In the upper nonfueled region, friction and elevation pressure drops are calculated from the top of the active fuel to the upper tieplate. At this point, flow is re introduced from the water rods and the fluid properties are recalculated assuming thermal equilibrium. Reversible and nonreversible pressure changes across the upper tieplate into the nonrodded channel section are evaluated. Next, friction loss in this section is calculated, followed by acceleration into the plenum above the core.

"Hot" and Average Bundle Modeling

The "hot" and average bundle hydraulic modeling is as described above. Basically, a core is hydraulically modeled by as many as 12 different fuel types. A fuel type in this sense is described by orificing, fuel geometry (7x7, 8x8, or 8x8R), relative bundle power and leakage characteristics.

The "hot" bundle types are characterized by higher relative power. Because of power differences, "hot" and average central orifice region fuel types will have higher relative powers than corresponding peripheral region "hot" and average fuel types. A typical distribution for a reload core containing 7x7, 8x8, and 8x8R fuel is shown in Table 4A.

Sensitivity studies demonstrate that there is no need to divide the core power distribution into smaller increments. The core pressure drop, which provides the driving head for establishing the "hot" channel flows, is determined predominantly by the average channels. Thus, a core is adequately modeled by "hot" and average channels.

Orificing – Bundle Type Combinations

As discussed above, the various orifice bundle combinations are represented by the different fuel types used by the hydraulic model. This modeling is adequate to represent the various fuel assembly type/orificing patterns that occur in BWR's.

Digital Computer Code

Steady state, core hydraulic analyses are performed using a digital computer program which hydraulically simulates BWR cores. The program user must specify the reactor core power level and distribution, inlet flow conditions, core operating pressure, and a hydraulic description of the reactor fuel assemblies. The program will calculate either the required total flow and flow distribution for a specified core pressure drop or the flow distribution and core pressure drop for a specified total core flow rate.

The hydraulic model allows for up to 12 distinct parallel channels connected to common upper and lower plena. The power level and distribution, geometry, and hydraulic characteristics must be specified for each channel type and may differ in each channel type. A single parallel bypass region is also included in the hydraulic model. The hydraulic model is a constant pressure model, in the sense that all fluid properties are evaluated at a constant, user specified core operating pressure.

The solution procedure consists of a trial and error iteration of flow rates and pressure drops with concurrent calculation of enthalpy, quality and void distributions in individual channel types.

The iterations are usually initiated with physically derived initial guesses and generally involve either simple linear or quadratic interpolation techniques. User defined convergence criteria are employed.

Modeling of Other Primary System Components

Modeling considerations related to other primary system components (e.g., the steam separator, jet pump, etc) are discussed in NEDO-10802, "Analytical Methods of Plant Transient Evaluations for the General Electric Boiling Water Reactor", dated February 1973. In general, however, the other primary system components outside of the core shroud and above the reactor water level are not explicitly modeled in the steady state thermal hydraulic model.

Water Rod Modeling

The flow rate through each water rod in a given fuel assembly is iteratively calculated by balancing the lower to upper tieplate pressure drop in the external active flow region with the internal overall water rod pressure drop. Hydraulic modeling of the water rods includes entrance and exit losses, friction, elevation, and acceleration heating losses. The entrance loss modeling assumes a single phase liquid, while the exit loss formulation includes a two phase multiplier. The default value is the homogeneous multiplier, but it is possible to input other values in a linear with quality formulation, i.e.:

$$\phi^2_{\mathrm{TPL}} = 1 + \mathrm{C}_1 \cdot \mathrm{X}$$

where X is the quality.

A similar two phase multiplier formulation is included in the nodal calculation of the water rod frictional losses.

Energy deposition rates in the water rod coolant are calculated nodally and consider the effects of gamma heating, neutron slowing down, and the combined convective conductive heat transfer rate from the external coolant. The nodal energy balance is used to determine the quality and void distributions required in the hydraulic analysis.

The water rod hydraulic modeling assumes that all of the water rods in a given fuel assembly are geometrically and hydraulically identical. The water rods may be either round or square in cross sectional shape and can be of different outer diameter and/or clad wall thickness than the fueled rods in the same assembly.

Applicability of References 4-4, 4-5 and 4-6

References 4-4, 4-5 and 4-6 provide the models for two phase friction and local pressure drops and void fraction.

The void fraction correlation used is a version of the Zuber Findlay model (Reference 11), where the concentration and void drift coefficient are based on comparison with a large quantity of world wide data (see attached references).

The General Electric Company has taken a significant amount of single- and two-phase pressure drop data in full scale geometries representative of 7x7, 8x8 and 8x8R fuel and correlated the two-phase multipliers on a best fit using the two-phase models reported in References 4-4 and 4-5 in conjunction with the void fraction model. The typical test conditions (Table 4B) are representative of BWR operating ranges. Figure 4A demonstrates the applicability of these models.

Void Fraction Models

1. Isbin, H.S., Rodrrguez, H.A., Larson, H.C., and Pattle, B.D., "Void Fractions in Two Phase Flow", A.I. Ch.E. Journal, Volume 5, No. 4, 427-432, December 1959.
2. Isbin, H.S., Sher, N.C., Eddy, K.C., "Void Fractions in Two Phase Steam-Water Flow", A.I. Ch.E. Journal, Volume 3, No. 1, 136-142, March 1957.
3. Marchaterre, J.F., "The Effect of Pressure on Boiling Density in Multiple Rectangular Channels", ANL-5522, February 1956.
4. Janssen, E., and Kervinen, J.A., "Two Phase Pressure Drop in Staight Pipes and Channels; Water Steam Mixtures at 600 to 1400 psia", GEAP-4616, May 1964.
5. Cook, W.H., "Boiling Density in Vertical Rectangular Multichannel Section with Natural Circulation", ANL 5621, November 1956.
6. Mauer, G.W., "A Method of Predicting Steady State Boiling Vapor Fractions in Reactor Coolant Channels", WAPD-BT-19, June 1960.
7. Rouhani, S.Z., "Void Measurements in the Region of Subcooled and Low Quality Boiling", Symposium on Two-Phase Flow, University of Exeter, Devon, England, June 1965.
8. Firstenberg, A., and Neal, L.G., "Kinetic Studies of Heterogeneous Water Reactors", STL 372-38, April 15, 1966.
9. Ferrel, J.K., "A Study of Convection Boiling Inside Channels", North Carolina State University, Raleigh, N.C., September 30, 1964.
10. Rouhani, S.Z., "Void Measurements in the Region of Subcooled and Low Quality Boiling", Part II, AE-RTL-788, Akticbolaget, Atomenergi, Studsvik, Sweden, April 1966.
11. Christensen. H., "Power to Void Transfer Functions", ANL-6385, July 1961.
12. Egen, R.A., Dingee, D.A., Chastain, J.W., "Vapor Formation and Behavior in Boiling Heat Transfer", BMI-1163, February 1957.

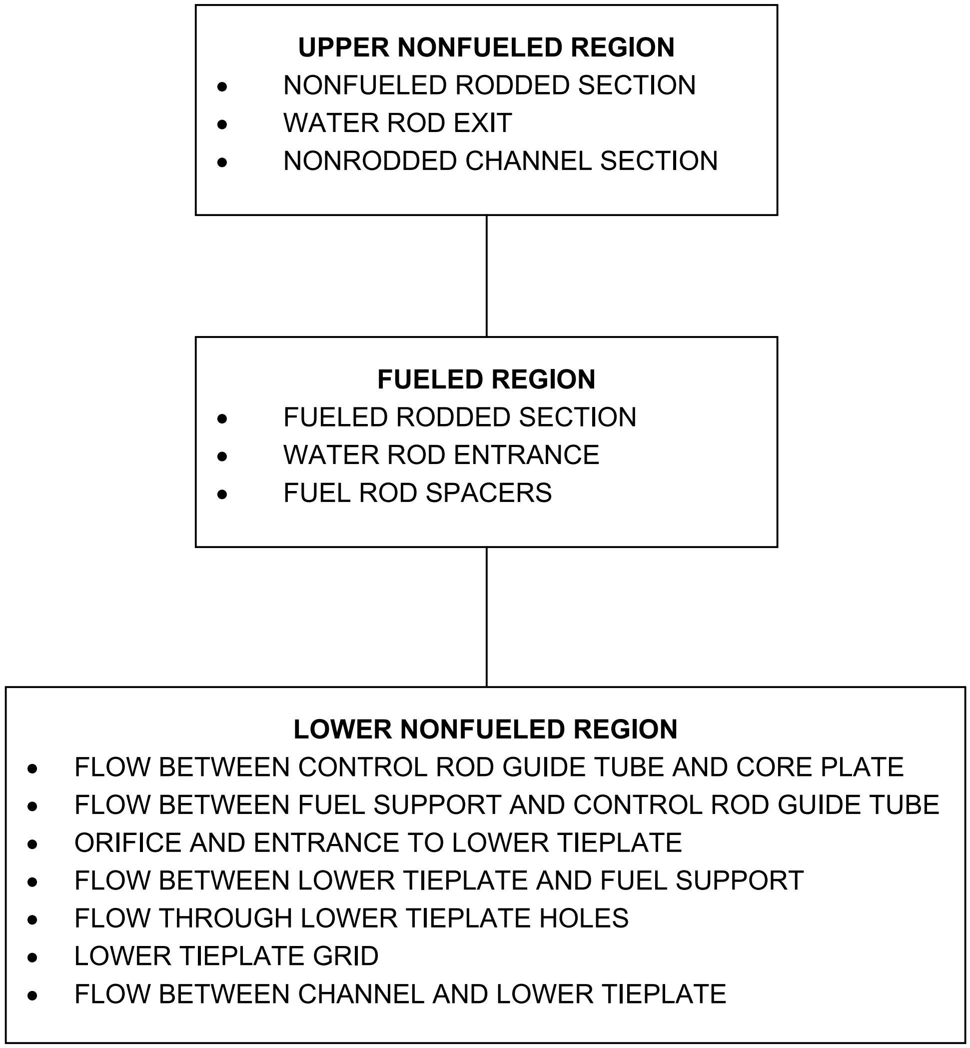

Figure 4A. Fuel Bundle – Support Hydraulic Modeling

4A. Description of a Typical Reload Core

Type	1	2	3	4	5	6	7	8	9
Identification	8x8R	8x8R	8x8R	8x8	8x8	8x8	7x7	7x7	7x7
Orifice Zone	Central	Central	Periph-eral	Central	Central	Periph-eral	Central	Central	Periph-eral
Number Fuel Assemblies	1	263	0	1	299	56	1	107	36
Relative Assembly Power	1.40	1.039	0.700	1.40	1.039	0.700	1.4	1.039	0.700

4B. Typical Range of Test Data

Measured Parameter	Test Conditions
Adiabatic Tests:	
Spacer single-phase loss coefficient	NRe* = 0.5 x 10^5 to 3.5 x 10^5
Lower tieplate + orifice single-phase loss coefficient	T = 100 to 500°F
Upper tieplate single-phase friction factor	P = 800 to 1400 psia
Spacer two-phase loss coefficient	G = 0.5 x 10^6 to 1.5 x 10^6 lb/h-ft^2
Two-phase friction multiplier	X = 0 to 40%
Diabetic Tests:	
Heated bundle pressure drop	P = 800 to 1400 psia
	G = 0.5 x 10^6 to 1.5 x 10^6 lb/h-ft^2
*Reynolds number	

REQUEST: Subsection 4.1
Provide or reference the friction factors used in modeling.

RESPONSE:
[[

]]

REQUEST: Subsection 4.2
Provide or reference the tests discussed in the section.

RESPONSE:
Figure 4.2-1, attached, is a graphical presentation of the test results showing bundle pressure drop vs. bundle power with mass flux as a parameter. The solid line is GE’s analytical prediction method of relating the three components of bundle power, flow, and pressure drop together. The tests were performed in this ATLAS test facility on a 8x8R assembly.

[[

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Figure 4.2-1. Overall Pressure Drop vs. Bundle Power, BWR/2–5 (1,000 psia)

REQUEST: Subsection 4.3

A term of the acceleration of gravity is required for Equation 4-3, if r is in the conventional units of lbm/ft^3.

RESPONSE:

Using the conventional units of lbm/ft^3 for density in Equation 4-3, the acceleration due to gravity is unity and, therefore, was not included in the equation. Equation 4-3 has been modified to read:

$$\Delta P_E = \rho_g L.$$

REQUEST: Subsection 4.5 – Bypass Flow

Reference tests and analyses from which the empirical fits were established.

RESPONSE:

The experimental tests and analyses that have been performed to establish the flow coefficients for the bypass flow paths are described in Subsections 4.2, 5.2 and 5.3 in Reference l. This reference has been added to Subsection 4.5.

REFERENCES:

1. "Supplemental information for Plant Modification to Eliminate Significant In-Core Vibration," NEDE-21156 (Proprietary), January 1976.

REQUEST: Section 5 – Operating Limits

Provide a section heading for the introductory paragraphs of this section to be consistent with the format of the other sections.

The design Linear Heat Generation Rate and Average Planar Linear Heat Generation Rate vs. exposure are important operating limits which should be addressed in this section for each fuel type. A brief discussion of their relation to reload transient and accident analysis should be included. Their relation to the 1% plastic cladding strain and 2200°F PCT for local transients (e.g., RWE, FLE) and LOCA, respectively, should be addressed. Additionally, gross core thermal power reductions toward EOC, from 100% power, may be required to maintain adequate margin between the lowest safety valve setpoint and peak pressure, from the most severe overpressure event. This potential gross core power (operating) limitation should be briefly discussed in this section.

The design requirement that the peak transient reactor vessel pressure not exceed 110% of the design pressure should also be included in the transient limits paragraph.

This section should also include a brief discussion of the operating procedures which BWR's follow during normal operation, to minimize PCI fuel failures. Reference to a more detailed discussion would also be appropriate.

RESPONSE:

The introductory paragraphs of this section develop the safety analysis philosophy which is used in subsequent sections to establish the limits for specific events. As such, specific limits and event descriptions are not contained.

The discussion of the limits for specific events is contained in the section which describes that event. If specific limits for a given event were described in a general section, they might be misinterpreted and applied to a different category of events. This could result in inappropriate limits being established.

As examples, the limit of 110% of design pressure is the General Electric interpretation of an ASME Code requirement. It is provided in the safety analysis as a demonstration that the Code requirement is satisfied for a given reload.

The event analyzed has been demonstrated to be more severe than any abnormal operational transient. Therefore, it is not included as a limit for transients.

As discussed in the response to the request on Subsection 2.5.3.1.4, General Electric has concluded that pellet cladding interaction failures do not represent a significant safety concern. Therefore, discussion of a recommendation to minimize an operational inconvenience is not appropriate for a section describing limitations placed on a reactor for safety considerations. This is also applicable for the recommended power reduction to maintain margin to the lowest safety valve setpoint.

REQUEST: Subsection 5.1.1 – Statistical Model

Reference and provide the GE report which describes the BWR process computer function.

The design power distribution (histogram), which is an important aspect of the statistical model used to establish the fuel cladding integrity safety limit, does not distinguish between 7x7 (49 rods) and 8x8 or 8x8R (64 rods) fuel types. For a mixed core reload, configurations there will, in general, be assembly wise power histograms for each fuel type. Provide either a description of the modification made to the statistical procedure for this situation or the justification for using a single design power distribution, for calculating the number of rods in the core in boiling transition. Discuss the procedures used to develop the histogram(s).

RESPONSE:

The GETAB fuel cladding integrity safety limit analysis is performed by a Monte Carlo controlled computer code which only represents the process computer functional. Therefore, the topical report on the process computer is not applicable to this calculational procedure. The reference for the calculational procedure and uncertainties is Appendix IV in Reference 5-1.

The current design procedure is to perform a bounding analysis establishing a single valued reload safety limit MCPR which applies to all the reload cycles for a given class of reload cores. As such, the procedure to develop the design power distribution for the statistical analysis is, first, to search for a reload core loading configuration which would yield the worst CPR distribution in the core among the reload cycles, and, second, to search for the control rod pattern which yields as many fuel assemblies as possible at and near the MCPR operating limit at rated reactor power as per the procedure described in detail in Appendix IV, GETAB Licensing Topical Report NEDO-10958.

In generating the design power distribution, the 8x8 and the 8x8R reload cores are treated as different classes of reload cores because of differences in fuel type and core loading configuration. A 251/764 core 8x8R equilibrium cycle for the 8x8R reload core and a 251/764 core-7x7/8x8 first mixed cycle for the 8x8 reload core are selected for the statistical analysis. This selection was based on the results of evaluations which indicated that a least CPR mismatch between different fuel batches is expected in these cycles, thereby providing a worst CPR distribution among the reload cycles for their respective reload core class.

The term “worst” is used here describing the power distribution in the context of “relative” rather than “absolute” sense. Therefore, it is entirely possible to have some variation in high power flattening from the worst distribution for one case to the other (i.e., between the histograms used in establishing the safety limit MCPR for the 8x8 and 8x8R reload cores). What is important is to assure that the licensing basis histogram is conservative relative to the expected operating histograms for their respective reload core class.

The design basis CPR and relative bundle power histograms for the 8x8R reload core (Figure 1) are compared with two sets of operating histograms (Figures 2 and 3). The operating histograms include the histograms representing both severe and typical of those expected at the BWR/4 8x8R reload core when the reactor is operated at rated core power with the most limiting bundle close to the MCPR operating limit. As can be seen, the expected operating distributions are much more peaked than the ones assumed in the licensing basis, thus yielding an expected operating 99.9% statistical limit MCPR of approximately 1.01 to 1.03. This example illustrates the degree of conservatism included in the licensing basis safety limit MCPR.

In addition, an analysis was performed to evaluate the conservatism in the technical specification safety limit MCPR which was derived for the design basis power distribution. The result showed that the actual operating power distribution was much more peaked than the one assumed in design analysis, thus yielding the 99.9% statistical limit MCPR of 1.00, which is 5% lower than the technical specification safety limit MCPR of 1.05 (7x7 initial core). The following parameters were used in the analyses:

Operating Plant Data:	
Core Power:	3252 MW (99% rated)
Operating MCPR:	1.25
MCPR Operating Limit:	1.25
Radial Power Distribution:	Figure 5.1.1 4
The 99.9% Statistical Limit MCPR:	1.00
Design Basis:	
Technical Specification Safety Limit MCPR:	1.05
Design Basis Power Distribution:	Figure 5.1.1 5

The above results were discussed briefly in page 2 of Reference 1 in response to the NRC question on the effect of the TIP uncertainty on safety limit MCPR.

REFERENCES:

1. "Process Computer Performance Evaluation Accuracy Amendment 1", NEDC-20340-1, December 1974.

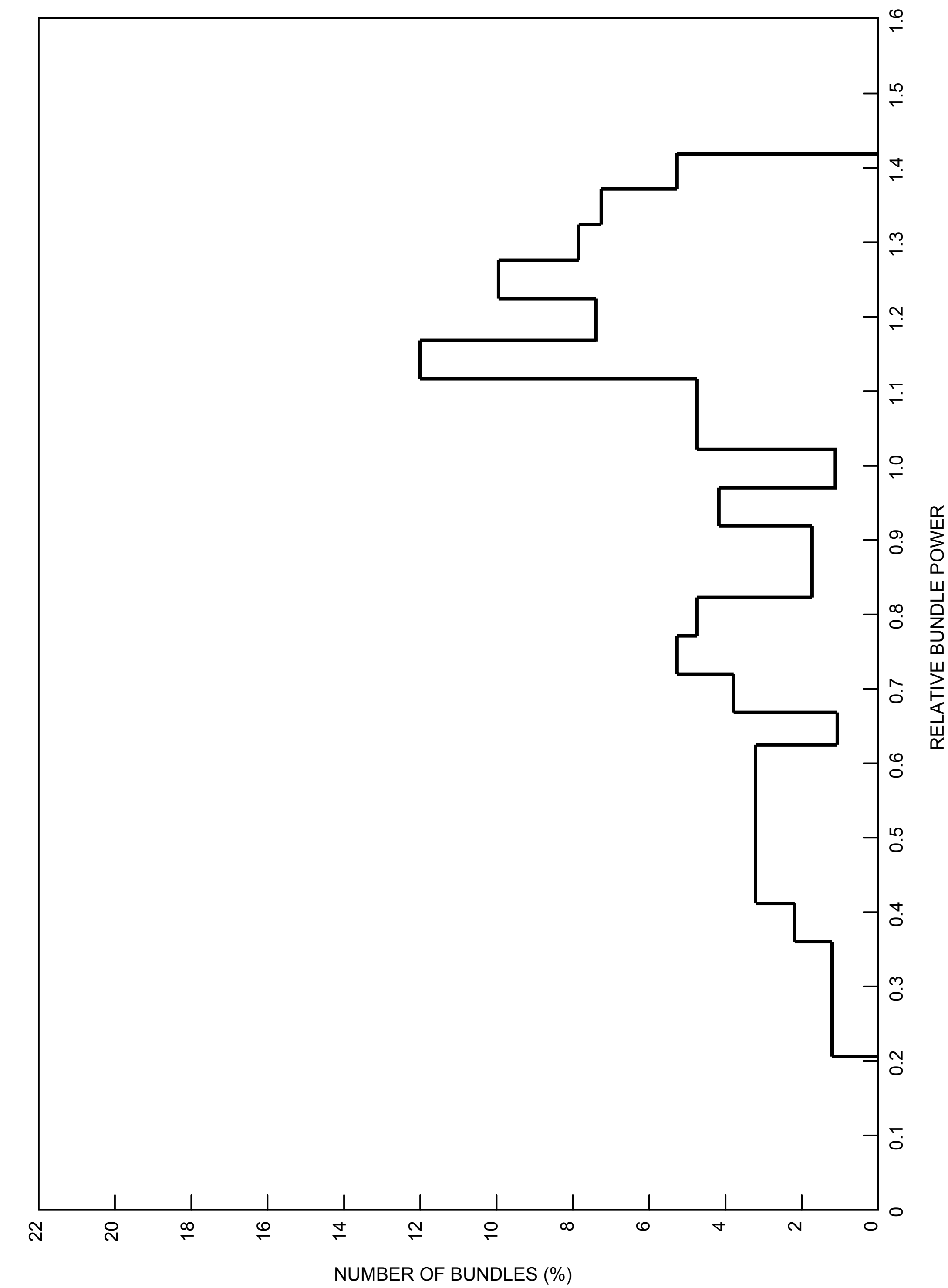

Figure 5.1.1-1a. Relative Bundle Power Histogram for Power Distribution Used in Statistical Analysis for BWR/4 8x8R Reload Core

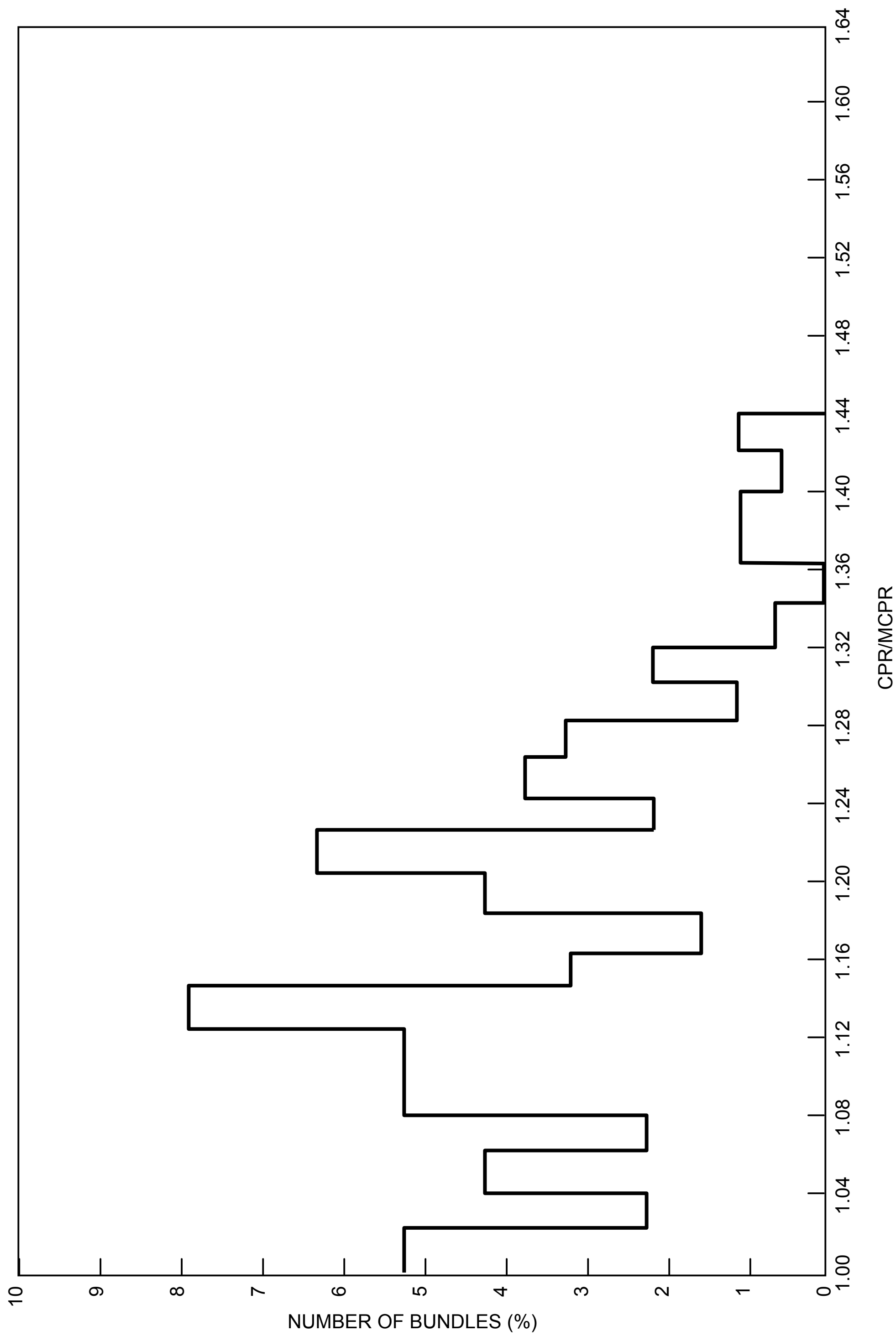

Figure 5.1.1-1b. Normalized CPR Histogram Used in Statistical Analysis for BWR/4 8x8R Reload Core

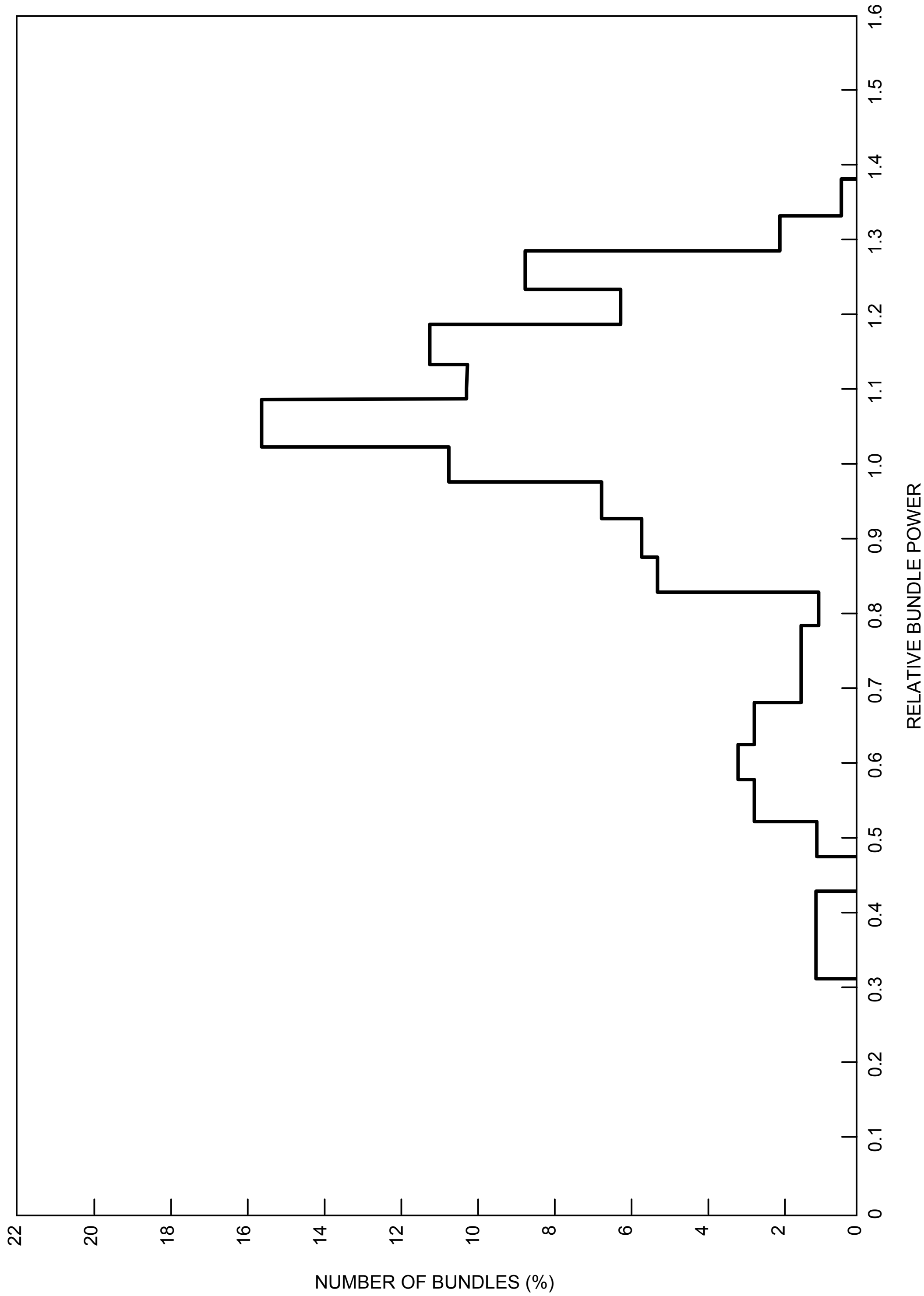

Figure 5.1.1-2a. Relative Bundle Histogram for a Typical 8x8R Reload Core Operating Power Distribution

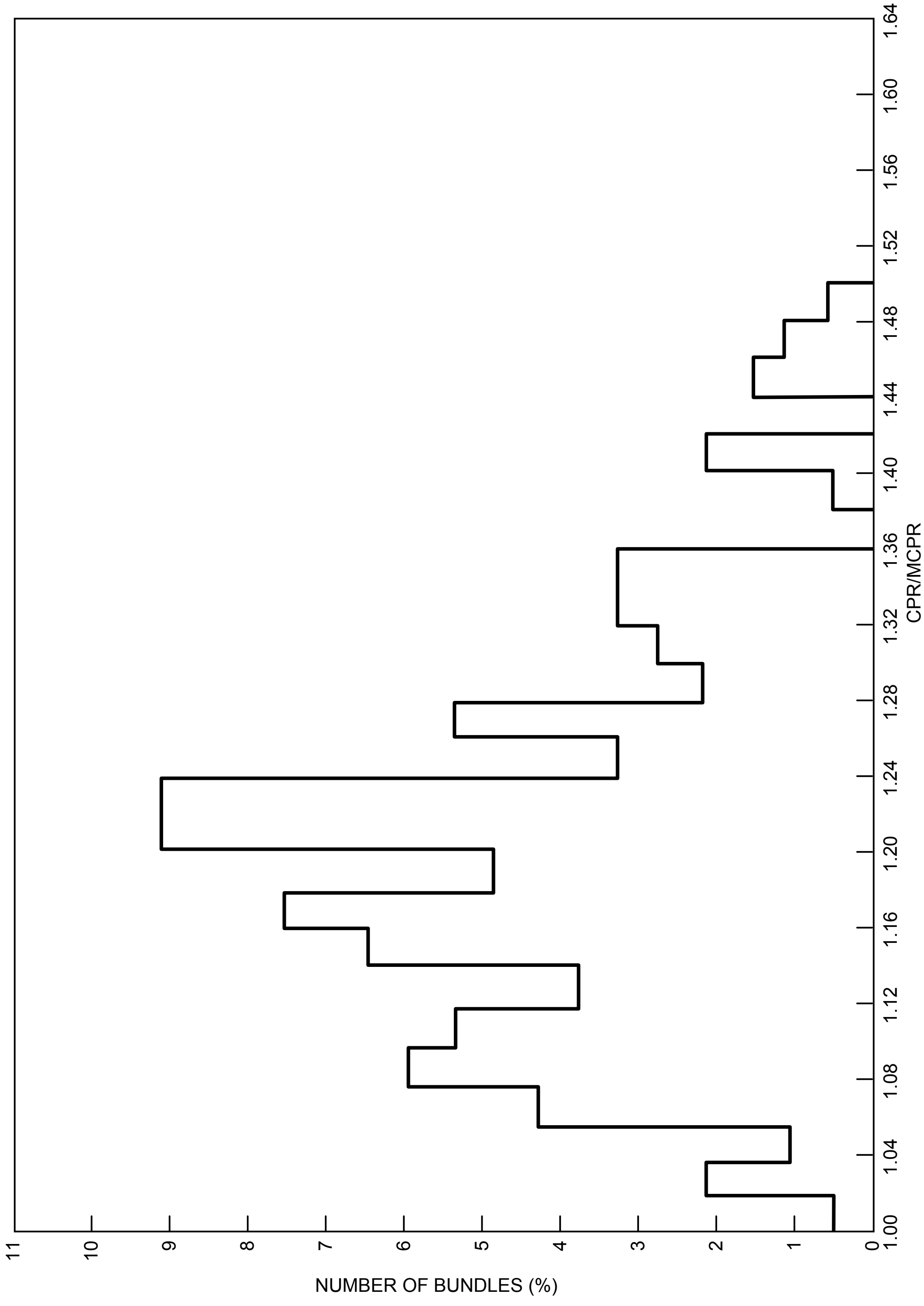

Figure 5.1.1-2b. Normalized CPR Histogram For a Typical 8x8R Reload Core Operating Power Distribution

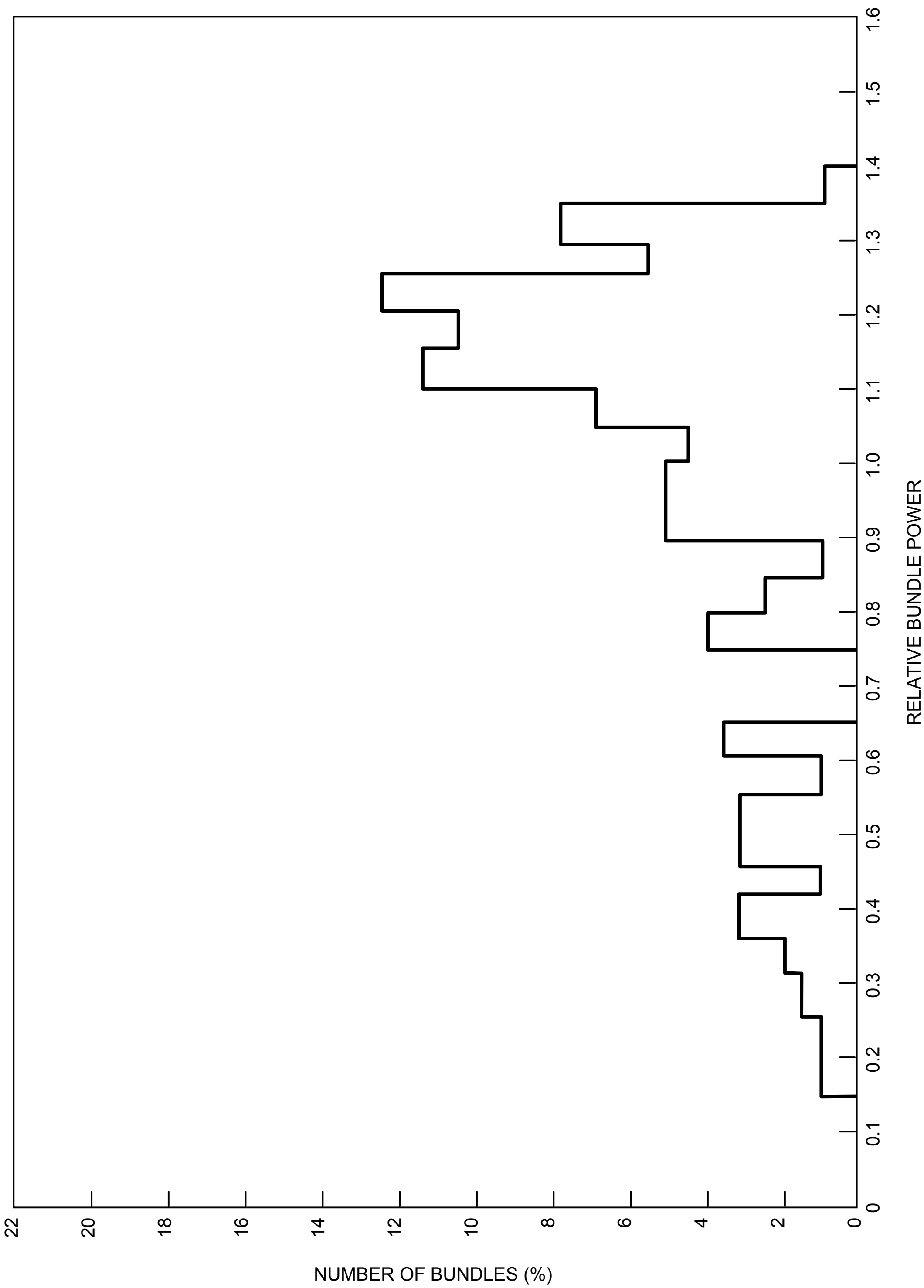

Figure 5.1.1-3a. Relative Bundle Histogram for Relatively Severe 8x8R Reload Core Operating Power Distribution

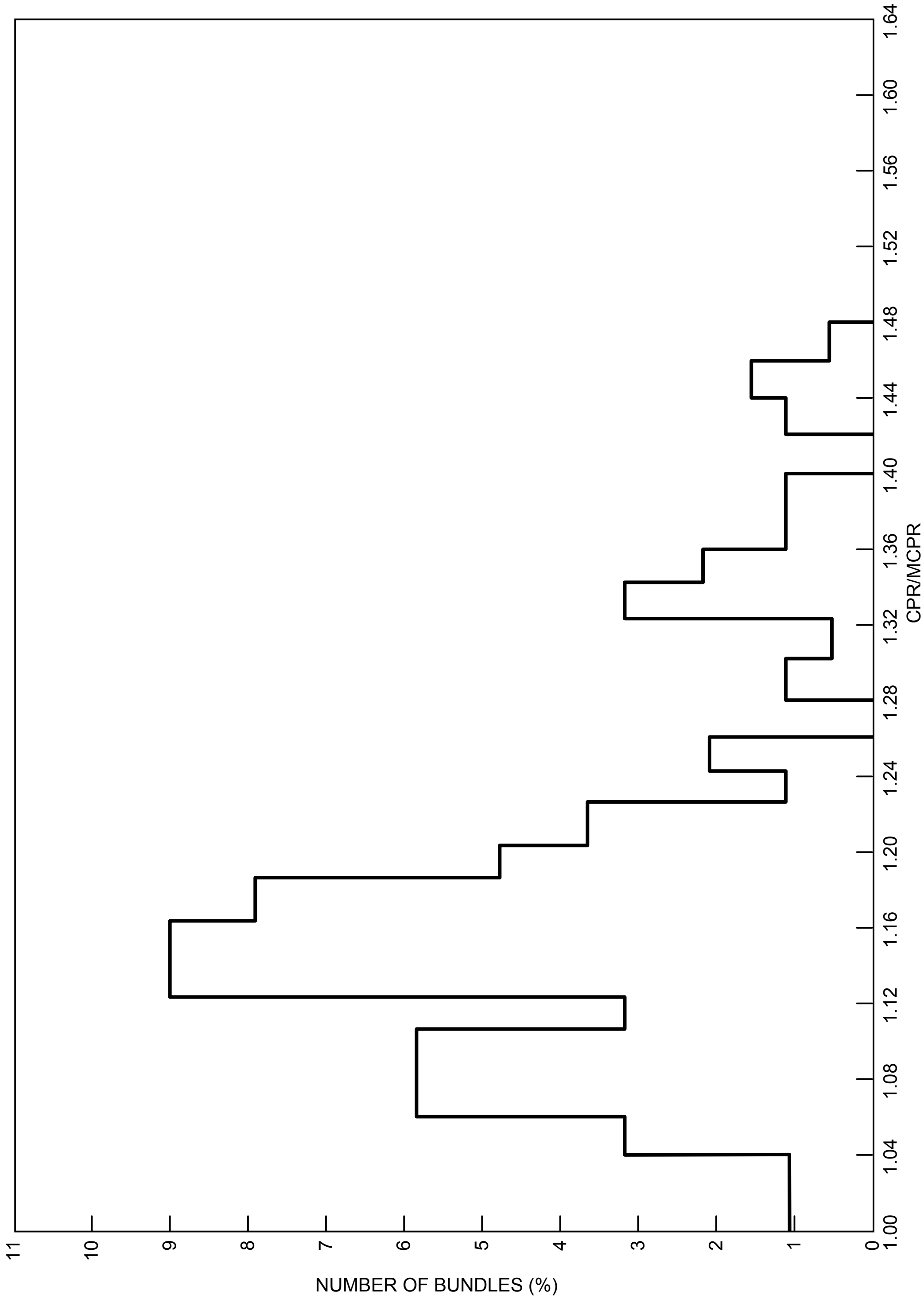

Figure 5.1.1-3b. Normalized CPR Histogram For Relatively Severe 8x8R Reload Core Operating Power Distribution

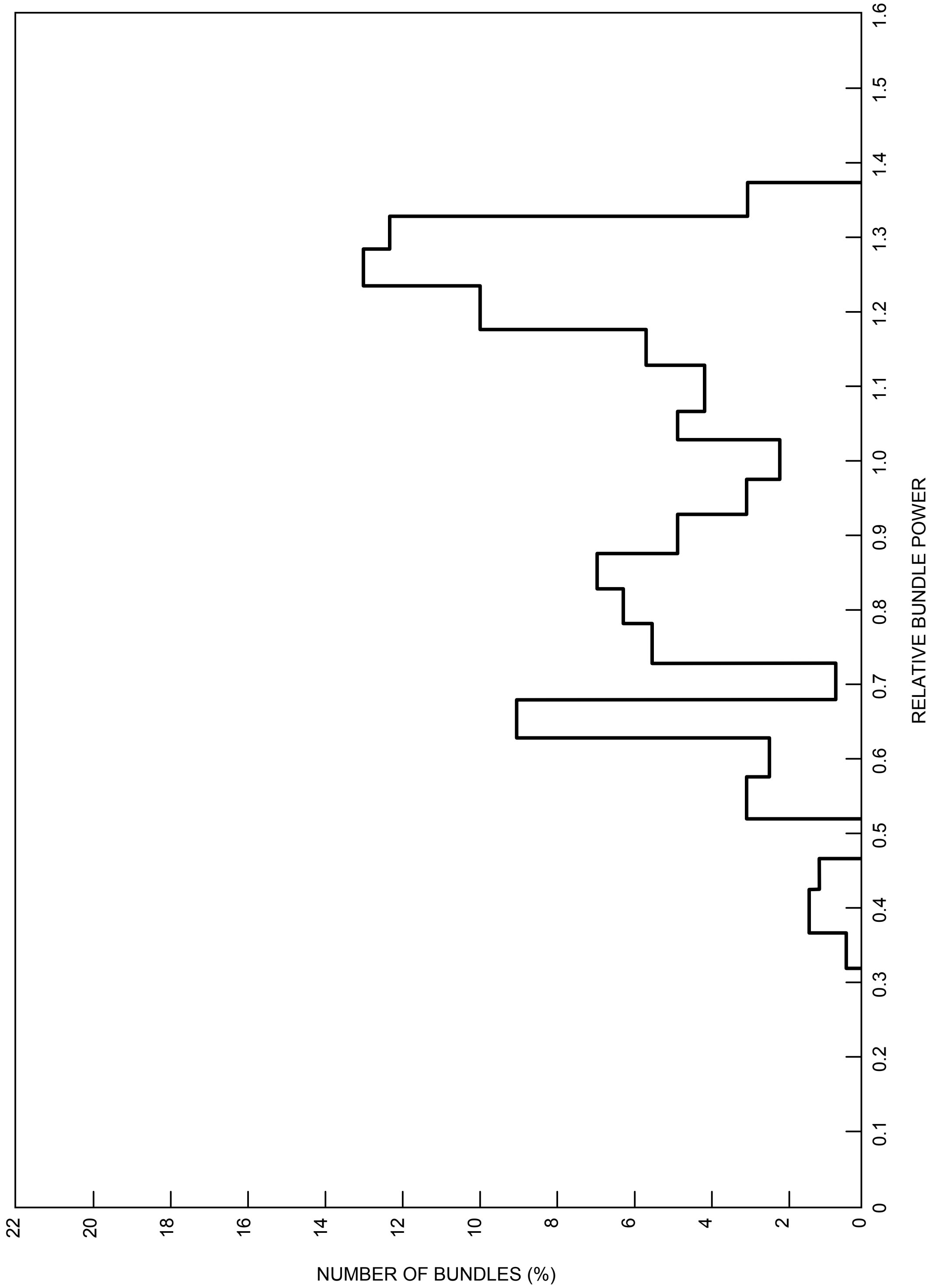

Figure 5.1.1-4. Relative Bundle Power Histogram for Actual Power Distribution

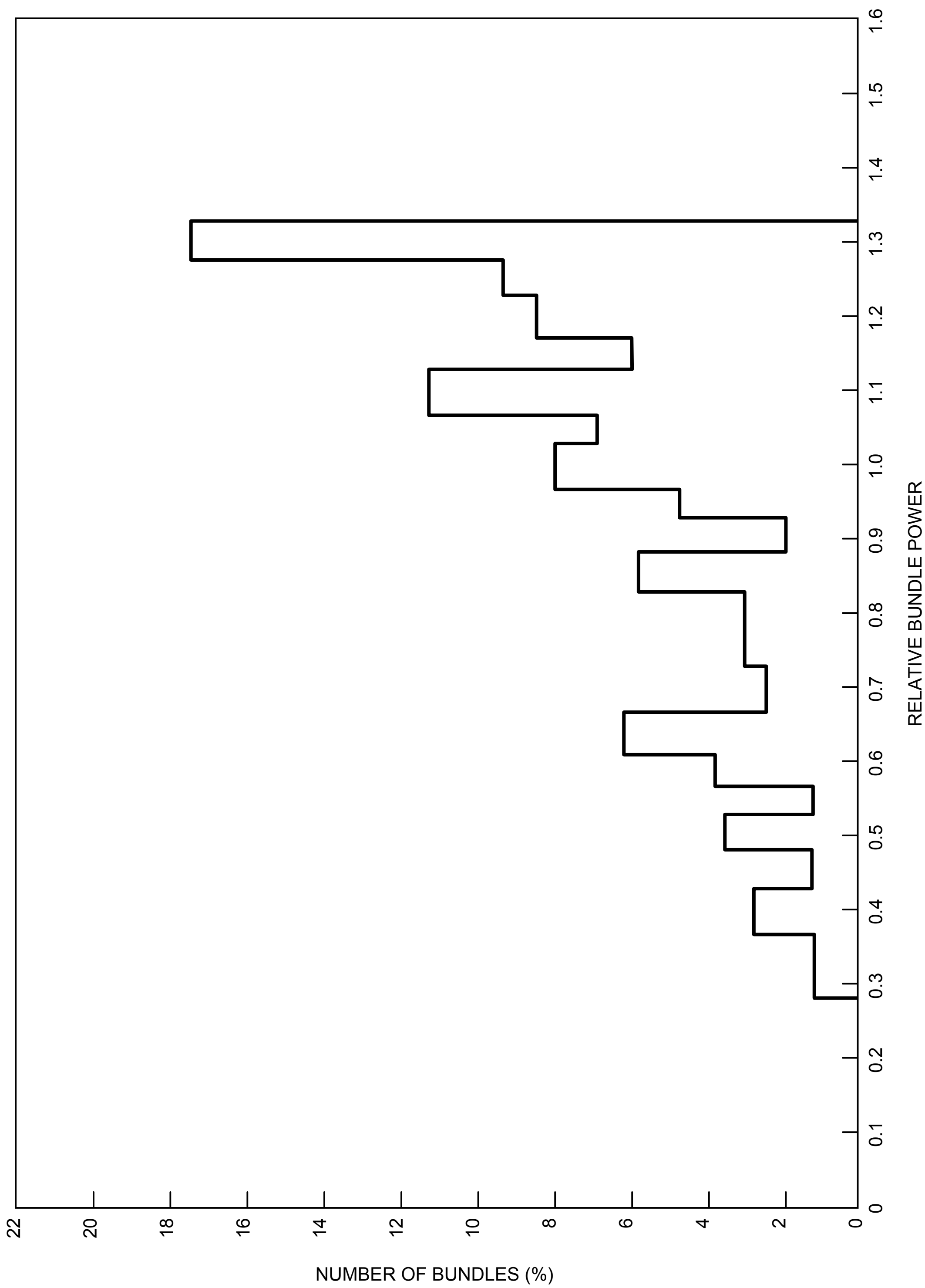

Figure 5.1.1-5. Relative Bundle Power Histogram for Power Distribution Used in Statistical Analysis

REQUEST: Subsection 5.1.2 – Bounding BWR Statistical Analysis

(a) Provide the results of the bounding statistical analysis for 8x8R reloads. Describe the reactor core selected, the relative bundle power histogram assumed, and the rod-by-rod R-factor distribution.

Provide a table of the rod-by-rod R-factor distribution used for the bounding statistical analysis for each fuel type (i.e., 7x7, 8x8, 8x8R) for each class of plant. Indicate fuel enrichments, if appropriate, as well as the cycle exposure.

(b) Using available test data (e.g., Figure 9-10 of Reference 1), quantitatively assess the effect of the nonconservative bias in the GEXL correlation for cells with axially varying R-Factors, in connection with the development of the 1.07 safety limit minimum critical power ratio for normal core operating conditions [i.e., evaluate the magnitude of the underprediction of the number of rods in boiling transition contributed from 8x8R assemblies with axially varying R-Factors (e.g., controlled cells) or conversely evaluate the extent of the underprediction of the SLMCPR (too low a SLMCPR) resulting from an underprediction of the number of rods in boiling transition in assemblies with axially varying R-Factors].

Using available test data, quantitatively assess the effect of the bias described above in connection with the development and applicability of the 1.07 SLMCPR relative to core operations at reduced flow and increased subcooling (e.g., during a TT w/o BP event with prompt recirculation pump trip), since the bias appears to increase for these conditions relative to normal operating conditions.

RESPONSE:

Results of the bounding statistical analysis for 8x8R reloads are presented in revised Subsection 5.1.2. A description of the reactor core selected, the relative bundle power histograms assumed, and the rod-by-rod R-factor distributions are also included. Because R-factor distributions are included in Subsection 5.1.2, the pin-to-pin power distributions which are used to determine these R-factor distributions have been deleted.

Verification that the R-factor uncertainty associated with uncertainties in calculating fuel rod peaking factors is not greater for 8x8R fuel type then for the 8x8 and 7x7 fuel types is as follows. In Reference 5-1, a conservative relation between σ_R and σ_P was shown to be:

$$\sigma_R = \frac{1}{2}\sigma_P$$

where

σ_R = standard deviation of the R factor

σ_P = standard deviation of the local peaking factor

From Equation (1), it can be observed that the standard deviation of the R-factor is directly proportional to that of the local power peaking. The evaluation of the uncertainty of the R-factor, therefore, depends on the determination of the uncertainties of the surrounding local power peaking factors. These local peaking factor uncertainties for the 8x8R fuel types are examined next.

The calculations of the local peaking factors for the different 8x8R, 8x8 and 7x7 fuel designs were done by the same lattice physics code. Since the same numerical scheme was employed, the nuclear uncertainties of the local peaking factors for these different fuel types should be nearly the same. Therefore, the R-factor uncertainties for the 8x8R fuel types are not greater than the 8x8 and 7x7 designs.

In addition, studies have been performed to evaluate the local peaking uncertainties for the 8x8R fuel design. They were done by using the Monte Carlo calculations and high temperature critical experiment data. These studies verified that the 8x8R local peaking uncertainty was about the same as the 8x8 and 7x7 fuel types. Details of these studies are discussed in Subsection 3.2.

The TIP detectors are located at the narrow-narrow water gap (i.e., the noncontrol blade side) of the four bundle assemblies. From an analytical viewpoint, the main source of TIP uncertainty is due to the correlation constants. The correlation constants for the 8x8R and the 8x8 fuel designs are obtained from data utilizing the same lattice physics code. These correlation constants are highly influenced by the localized effect of the narrow-narrow water gap and the surrounding thermal energy group nuclear interactions. The uncertainties of the correlation constants for the 8x8R and 8x8 fuel types are about the same. This is due to two factors:

1. the same numerical scheme is used for their generations, and

2. the extra water rod is sufficiently far away from the narrow narrow gap (several diffusion lengths) that its effect on the uncertainty of the correlation constants are negligible.

Therefore, the TIP measurement is essentially decoupled from small changes at the center of the fuel lattice. The extra water rod in the 8x8R fuel does not contribute additional TIP uncertainty.

In developing the safety limit MCPR of 1.07 for 8x8R reload cores, the partially controlled bundles (PCB) contributed only 0.3% of the total expected number of rods subject to boiling transition (ENRSBT) during a transient. If the GEXL critical power bias (based on the data shown in Figure 9-10 of Reference 1) is considered, which is ~0.015[1] at the operating condition of hot PCB (mass flux ~1.10 Mlb/hr ft^2, subcooling ~25 Btu/lb), the underprediction of the ENRSBT is only ~0.26, which can be translated into an underprediction of the safety limit MCPR by <0.0002, which is negligibly small.

However, it should be emphasized that the test data as shown in Figure 9-10 of Reference 1 fall well within the GEXL critical power predictability (with one sigma, 3.6%, as used in SLMCPR analysis), especially when considering that the deviation is only ~1.5% at typical rated operation condition. In other words, for the assemblies with axially varying R-factor, the GEXL prediction deviation is well covered by the bands of ±7% of the GEXL predictability for the constant R-factor assemblies. The bias shown in Figure 9-10 of Reference 1, a single set of test data, is not expected to be representative of all axially varying R factor cases.

When the core is at reduced flow and increased subcooling condition, the GEXL critical power bias shown in Figure 9-10 of Reference 1 is ~0.05 for an axially varying R-factor assembly, which is based

[1] Based on 14 datapoints (Figure 9-10 of Reference 1) with mass flux from 1.00 to 1.25 Mlb/hr ft^2 and subcooling from 18 to 50 Btu/lb.

on 8 data points with a mass flux of 0.75-1.00 Mlb/hr ft^2 and subcooling of 40-70 Btu/lb. For the development of the 1.07 safety limit MCPR for 8x8R reloads, the underprediction in ENRSBT is ~1.13 rods (from PCB) and it is equivalent to an underprediction in safety limit MCPR by ~0.0006. This small difference has no apparent effect on safety limit MCPR and can be neglected.

In conclusion, the GEXL bias shown in the available test data for the axially varying R-factor assembly has no impact at all in developing the safety limit MCPR of 1.07 for 8x8R reloads.

REFERENCES:

1. GE Letter (R.E. Engel) to NRC (D. Eisenhut), "Fuel Assembly Loading Error", November 30, 1977.

REQUEST: Subsection 5.2 – MCPR Operating Limit

Discuss the plant transient code models (especially the multinoded thermal-hydraulic and heat transfer relationships) in References 5-2, 5-3 and 5-4, which derive their input parameter values from the fuel mechanical design (e.g., fuel rod diameters, stack lengths). Discuss which of these input parameters utilize a "weighted average" value of the various fuel types and which utilize a "limiting" fuel type value, for mixed reload core configurations (7x7/8x8/8x8R). Provide justification in each case. Describe how these fuel related plant transient code inputs are updated from plant to plant, from one cycle (reload) to the next (reload), as the core configuration changes; e.g., from a mixed core (7x7/8x8 8x8R) to a core having a single fuel type (8x8R). Justify the methods or assumptions used, by providing sensitivity studies where appropriate.

Several of the plant specific inputs appearing in Table 5-6 appear to be incorrect. Please make the corrections as noted in the attached Table A.

Several of the pressure relief system characteristics appearing in Table 5-4 appear to be incorrect. Please make the corrections as noted in the attached Table B.

The prompt reactor period to τ_o (1/β) plays a significant role in the Reactor Kinetics model and the dynamic behavior of the core and plant system. The reload supplement must contain the value of to τ_o (or β and l*) used for each reload analysis for each plant.

Provide or reference a description of the bases (models, methods and assumptions) used to develop the nominal EOC and exposure-dependent scram reactivity function for the plant transient analysis.

Provide or reference additional discussion of the source of the uncertainties and biases in the derivation of the nuclear input data. State the approximate magnitudes (% of nominal value) of the uncertainties and biases for each parameter. Provide or reference the results of sensitivity studies, which show the influence of the DCF's on the key transient output parameters (e.g., ΔCPR, peak pressure, fuel temperature) values for the limiting core wide events. Relative to Table 5-5, indicate which moderate frequency events will be analyzed as power increase transients and which will be analyzed as power decrease transients.

Discuss the extent to which plant-specific values are used for physical parameter inputs to the system transient code (e.g., steamline volumes, core volumes, plenum volumes, system masses) when performing plant specific reload licensing calculations.

Provide a detailed block diagram which illustrates the flow of the key inputs and outputs among the various computer codes (nuclear, system, hydraulic, thermal-hydraulic) used in the determination of the MCPR operating limit (i.e., ΔCPR) for a limiting local (e.g., RWE) event. Provide an additional block diagram for core wide events. Indicate where iterations may be required.

Justify the use of selected axial profile distribution given in Table 5-7 for the SCAT code analysis for all exposures and cycles of all of the plants (Table 1-1). Discuss why the selected profile, when used in connection with various radial and local peaking (R) factors during the cycle, is conservative. Discuss the sensitivity of transient ΔCPR to axial power profile.

The initial operating limit MCPR, assumed for CPR transient analysis, must equal or exceed the finally established operating limit in order to conservatively determine the ΔCPR. Provide a sensitivity plot of this effect and a statement of its consideration in reload thermal hydraulic analysis.

The brief paragraphs, which discuss the operating limit MCPR/low flow correction curves, K_f, for reduced core flow conditions, do not adequately describe the basis for the construction of the generic curves shown in Figure 5-7. A substantially more detailed discussion of the procedure employed is required. Describe the models, methods and assumptions used, which make the curves generic for BWR/2, BWR/3 and BWR/4 plants incorporating up to three different fuel designs. Either justify why the new fuel design (8x8R) is conservatively bounded by the analysis or provide an additional thermal hydraulic analysis to demonstrate that the indicated curves are bounding. Clearly state the criteria used for developing the curves for the automatic and manual flow control modes.

The pressure ordinate values in Figure 5-8 appear to be incorrect as shown. Please correct. The units and ordinate values for the peak surface heat flux, on the same figure, appear to be inconsistent. Please correct.

The use of an <u>example</u> of an event (RWE), which is more severe (larger ΔCPR at reduced power and flow, to show that the flow increase event and (K_f curves) is bounding at reduced flow is <u>not</u> adequate. Add the generic results to Table 5-8 of all other anticipated transients which become more severe at reduced flow and power; e.g., inadvertent startup of an idle loop, feedwater flow controller failure (increase). Explain the methods which were used to make the analysis of these events generic for all plants appearing in Table 1-1.

Describe the procedure used for establishing the slope of the linear approximation curve to the actual exponential power decay (coastdown) curve beyond end-of-cycle. Reference 5-8 does not discuss 8x8R fuel and does not present a generic analysis. Provide a generic analysis and discussion for an 8x8R reload core to support the conclusion that coastdown operation beyond full power operation is conservatively bounded by the EOC analysis.

RESPONSE:

The plant transient code models heat transfer with a single fuel element representing the entire core (Reference 5-2 in the report). Current procedures require this element to be the dominant fuel type (7x7, 8x8, or 8x8R), not a weighted average. As successive reloads move the core towards all 8x8R fuel, the inputs are changed from 7x7 and 8x8 towards 8x8R when that type becomes dominant (i.e.,

≥50% of the bundles). Fuel parameters input are: rod length, rod diameter, clad thickness, pellet diameter and the number of fueled rods per bundle.

Fuel-clad gap conductance, on the other hand, is a weighted average (rather than the expected value of a pellet with 1.0 peaking). It is calculated for a single fuel type core and is dependent on both fuel type and product line.

Sensitivity studies (Reference 5-2) have shown a weak effect of fuel time constant on the peak transient results. The trend is towards higher pressure and heat flux peaks on limiting pressurization events for a smaller fuel time constant. (Smaller rod size → smaller time constant.)

In addition, to determine the effect of using the dominant fuel type characteristics (i.e., fuel dimensions) to determine the core average transient response, as opposed to the fuel type with the fastest time constant, the generator load rejection without bypass transients were performed considering each of the three (7x7, 8x8 and 8x8 retrofit) fuel types to be dominant. The nuclear and thermal hydraulic inputs were based upon the following core makeup:

Fuel Type	No. Bundles in Core	Active Fuel Length (in.)
7x7	324 (42.4%)	144
8x8	187 (24.5%)	144
8x8 Retrofit	253 (33.1%)	150
Total	764	

The results of the core average transient response were:

Maximum	Fuel Type Assumed Dominant		
	8x8R	8x8	7x7
Steam line Pressure (psig)	1196	1196	1197
Vessel Pressure (psig)	1239	1239	1241
Neutron Flux (% initial)	289.7	292.1	305.9
Heat Flux (% initial)	112.4	112.4	111.9

These results indicate that varying the fuel type has only insignificant effects on the core average transient response.

The values of the plant inputs in Table 5-6 were corrected in Amendment 1. At present, nominal transient input values and allowable tolerances are given in Table 5-6 and GETAB initial conditions are documented in Table 5-8. Nominal transient input values were placed in Table 5-6 to preclude revisions with every reload. The values in Table 5-4 have been corrected. As indicated in the response to the request on Section 5 of Appendix A, the prompt reactor period should not be included in each reload submittal.

The core condition which serves as the starting point for the EOC scram reactivity calculation is determined by performing a Haling calculation.

At exposure points before EOC, the Haling power distribution is used to accumulate exposure from BOC to the exposure point of interest. The required control rod inventory for the before-EOC points are designed to achieve criticality, minimize scram reactivity response, and yield reasonable thermal performance.

Further descriptions of the models, methods, and assumptions for the calculation of the scram reactivity are given in Reference 1.

The results of sensitivity studies which show the influence of design conservative factors (DCF) on the key transient output parameters for the limiting events are found in Reference 5-2.

The moderate frequency events which are analyzed as power decrease transients are those events which result in a core coolant flow decrease. These events are: (1) recirculation flow controller failure-decreasing flow; (2) trip of one recirculation pump; and (3) trip of two recirculation MG set drive motors. Identification of power decrease and increase transients is given in Table 5-5.

Plant specific values are provided for physical parameter inputs to the system transient code as described in Reference 5-2. The parameters for each plant are checked with the plant owner and updated for each reload as necessary with current parameter data when performing plant specific reload licensing calculations.

Steady-state hydraulic calculations such as core bypass flow and core and channel pressure drops are initially performed. The results of these analyses are used as input for the nuclear evaluations (Section 3) to determine the nuclear transient input values. The hydraulic and nuclear evaluation results are then used as inputs to the core-wide transient analyses (Subsection 5.2). Operating MCPR limits (Subsection 5.2) for core-wide transients are determined from the results of the transient analyses. The nuclear models are used to determine the RWE transient results and MCPR as a function of rod position. Information relative to the models employed are documented in the above indicated section or subsection.

A description of the transient analysis procedures for abnormal operational transients is also given in Section 6.5.2 of Reference 5-1. Justification of the axial profile used in the transient analyses is given in Appendix V of Reference 5-l

The initial MCPR assumed for transient analyses is usually greater than or equal to the GETAB operating limit. The attached Figure 5.2-1 illustrates the effect of the initial MCPR on transient ΔCPR for a typical BWR core. This figure indicates that the change in DCPR is 0.01 for a 0.05 change in initial MCPR. Therefore, in some cases, nonlimiting GETAB transient analyses may be initiated from a MCPR below the operating limit because the higher operating limit MCPR more than offsets the increase in ΔCPR for the event. This may also be applied to limiting transients if the difference between the operating limit and the initial MCPR is small (0.01 or 0.02).

The method of development and justification of the K_f curves were previously provided to the NRC in response to questions on Quad Cities 2, Reload 1, in 1975. The purpose of the K_f factor is to define MCPR operating limit requirements at other than rated flow conditions. Specifically, the K_f factor provides the required thermal margin to protect against a slow flow increase transient (where a flux scram does not occur), which is the most limiting reduced core flow event

Development Criteria – Manual Flow Control Mode

The manual flow control mode K_f factors were calculated such that at the maximum flow state (as limited by the pump scoop tube setpoint) and the corresponding core power (along the reference flow control line) the relative power of the limiting bundle was adjusted (upward) until the MCPR was at the safety limit. Holding this relative power distribution constant, MCPR was calculated at several points along the flow control line.

The ratio of the MCPR's calculated at these points, divided by the operating limit MCPR, determines K_f at each point. This procedure was repeated for each maximum limit (scoop tube setting). The following is an example derivation of the 102.5% flow K_f curve for a BWR/4 core containing an essentially equal number of 7x7, 8x8, and 8x8R fuel bundles.

The power-flow relations used in this example are provided in Figure 5.2 2. Pertinent core parameters are:

Rated Core Power		3293 MW	
Rated Core Flow		102.5 Mlb/hr	
Total Number of Fuel Assemblies		764	
	7x7	**8x8**	**8x8R**
Number of Fuel Assemblies	254	255	255
R-Factor	1.100	1.100	1.051
Maximum Radial Power Factor	1.499	1.624	1.771

At 102.5% rated flow and 101.8% rated power, the MCPR is 1.06 for the maximum powered assemblies listed above. The MCPR for each of these fuel assemblies as a function of the core power and flow is tabulated in Table 5.2-1. Assuming a 1.20 operating limit MCPR for each of the fuel types, a K_f curve can be generated for the 102.5% maximum flow line using the relation of

$K_f = \frac{\text{MCPR (W)}}{1.20}$, where MCPR (W) is the MCPR at a given core flow fraction from Table 5.2-1.

Note that as a particular plant's operating limit MCPR increases above 1.20, the K_f curves (Figure 5-7) will become increasingly conservative (i.e., if the operating limit was to increase for a given reload 1.20 to 1.30, the curves would be overly conservative by the ratio 1.30/1.20, because the change in critical power ratio with core flow should not change from cycle to cycle). This "overconservatism" is presently being reviewed and future changes to the K_f approach are being considered.

Figure 5.2-3 compares the derived 102.5% core flow K_f curve with the standard curves (Figure 5-7). These results show that the 8x8R fuel closely follows the 102.5% line, but the 7x7 fuel is 3.3% nonconservative at 40% core flow. However, the K_f curves are conservative for all cores with 7x7 operating limits greater than 1.23.

Development Criteria – Automatic Flow Control Mode

For operation in the automatic flow control mode, the same procedure is employed except the initial power distribution is established such that the MCPR is equal to the operating limit MCPR at rated power and flow. Thus, it is assured that automatic power/flow increases will result in meeting the operating limit MCPR for any power/flow increase up to and including the rated power condition. However, an inadvertent flow increase in this mode will result in substantial margin to the safety limit MCPR.

The development of the generic K_f curves in the fuel reload licensing topical report employed a generic power-flow curve. The resulting critical power-flow relation is dominated by the GEXL correlation, not by plant size, power density, or slight variations in core inlet enthalpy and pressure.

The peak vessel dome pressure of 1180 psia has been changed to 1080 psia, and the ordinate values of l.0, 1.1 and 1.2 for the peak surface heat flux (% of initial) have been replaced with the values 100.0, 110.0 and 120.0, respectively.

The generic K_f curves were developed on the basis that the flow increase event was the most limiting at reduced flow. This basis was justified by studies performed on BWR/3 and 4 FSAR data (7x7 fuel). Specifically, inadvertent startup of an idle recirculation pump and feedwater flow controller failure (maximum demand) events were analyzed at reduced power and flow as summarized in Table 5.2-2, demonstrating the adequacy of K_f.

The most recent evaluations in this area were performed on a standard 764 bundle BWR/5 core with retrofit fuel, simulating equilibrium core conditions. The operating limit MCPR for this core was 1.27, and the safety limit MCPR was 1.07. The results of inadvertent cold loop startup and feedwater flow controller failure (increase) transients initiated from various power/flow states are presented in Table 5.2-3. In each case, the initial CPR was determined such that the minimum, CPR during the transient was l.07. Thus, as long as the product of K_f and the operating limit MCPR is greater than the initial CPR necessary to satisfy the safety limit MCPR, the K_f curves are conservative. This comparison is provided for the 102.5% maximum core flow K_f curve in Table 5.2 3 and demonstrates conservatism at all flows except 75% flow where this K_f curve is only slightly nonconservltive by 0.005.

The results of these analyses (Table 5.2 3) are considered generic because the data used bound the BWR/2 4 projected equilibrium conditions. While there may be variations between plants in their reduced power and flow transient response, a similar variation would also exist for the limiting rated power and flow transient, creating a sufficiently high operating limit MCPR that, when combined with the K_f curves, an appropriately high reduced flow operating limit is established.

The curve of power level decay versus exposure is determined by calculating the exposure capability at selected power levels with the BWR simulator code.

With respect to the pressurization transients discussed in Section 5.2, the primary reason for the decrease in transient severity during coastdown is the dynamic void coefficient, which becomes less negative as the power level is reduced. This is an inherent characteristic of the BWR, which does not change with the introduction of the 8x8 retrofit reload lattice.

REFERENCES:

1. Letter, G.L. Gorey (GE) to W.R. Butler (NRC), "1-Dimensional Methods for Calculating Scram Reactivity," 12 March 1976.

5.2-1 MCPR

Power	Flow	8x8R	8x8	7x7
101.8%	102.5%	1.060	1.060	1.060
100	100	1.075	1.075	1.073
93	90	1.129	1.131	1.128
86	80	1.182	1.186	1.184
78.9	70	1.234	1.241	1.242
71.5	60	1.287	1.298	1.304
63.9	50	1.339	1.353	1.365
56.0	40	1.389	1.407	1.428

5.2-2 Evaluation of BWR 3 & 4 Abnormal Operational Transients Initiated from Lower Power States SAR Data (7x7 Fuel)

Event	Power	Flow	Initial CPR	ΔCPR	Tech Spec K_f	Required K_f*
Idle Recirculation Loop Startup	60%	45%	1.64	0.08	1.13	1.0
Feedwater Control Failure (Max Demand)	65%	51.5%	1.60	0.06	1.105	1.0

*The required K_f is derived such that the MCPR safety limit is satisfied assuming an operating limit MCPR of 1.20.

5.2-3 Evaluation of BWR/5 Abnormal Operational Transients Initiated from Lower Power States (8x8R Fuel)

Event	Power	Flow	ΔCPR	Initial CPR[1]	K_f* 1.27 [2]
Feedwater Controller Failure (Increase)	105	100	0.179	1.249	1.270
Feedwater Controller Failure (increase)	100	91	0.185	1.255	1.270
Feedwater Controller Failure (increase)	97	85	0.199	1.269	1.270
Feedwater Controller Failure (increase)	91	75	0.205	1.275	1.270
Feedwater Controller Failure (increase)	85	61	0.223	1.293	1.353
Idle Recirculation Loop Startup	63.9	49.7	0.152	1.222	1.412
Idle Recirculation Loop Startup	56.2	40	0.361	1.431	1.463
Idle Recirculation Loop Startup	63	36	0.207	1.277	1.483

1 The initial CPR was determined by iteration on the bundle power such that the minimum CPR during the transient was 1.07.

2 K_f based on maximum core flow of 102.5%. The operating limit MCPR is 1.27.

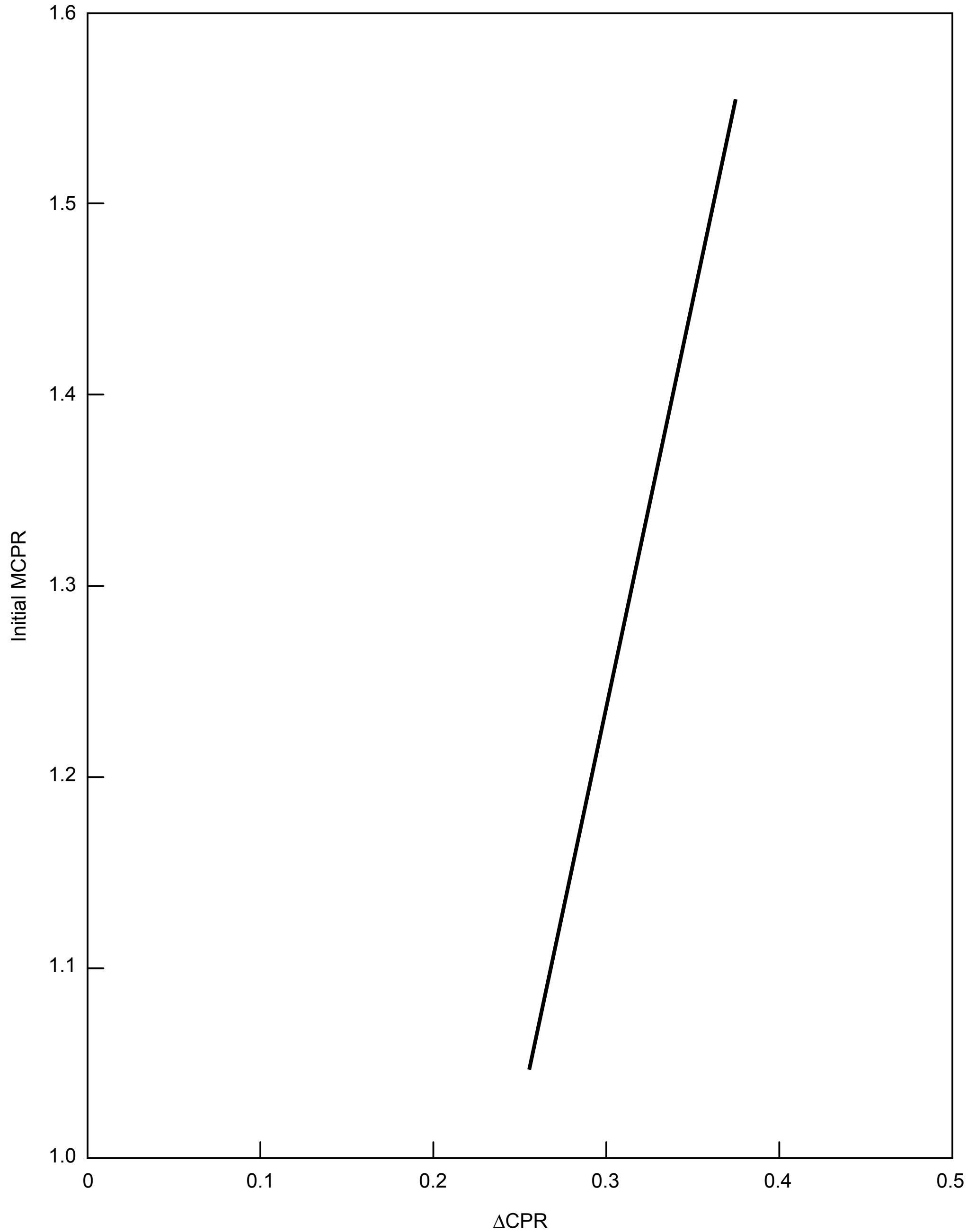

Figure 5.2-1. Effects of Initial MCPR on ΔCPR Typical BWR Core

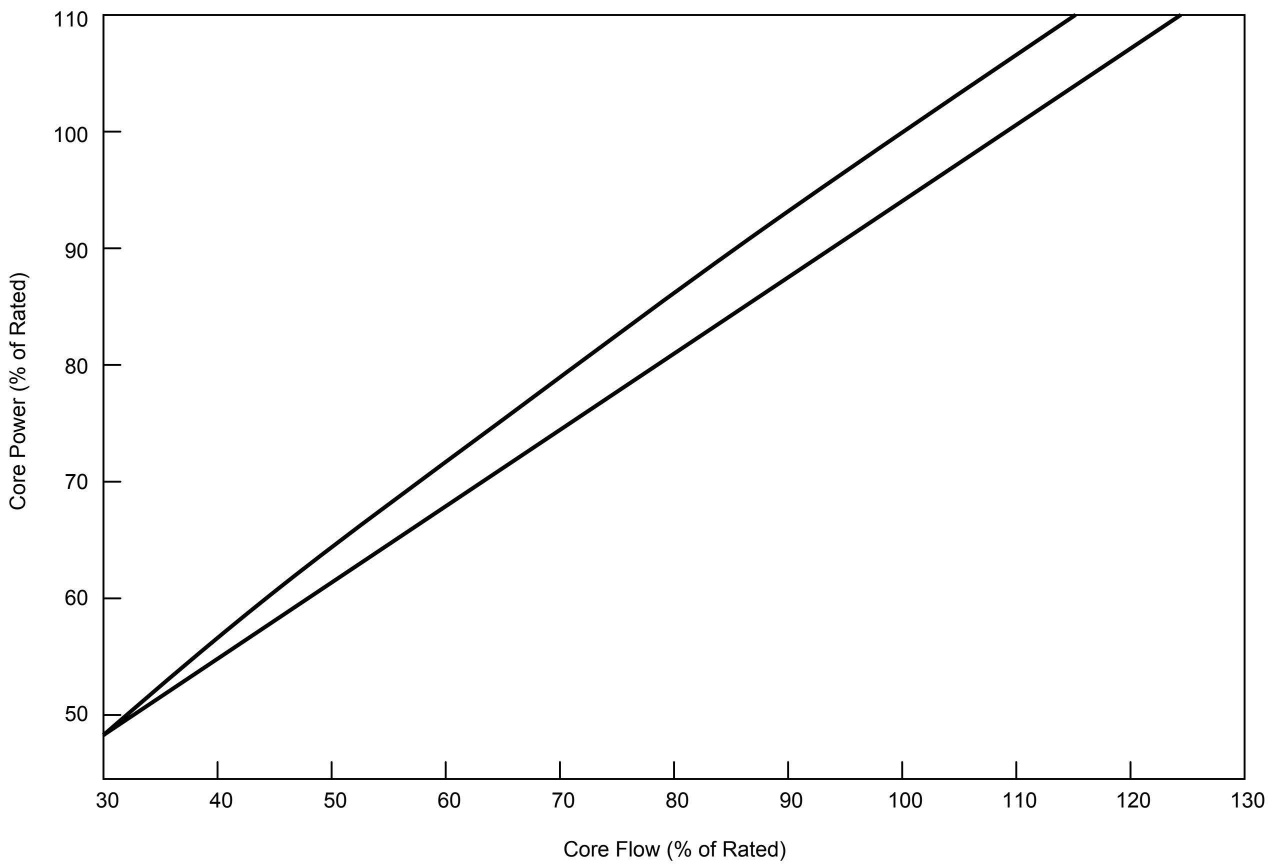

Figure 5.2-2. BWR/2–4 Power-Flow Map

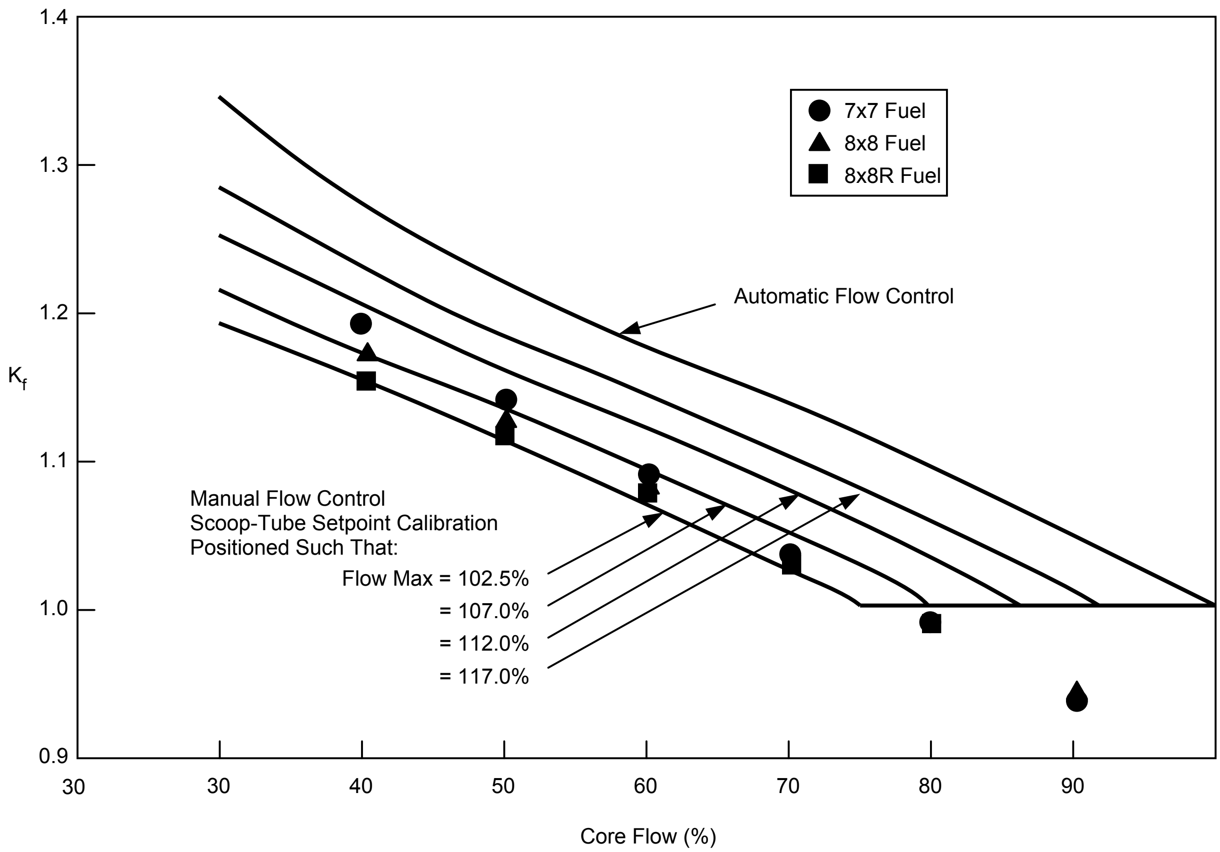

Figure 5.2-3. K_f Factor

REQUEST: Subsection 5.2.1 – Transient Descriptions

The CPR sensitivity study results presented in Section 5.2.1, as a justification for eliminating most of the FSAR transients from reload reanalysis, is not acceptable. It does not address the 8x8R fuel design. Furthermore, the CPR sensitivity, indicated by the GEXL correlation, is only one component of the total methodology used for determining the ΔCPR for an event. The magnitude of the changes in the various thermal-hydraulic parameters (i.e., REDY code output) during an anticipated transient is just as important. Therefore, provide the results of the SCAT code transient analyses, which show the change in critical power ratio for the most limiting event in <u>each</u> of the eight transient parameter (variation) categories on page 5-10. The analyses should model a representative BWR plant. Provide justification for the plant modeling, reactivity coefficients and scram curves selected. Compare these results to the results of events which are proposed for analysis on plant specific reload applications (e.g., TT w/o BP, LFWH, HPCI startup, FWCF, RWE). The results should be presented for 7x7, 8x8 and 8x8R fuel types in a representative mixed core loading situation. Provide an assessment of the change, from cycle-to-cycle, in relative and absolute severity of the events analyzed above due to exposure effects. Provide conclusions relating to the adequacy of the limited group of events selected for reload reanalysis for all future fuel cycles.

For relatively slow transients (e.g., loss of feedwater heating), the assumption that the local (thermal) power distribution remains constant (and hence the R-factor remains constant) may not be valid. Provide an evaluation of the validity or conservatism of this assumption for relatively slow events. Evaluate this assumption for each of the various fuel designs (7x7, 8x8, 8x8R).

The following comments apply to all of the transient description subsections:

1. The first paragraph should be labeled as “Identification of Causes.” This paragraph should appear for all events considered. Noteworthy examples should be included.

2. The second “Starting Conditions and Assumptions” paragraph should also indicate:
   (a) The percent of the licensed power level/steam flow which includes an allowance for the thermal power level uncertainty (i.e., ≥2%). An allowance is required for all transient analyses except for the 25 psi margin to safety valve setpoint analyses (as specified in Standard Review Plan Sections 5.22 and 15).
   (b) The initial state of the plant process variables.
   (c) The operation of the reactor protection systems (i.e., relief valves, scram). (See “Standard Format and Content of Safety Analysis Reports.”)

3. The extent and depth of information provided in the “Event Description” paragraph is an inadequate qualitative description of the behavior of the major plant process variables (e.g., pressure, steam flow) and core variables (neutron flux, heat flux, voids) as well as the responses of the important reactor system equipment during the event. The time variations in the variables discussed should be qualitatively related to the time behavior of other key parameters and system equipment. The discussion should be representative of a typical plant. Any differences in the general transient behavior of the process variables due to plant-to-plant differences in equipment design (e.g., transient feedwater flow due to FW pump power source, ATWS RPT) should be noted on a plant-by-plant basis.

4. Provide a “Results” paragraph which states which results will be presented for each of the transients analyzed (e.g., pressure flux, steam flow histories, peak vessel pressure, peak kW/ft ΔCPR). State the section in the document which discusses the appropriate limits for these parameters. Where interpretation is the results is not obvious (e.g., RWE), provide a brief explanation. Provide a brief explanation of the REDY code parameters plotted as part of core-wide analyses (e.g., “WR Sensed Level”).

Some plants reanalyze the Feedwater Control Failure-Increasing Flow for reloads. Furthermore, this event may be limiting for many plants in the event that transient (TT w/o BP) recategorization is approved. Therefore, provide a subsection which describes the Feedwater Control Failure—Increasing Flow event. Include in the section all the paragraph headings addressed in the other subsections of Subsection 5.2.1.

It is assumed that <u>all</u> of the transient events described in Subsection 5.2.1 will be analyzed by all of the plants listed in Table 1-l for the supplemental reload licensing submittal. If this is not the case, a justification for eliminating one or more events from the reload analysis must be provided. A summary table, showing the plants, with the moderate frequency events analyzed, should be included in the Topical if appropriate.

RESPONSE:

The entire spectrum of transients covered in each of the eight transient parameter (variation) categories is given in the individual plant FSAR's for typical operating plants. A review of these transient results (e.g., pressure, power, flow) can be used to determine which categories and which transients in each category have the potential for being limiting. From this it is clearly established that the limiting transients will always be in the four groups of transients identified on page 5-11. These are the transients which involve significant effects on heat flux and reactor vessel pressure peaks. The fact that there are differences between the transient results when GETAB is used rather than the previous thermal analysis basis is recognized.

Other transients already analyzed in FSAR's have relatively less severe effects on heat flux performance and transient pressure peaks. However, the differences due to the GETAB analysis are not significant enough to warrant reanalysis. The reasons for this become obvious when the results of the study of the relative dependence of CPR upon various thermal-hydraulic parameters presented in Subsection 5.2.1 are reviewed. As stated, CPR is most sensitive to R-factor and bundle power. Because R-factor is a function of bundle geometry and local power distributions and does not change during a transient, CPR becomes primarily a function of changes in power. Further verification of the results can be found in GESSAR.

Therefore, General Electric has determined that the effect on CPR, as caused by the four groups of power increase transients, is so much more significant than the effects of any of all the other known transient possibilities, that reanalysis of the remainder of the transients is not warranted. However, to provide complete assurance that the most limiting transients are always identified and analyzed for each reload, all of the transients identified in Subsection 5.2.1 are analyzed for each plant (except inadvertent HPCI startup only for plants with HPCI). The Supplemental Reload Submittal will report the results of only the most limiting transient in each of the four groups.

The sensitivity of the plant response to transients due to exposure has been demonstrated by reload analyses for some operating plants for as many as eight operating cycles. Analyses of plants which have been operating for so many years show that, although the overall severity of transients is affected by degradation in scram until equilibrium is reached, the spectrum of limiting transients does not basically change. It has been found that when end-of-cycle conditions are significantly more severe than mid-cycle conditions, the operating limit determining transients may change. For example, if at end-of-cycle, the turbine trip without bypass is limiting, it may be found in a mid-cycle analysis that the TT w/o BP is so much less severe that the rod withdrawal error or loss of feedwater heating transient may be limiting. These predictable changes due to exposure are caused primarily by the sensitivity of the negative scram reactivity characteristic to exposure up to equilibrium cycle. The introduction of 8x8R fuel is not expected to change this since, as shown above, the plant response is basically no different than with 8x8 fuel.

The increase in inlet subcooling due to the loss of feedwater heating causes the core power to increase and moves the boiling boundary higher in the core. The result is that the axial power will shift lower in the core (~6 inches). This will have a slight affect on the local peaking distribution at the new boiling boundary. The effect on R-factor is minor because the bundle R-factor is the axial averaged rod R-factor (see responses to NRC questions on fuel assembly loading error in Reference 1), and this 6-inch change in the peak power represents only l/24 of the total fuel length.

In analyzing this type of transient, the licensing basis assumes that the axial power shape is held constant (peaked at the middle of the fuel assembly). As noted above, these events result in an axial power shift lower in the bundle, which in the actual case will cause a smaller ΔCPR because bottom peaked power shapes yield higher critical power ratios than middle peaked, as was demonstrated in Section V of Reference 5-l. That is, the licensing basis calculation is conservative because credit is not taken for the shift in power shape.

Transients are precipitated by a single operator error or equipment malfunction. The initiating event for each transient has been included in either the first paragraph or event description portion of each transient description in Subsection 5.2.1. It should also be noted that this report was not written to conform to the Standard review plans which apply only for plants seeking construction and operating permits. Response to the recommendations for the "Starting Conditions and Assumptions" paragraph are as follows:

1. The percent of licensed power level/steam flow used in individual plant analyses is identified in the Supplemental Reload Submittal for each limiting transient. The 2% adder for power level uncertainty is not included as discussed in Reference 2. This letter indicates that the licensing basis, as defined in the bases statement on fuel cladding integrity in typical plant technical specifications, describes the conservatisms incorporated in transient analyses. Therefore, application of a 2% adder for uncertainties in power level is considered unnecessary.

2. The initial state of the plant process variables is identified in Tables 5-4, 5-6 and new Table 5-8. In addition, cycle-dependent data is provided in each Supplemental Reload Submittal.

3. The operation of the affected reactor protection systems is discussed in each transient description provided in Subsection 5.2.1. The individual plant FSAR provides a completely detailed, comprehensive description of the operation of the reactor protection system.

The intent of the "Event Description" paragraph is to provide a concise summary of events which assumes some prior familiarity with BWR plants. For a comprehensive discussion of the behavior of major process variables, core variables or the responses of reactor system equipment, see the individual plant FSAR and Reference 5-2 of the report.

All the "results" information requested are described in Appendix A, which shows the format and content of the Supplemental Reload Submittal. Concern for nonobvious interpretations is reduced by the new Appendix A format in Revision 1. An explanation of the REDY code parameters is provided in NEDO 10802.

The feedwater controller failure (maximum demand) event mentioned in this question was erroneously omitted in the original document. In description of the pressure regulator failure (open) event was included in its place. This error was corrected in Amendment 1.

The transient events on page 5-11 are analyzed each cycle for all plants with appropriate omissions for plants without HPCI. It should be noted, however, that only the limiting transient for each group is reported in the Supplemental Reload Submittal. In general, for plants with the mechanical hydraulic control (MHC) feature in their turbines, the limiting transient will usually be turbine trip without bypass. Conversely, if the turbines are equipped with electrical hydraulic control (EHC), the limiting transient will usually be generator load rejection without bypass. The loss of feedwater heating transient is usually more severe than inadvertent HPCI startup.

REFERENCES:

1. Letter, R.E. Engel (GE) to D.G. Eisenhut (NRC), “Fuel Assembly Loading Error,” November 30, 1977.

2. Letter, R.L. Gridley (GE) to D.G. Eisenhut (NRC), “MSIV Closure with Flux Scram-Sensitivity of Peak Vessel Pressure to Initial Power Level”, September 12, 1977.

REQUEST: Subsection 5.2.1.1 – Generator Load Rejection Without Bypass

The “Starting Condition and Assumption” should include the assumed performance of the control, and relief valves. The closure time of the Turbine Controls valves should also be included in this paragraph. “Event Description” must be improved per item (c) above.

RESPONSE:

After the steamline pressure reaches the setpoint, the turbine control valves are assumed to operate in a fast closure mode, and the relief valves will open at the rate and delay time shown in Table 5-4. The closure time appears in the “Event Description” paragraph.

REQUEST: Subsection 5.2.1.2 – Turbine Trip Without Bypass

The “Starting Conditions and Assumptions” comments are the same as for 5.2.1.1.

RESPONSE:

The description in the “Starting Conditions and Assumptions” paragraph for this transient refers to the generator load rejection because the “Starting Conditions and Assumptions” for both transients are identical. Refer to the response to the request on Subsection 5.2.1.1.

REQUEST: Subsection 5.2.1.3 – Loss of Feedwater Heating

Provide a list of plants with an assumed 100°F LFWH capability and those with a documented 80°F LFWH capability. The “Event Description” paragraph must be improved per item (c) above.

RESPONSE:

	LFWH Capability (°F)
BWR/2	
Nine Mile Point 1	100
BWR/3	
Monticello	100
Millstone	100
Pilgrim	100
Quad Cities 1	145
Quad Cities 2	145
Dresden 2	145
Dresden 3	145
BWR/4	
Vermont Yankee	100
Duane Arnold	100
Cooper	100
Fitzpatrick	80
Hatch 1	100
Brunswick 1	100
Brunswick 2	100
Peach Bottom 2	100
Peach Bottom 3	100
Browns Ferry 1	100
Browns Ferry 2	100
Browns Ferry 3	100

REQUEST: Subsection 5.2.1.4 – Inadvertent Start of HPCI Pump

Provide an "Identification of Cause" paragraph. Provide a table which shows which plants have HPCI and which do not. The "Event Description" paragraph must be improved per item (c) above.

RESPONSE:

	HPCI (Yes/No)
BWR/2	
Nine Mile Point 1	No
BWR/3	
Monticello	Yes
Millstone	No
Pilgrim	Yes
Quad Cities 1	Yes
Quad Cities 2	Yes
Dresden 2	Yes
Dresden 3	Yes
BWR/4	
Vermont Yankee	Yes
Duane Arnold	Yes
Cooper	Yes
Fitzpatrick	Yes
Hatch 1	Yes
Brunswick 1	Yes
Brunswick 2	Yes
Peach Bottom 2	Yes
Peach Bottom 3	Yes
Browns Ferry 1	Yes
Browns Ferry 2	Yes
Browns Ferry 3	Yes

REQUEST: Subsection 5.2.1.5 – Rod Withdrawal Error

I. Provide an "Identification of Cause" paragraph. Discuss the basis for the initial power level assumption. Provide a discussion of the allowances made for various combinations of failed LPRM strings in the RWE analysis. Provide a discussion of the allowance made for the response loss in the local detectors due to excessive voiding in the bypass region. Provide a discussion of the plots to be presented on reloads and their interpretation. Discuss the LHGR limits for each type (and rod type). To what fuel design is the 17.5 kW/ft thermal design limit refer? Provide a statement that the resulting ΔkW/ft (above the respective design values) or peak kW/ft will be provided in the reload supplement for each fuel type.

For calculation of the rod withdrawal error using the 3D BWR Simulator:

1. How many of the 24 mesh point divisions which are available for use are actually used?

2. Is a full core calculation performed for an off-center rod or are quarter-or half-core symmetries sometimes assumed?

The multiplier to the APRM and RBM rod withdrawal block setpoints adjusts the setpoint downward in the event of plant operation with the operating maximum total peaking factor, MTPF, (for any fuel type) greater than the (respective fuel type) design maximum total peaking factor. The multiplier is of the form:

$$\text{Multiplier} = \text{Min}\left(\frac{\text{Design MTPF}}{\text{Operating MTPF}}\right),\ \text{i} = 1,\text{n (n = Fuel type)}$$

Provide a discussion in this section of the purpose of the multiplier. Provide a workable definition of the design Maximum Total Peaking Factor which can be used for a core reload configuration with up to three fuel types (7x7, 8x8, 8x8R) and is also consistent with the process computer programming. Discuss the algorithm used by the process computer to calculate the operating MTPF for each fuel type. Show that the definition is consistent with process computer function. Based on the design MTPF definition given, what adjustment to a proposed Technical Specification design MTPF will be required if the actual core loading (number of assemblies of each fuel type) differs from the reference core loading. Discuss the significance of cycle-to-cycle changes in the calculated design MTPF's as 8x8R reload assemblies replace assemblies of other fuel designs (8x8, 7x7). Show that these changes to the design MTPF values are consistent with the process computer program algorithm. Show that the MTPF definition and resulting multiplier give an adequate reduction in the RBM setpoints in view of the assumed initial (design) kW/ft of each type, the calculated ΔkW/ft for each fuel type as well as the kW/ft (corresponding to 1% strain) for each fuel type, for the most severe RWE.

Provide a similar discussion of the adequacy of this multiplier and the MTPF definition in connection with the adjustment to the High Flux Scram setpoint. Provide a statement that the required changes to these design MTPF's will be provided in the reload supplement.

II. The Control Rod Withdrawal Error (RWE) is a localized transient which can have a significant axial R-Factor variation when the rod block occurs to terminate the event. For a representative limiting RWE transient case provide the following information:

(a) R-Factor vs. axial location for 7x7, 8x8 and 8x8R fuel assembly types in a D-Lattice core for rod blocks occurring at 3.0, 5.0, 7.0 and 9.0 ft withdrawn.

(b) Axially average R-Factor, the minimum and maximum R-Factor for each fuel type and ft withdrawn case given in (a) above.

(c) For typical rated steady-state operating conditions using available test data, discuss the magnitude and trend in the GEXL correlation basis for rod blocks between 9.0 ft and 3.0 ft withdrawn in a C-Lattice core.

(d) Discuss the expected effects for a D-Lattice core relative to a C-Lattice core in connection with question (c) above. Address the R-Factor axial variation for the two lattice types for 8x8R fuel.

(e) Discuss any conservatisms not previously taken credit for in the GETAB methods which could serve to offset the under prediction of critical bundle power for the range of axial variation in R-Factor.

(f) Provide a table of the experimental data points and the corresponding GEXL predictions for the data given in Figure 9-10 of Reference 1.

RESPONSE:

I. A rod withdrawal error occurs when a reactor operator makes a procedural error and withdraws the maximum worth rod to its fully withdrawn position. This information was presented in the original submittal in the event description.

Rod Withdrawal Error calculations have been performed for a statistically significant number of normal operating conditions. These studies show that the effects of a rod withdrawal error are greatest at rated power and flow conditions. These studies also show that there is a lower bound for which a rod withdrawal error will not violate the safety limits. However, to provide conservatism in the analysis, all Rod Withdrawal Error analyses are assumed to start with an “on-limits” core at rated power and flow.

The RWE analysis is performed for 10 failure conditions representing all combinations of none, one and two failed strings. A composite response curve is formed which is the lowest response of each of the 10 failure conditions at each of rod withdrawal data points. The composite curve is used to determine the RBM setpoint. This insures that the worst responding failure is used throughout the rod withdrawal.

Under normal conditions, excessive bypass voiding does not occur. For plants which had plugged core plates, an additional analysis was performed to determine the response loss due to bypass voiding. This response loss was subtracted from the composite response curve before determining the RBM setpoint.

The standard tabulated data are provided in Section 10 of the Plant Supplemental Submittal, “Local Rod Withdrawal Error (With Limiting Instrument Failure) Transient Summary”. The Rod Block Monitor Reading at each rod position is presented, as is the change in CPR and the LHGR for each fuel type. The LHGR limits for each fuel type are given in the “Results and Consequences” as 13.4 kW/ft for the 8x8 fuel and 17.5/18.5 kW/ft for the 7x7 fuel. All 7x7 fuel operating in BWR/2,3 reactors listed in Table 1-1 have a MLHGR of 17.5 kW/ft and all 7x7 fuel operating in BWR/4 reactors listed in Table 1-l have a MLHGR of 18.5 kW/ft.

A statement has been added to the results and consequences paragraph to indicate that the maximum LHGR during this event will be reported in the Plant Supplemental Submittal.

A minimum of 24 axial mesh points are used for the neutronic and thermal hydraulics calculation. The LPRM readings used as input to the RBM channels are determined from the fluxes in the mesh points adjacent to the chamber. Therefore, eight are used for this part of the calculation. Full core calculations are performed for off-axis rods.

The purpose of the APRM flow-biased scram is to prevent fuel damage due to an abnormal operating transient from any point on the power flow map. The purpose of the multiplier is best illustrated by considering a transient from two different operating states. Assume a plant operating at rated power and flow with at least one fuel rod operating at its rated linear heat generation rate. By definition, the operating maximum total peaking factor is equal to the design maximum total peaking factor. If an abnormal operating transient with APRM flow-biased scram trip occurs from this state, the trip signal is generated when the core average neutron flux reaches 120% of rated and terminates the transient before the limiting fuel rod reaches its transient limit LHGR. Now, assume the same reactor is operating at 90% of rated thermal power, 100% of rated flow, and with at least one fuel rod operating at its rated linear heat generation rate. By definition, the operating maximum total peaking factor is equal to the design total peaking factor divided by 0.9. This requires that the APRM flow-biased scram trip be reduced from 120% to 108% of core average rated neutron flux. If the abnormal operating transient occurs from this state, the scram trip is again generated after a 20% increase in core average neutron flux and the margin to the transient limit LHGR is maintained. The multiplier ensures that a constant margin is maintained for all power distributions at constant flow.

The definition of the design maximum total peaking factor which is fuel type and plant dependent is:

$$\text{MTPF} = \frac{\text{MLHGR} \times \text{NBUN} \times \text{NRODS} \times \text{LF}}{\text{RP} \times \text{CHFF}}$$

where

MLHG	=	design linear heat generation rate limit (kW/ft);
NBUN	=	number of fuel bundles of all types in core;
NROD	=	number of active fuel rods per bundles of fuel being considered;
LF	=	active fuel length (ft);
RP	=	rated thermal power (kW); and
CHFF	=	cladding heat flux fraction.

The process computer is unaffected by the reload fuel design. Since the design total peaking factor is fuel type unique, it is unaffected by changes in loading pattern. Cycle-to-cycle changes require only the addition of data for new fuel types and deletion of data for types no longer in the core.

The APRM flow-biased rod block guards against exceeding the APRM scram trip point rather than guarding against the worst-case RWE. This is a function of the Rod Block Monitor, which takes its input signal from the LPRM strings surrounding the rod being withdrawn. Since the trip signals are taken from the local area where the error is occurring, no peaking factor multiplier is required.

The new design MTPF’s are provided in the application for changes to the Technical Specifications. However, General Electric is currently recommending a Technical Specification change which will define the APRM flow-biased scram and rod block setpoints in terms other than peaking factors. This will eliminate the need to redefine the peaking factor limit with every fuel change.

General Electric is recommending that the factor A/MTPF, which is used to adjust the APRM flow-biased trip settings, be replaced by F/MFLHGR. A is the design total peaking factor and may be different for each bundle type, MTPF is the maximum total peaking factor, F is the fraction of rated thermal power, and MFLHGR is the maximum fraction of limiting linear heat generation rate. The two expressions are equivalent, but the former must be evaluated and checked for every bundle type; the latter requires only one calculation for the most limiting point in the core.

II. Response to this question was given in Reference 1.

REFERENCES:

1. Letter, Ronald E. Engel to Paul S. Check, "NEDE-24011-P-A: Axially Varying R-Factors", May 15, 1979.

REQUEST: Subsection 5.2.2 – Margin Improvement Options

(NOTE: The following requested *Margin Improvement Options* information does not represent a complete set of staff questions and concerns related to this subject, since their generic evaluation (except for Exposure Dependent Limits) is not within the scope of review of this Topical. Accordingly, acceptable responses to these questions should not be interpreted as acceptance, by the staff, of any or all of these options.)

Single recirculation loop operation during power operation is not yet approved for operating BWR's. When approved, it will have an impact on the reload reanalyses. If NEDE-24011-P is to provide a reference for the acceptability of reload cores, as they are affected by single loop operation, then a discussion of the effect of single-loop operation on normal operation, abnormal operational transients and accidents must be addressed. Include a discussion of the single-loop operation MAPLHGR reduction factor and the basis for the factor. Discuss the procedure used for correcting the ARPM and RBM flow-biased rod block settings and flow-biased scram setting, to account for backflow during single-loop operation. Reference should be made to other generic or lead plant evaluations where appropriate. Discuss single-loop operation as it relates to the reload stability analyses. Discuss the effect of single-loop operation on the total core flow relative to active recirculation flow and the flow uncertainty. Relate these effects to the generic calculations of the fuel cladding integrity safety limit MCPR.

RESPONSE:

Single-loop operation is not a MCPR margin improvement option. Although the reactor is assumed to be operated at some reduced flow and power (on or below the rated flow control line) during single-loop operation, the operating limit MCPR corresponding to the core flow is no different during single-loop operation than it is during two-loop operation. Change in safety limit MCPR is negligible (see Reference 1).

Single-loop operation will not have an effect on reload analysis and is therefore not addressed in the generic reload fuel application licensing topical report. Single-loop operation is addressed in licensing amendment submittals for individual BWR operating plants. For example, see the lead plant licensing submittal in Reference 1.

REFERENCES:

1. "Pilgrim Nuclear Power Station Unit 1 License Amendment for Single-Loop Operation with Bypass Holes Plugged," NEDO-20929, January 1976.

REQUEST: Subsection 5.2.2.4 – Thermal Power Monitor

Provide the functional form of the TPM Transfer Function. Discuss the manner in which the time constants and any other parameters entering into the formulation are determined for a given plant. Explain the physical significance of all terms and/or factors in the transfer function. Discuss the performance (response of the TPM) for normal and abnormal events. Provide the calculated flux trip level vs. flux frequency (e.g., Turbine Trip w/o BP to LFWH). Discuss the influence (if any) of reloads on the values of the parameters in the TPS transfer function.

Provide examples of sources of "spurious" and "momentary flux spikes" referred to in the second paragraph, which the system is designed to filter out. Give the frequency of these events.

RESPONSE:

The thermal power monitor (TPM) transfer function is that of a low pass RC filter, where R is resistance and C is capacitance. Resistance and capacitance values were selected which result in a 6-sec (±l sec) time constant. This RC filter is shown on APRM scram trip electrical diagrams for plant-specific installations. The RC filter time constant of 6 sec envelopes the fuel time constants for 7x7, 8x8 and 8x8R fuel pins, which are in the range of approximately 7 to 10 sec.

The performance of the APRM simulated thermal power trip (thermal power monitor) is discussed in a revision to Subsection 5.2.2,4. Examples of events which could cause spurious scrams due to momentary flux spikes are also given in this revision. The frequency of these events may be found in the operations logs of the BWR operating plants.

From the above information, it is seen that the operation of the TPM is independent of the fuel loading in the core. Also, no normal, abnormal, or accident events, other than the loss-of-feedwater heater transient, are affected by the operation of the TPM.

REQUEST: Subsection 5.2.2.5 – Exposure Dependent Limits

It is understood that most of the initial conditions and other input parameters for the GETAB transient analysis are exposure dependent, while others are conservatively assumed to be at their most adverse values during the cycle. Indicate which parameters, if any, are taken at their most limiting values during the cycle. Discuss where these assumptions may vary depending on the transient event or plant considered. Include the fuel loading error.

The licensing calculations for the n+l cycle are performed significantly before the end of cycle n. This results in the necessity of estimating the cycle (n) burnup in order to calculate the EOC-(n+l) exposure increment. Thus, the actual n+l cycle core incremental exposure at which the all-rods-out end-of-cycle conditions is attained will, in general, be different from the predicted exposure increment. Furthermore, model uncertainties will also contribute to a difference between the actual and the predicted EOC exposure for cycle n+l. Since the exposure-dependent limits are referenced from EOC, which is not precisely determined until it is actually achieved, provide, in sufficient detail, the manner in which the exposure-dependent operating limits will be conservatively implemented during plant operations. Discuss the procedures to be used by the reactor operator to make exposure corrections to determine the actual EOC exposure, in order to establish when in the cycle the operating MCPR limits should change.

RESPONSE:

The GETAB transient analysis initial condition parameters that are taken at their most limiting values during the reload cycle are the nonfuel power fraction, the core flow, the reactor pressure, and the coolant inlet enthalpy. The limiting values of these parameters are the same for all transient events analyzed, but are plant specific, as indicated in Table 5-6 (page 5-65).

The fuel loading error analysis uses the same GETAB transient analysis initial condition parameters that are used in the GETAB analysis of abnormal operational transients.

At end of cycle n, the details of the actual end-of-cycle exposure distribution are known. The exact BOC n+l core configuration is then input to the three-dimensional BWR simulator mode. Using expected full-power conditions for cycle n+l, the cycle n+l exposure capability, E_{n+l} is calculated by Haling depletion from this model of the actual BOC n+l core.

Early in cycle n+l, the calculated exposure capability, E_{n+l} is documented and is sent to the reactor operator. If a change in operating limits is scheduled to occur at an exposure of EOC-x, the reactor operator implements that change when cycle n+l core average exposure accumulation reaches E_{n+l}—x, as monitored by the process computer.

The exposure-dependent operating limits are conservative, as implemented, for two reasons: (1) the operating limits are determined by conservative analytical procedures, and (2) the method of implementation is conservative. Each of these statements will be expanded below.

When analyzing transient performance at exposures prior to the end-of-cycle (EOC), all-rods-out condition, it is necessary to consider the effect of the control rods on the transient parameters, because the scram reactivity and the dynamic void coefficient are sensitive to the control rod pattern. At any given exposure point, there are many control rod patterns which will render the core critical and within thermal limits. To ensure that conservative values of the important dynamic parameters are calculated, it is necessary to select special control patterns. Conservative values of both the scram reactivity and dynamic void coefficient result when “black-white” control patterns are used. A black-white control pattern is one in which control rods are either fully inserted (black), or fully withdrawn (white).

The scram reactivity is minimized with black-white patterns because:

(1) the fully inserted control rods provide no contribution to the scram reactivity,

(2) the fully withdrawn control rods begin their insertion in a region of zero power; thus, their impact during the early portion of the scram is minimized; and

(3) there are no partially inserted control rods, which generally provide a major contribution during the early portion of the scram.

The magnitude of the dynamic void coefficient is maximized because the core average axial power distribution is shifted toward the bottom of the core with black-white control rod patterns, and as a result the core average void fraction is increased. Since the dynamic void coefficient is directly proportional to the core average void fraction, its magnitude is therefore increased.

The black-white rod patterns are always used to generate the dynamic parameters at midcycle exposure points, unless it is not possible to meet thermal limits with these patterns. If thermal limits cannot be met (usually due to high axial peaking in the bottom of the core, which exceeds the Technical Specification limit on linear heat generation rate), the black-white patterns are modified because the operating reactor would be prevented from operating with such patterns by means of its Technical Specification limits. Modifications to the black-white pattern are made by slight withdrawal of fully inserted rods and shallow insertion of withdrawn rods. Thus, a critical rod pattern which meets thermal limits is attained while minimizing the deviation from the black-white pattern.

The conservative nature of the limits, as implemented, can be shown most effectively by means of an illustration. In Figure 5.2.2.5-1, the MCPR at two intermediate exposures and at EOC have been plotted as a function of exposure. The solid circles represent the results derived from the black-white rod patterns, and the open circles represent limits derived from the nominally expected rod patterns. The MCPR limit varies smoothly between these points, as shown in the figure.

The Technical Specification limit on MCPR, however, is applied as a histogram, rather than as a smoothly varying function of exposure. The MCPR limit during any exposure interval is determined by the most limiting value at either end of that interval. For the example shown in Figure 5.2.2.5-1, the Technical Specification limit would be applied as shown by the heavy horizontal lines.

Now, if for some reason the projected EOC exposure differs from that which is actually attained, either of the following would result:

(a) The actual EOC exposure is greater than that projected. In this case, the step increases in MCPR limits shown in Figure 5.2.2.5-1 would occur sooner than actually required, and there is no potential violation of limits.

(b) The actual EOC exposure is less than that protected. In this case, the step increases in MCPR would occur later than required, and there is a potential violation of limits. This situation is illustrated in Figure 5.2.2.5-2. Note that there are small exposure intervals just prior to the step changes in limits for which the Technical Specification limit is slightly less conservative than that derived from the black-white patterns. However, the difference between these two MCPR limits is small. Also, note that the Technical Specification limit provides margin to the limit derived from the nominally expected rod patterns for reasonable deviations for end-of-cycle exposure.

Therefore, unless the projected and actual EOC exposures differ by a large amount, and differ in the non-conservative direction, there would be a negligible nonconservative difference between the Tech Spec limit and that required by the black-white rod patterns. Exposure differences of this magnitude are unlikely since the projected value used to determine MCPR limits is based upon the as-loaded core for the cycle of interest.

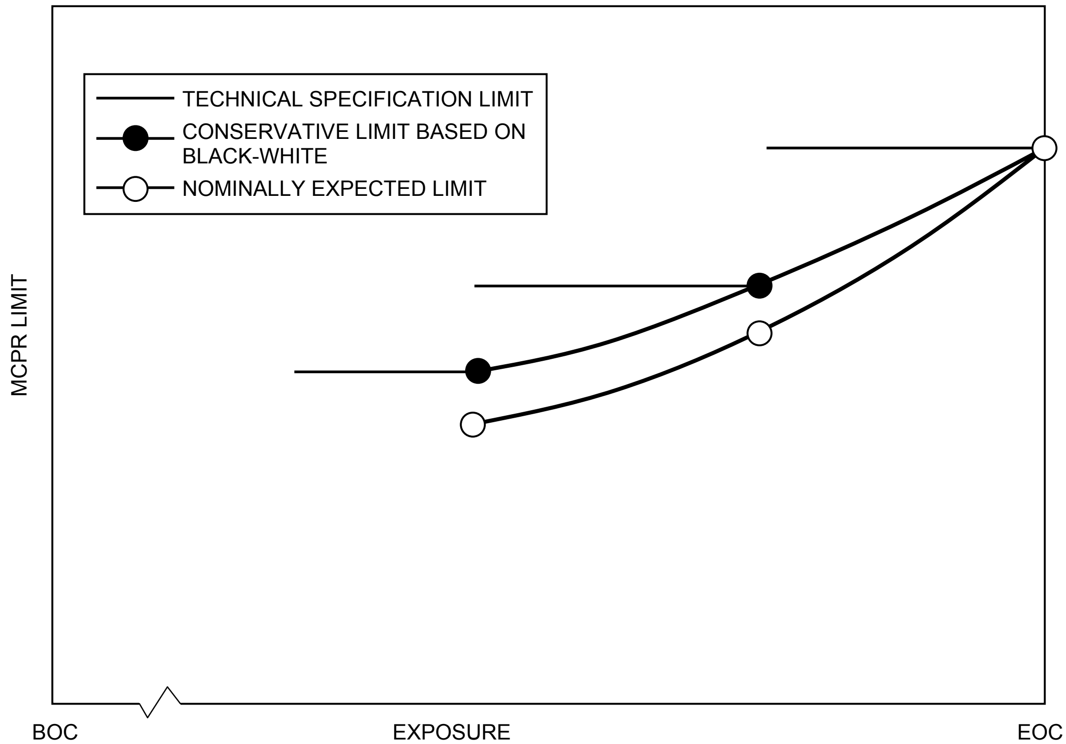

Figure 5.2.2.5-1. Typical Comparison of Exposure-Dependent MCPR Limit to Nominally Expected Limit

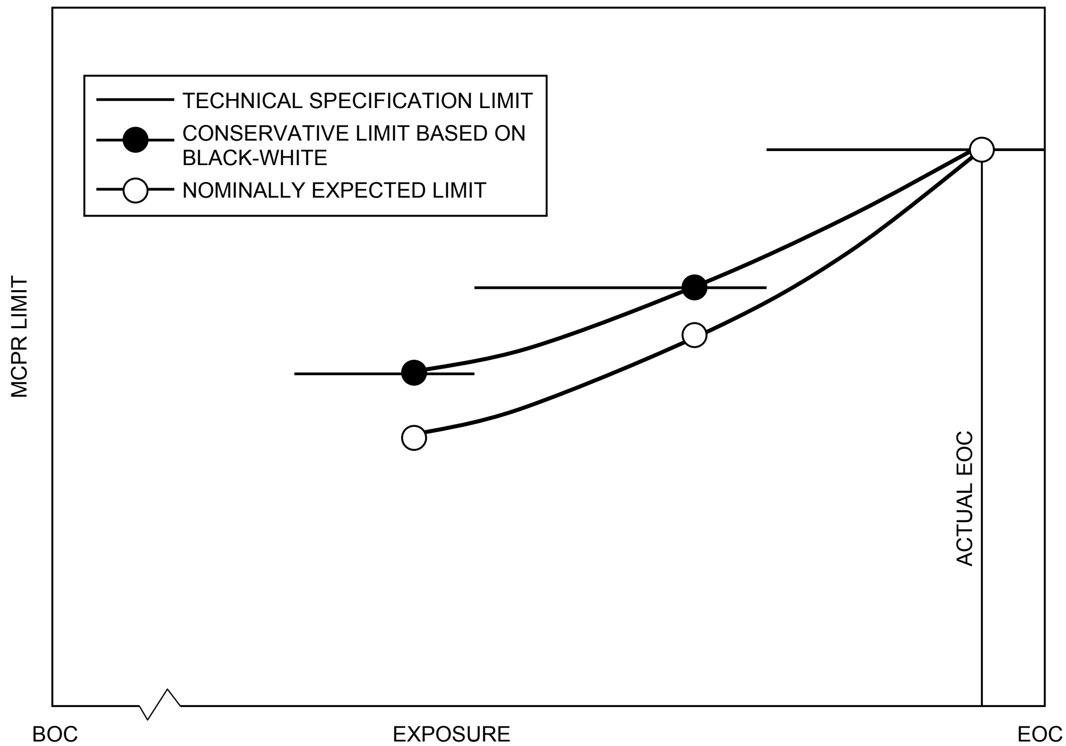

Figure 5.2.2.5-2. Typical Comparison of Calculated Exposure-Dependent Limit to Technical Specification Limit for EOC Exposure Less Than Projected

REQUEST: Subsection 5.2.3 – Effect (of Fuel Densification) on Local Power

A more explicit reference and detailed description of the applicability of the integral hypothesis is required. The same applies to the critical quality-boiling length to integral hypothesis. Identify reference more accurately (e.g., pg. 3-3 of Reference 5-1).

D.H. Lee in the British report AEEW-R-479, "An Experimental Investigation of Forced Convection Burnout in High Pressure Water, Part 4...," pages 11-16, extended the F factor correlation, such that C is a function of flux profile, pressure, hydraulic diameter, and mass flux. Provide a sensitivity study over the range of BWR conditions, with this extended F factor method, for an estimate of critical power reduction due to the presence of spikes.

RESPONSE:

There are three typographical errors on page 5-26. Line 1: Reference 5-13 should be Reference 5-11. Equation preceding Equation 5-2: There should be a coefficient C in front of the first integral. Line 2, last paragraph: q''_o / a''_1 should be q''_o / q''_1. The latter error was corrected in Amendment l; the first two have been corrected with this amendment.

According to the Integral Concept as used in connection with boiling transition (BT) correlations and predictions for a given channel: the occurrence of BT at some point along the channel for a given pressure, mass flux, and inlet subcooling, depends in some manner upon the heat flux profile over the entire boiling length upstream of that point. (This is in contrast to the Local Conditions Hypothesis, according to which the occurrence of BT at some point depends upon the heat flux at that point.)

The Integral Concept is also known as the Integral Method, Integral Hypothesis, Upstream Memory Effect, or just Memory Effect. This concept is applied in the Tong F-factor scheme (Reference 5-11 of the report and References 1 through 3 below), the Critical Quality vs. Boiling Length scheme (e.g., GEXL)

(References 5-1 and 4 below), and an Equivalent Critical Boiling Length scheme due to Silvestri (References 3 and 4). It is probably best described in Reference 3, pages 99 ff.

In Subsection 5.2.3.1, the expression "Integral Hypothesis is reserved specifically for the critical Quality vs. Boiling Length scheme, and it will be so used here. The applicability of the integral hypothesis to BWR operating conditions is described in References 5-1 and 3. In Reference 5-1, page 3-3, B&W round tube data for various axial profiles are shown to correlate well in the X_C vs. L_B plane. The diameter (0.45 in.) and quality range (10% to 40%) correspond closely to BWR conditions. Similar results were obtained with GE Freon annulus data and British round tube data, always at conditions similar to BWR conditions.

In Reference 3, page 99, data are described which were reported earlier by DeBortoli, et al (Reference 5). These data are for two test sections, the first section having uniform axial profile, and the second the same except for the addition of a short hot patch at the exit end. The data demonstrate that the Local Conditions Hypothesis is only valid for boiling transition (BT) at high subcooling. Reference 3 further points out that the X_C vs. L_B scheme is valid, and equivalent to the Tong F-factor method, for BWR conditions.

In summary, the X_C vs. L_B scheme (i.e., the Integral Hypothesis) has been shown to be applicable to BWR fuel assemblies with various axial profiles because it correlates both simple channel data at BWR conditions and full-scale simulated fuel bundle data at BWR conditions.

A procedure for evaluating the constant C in the Tong F-factor scheme is given in Reference 6 (also reported in Reference 2); in terms of S, the axial flux gradient at the point near the exit end of the heated length for which local flux = average flux (normalized with respect to average flux and diameter to render it dimensionless); P/A, the ratio of peak-to-average flux; D, the channel (hydraulic) diameter; and K, a pressure dependent factor. Thus,

$$C = \frac{KS}{D\,(P/A)} \qquad (5.2.3\text{-}1)$$

To assign a limiting value to C, the extreme BWR profile of Figure 5.2.3-1 has been selected for which S/(P/A) = 0.015 (no BWR profile is expected to yield a higher value of S/(P/A)). The other factors on the right of Equation 5.2.3-1 are K = 5.0 (corresponds to P = 1000 psia, see Reference 6), and D = 0.5 in. (typical for BWR's as noted in the report).

$$C = \frac{5.0}{0.5} \times 0.015 = 0.15 \text{ in.}^{-1}$$

When this value of C is inserted in Equation 5-3 of the report, there results:

$$\frac{q_0''}{q_1''} = 1.00291$$

Following the reasoning given on page 5-27 of NEDE-24011-P relative to application of the Tong F-factor method, the critical power with flux spiking is reduced only 0.25% for C = 0.15 in.$^{-1}$. Obviously, the value suggested by Tong

$$C = \frac{0.135}{D} = 0.27 \text{ in.}^{-1}$$

and used in Subsection 5.2.3.1 is more conservative. According to this, the critical power is reduced 0.47% due to the presence of flux spiking.

REFERENCES:

1. L.S. Tong, Boiling Heat Transfer and Two-Phase Flow, reprinted 1975 by R.E. Krieger Publishing Co., Inc., New York (Chapter 6).

2. J.G. Collier, Convective Boiling and Condensation, McGraw-Hill Book Co. (UK) Ltd., Maidenhead, Berkshire England, 1972 (Chapter 9).

3. R.T. Lahey, Jr., and F.J. Moody, The Thermal Hydraulics of a Boiling Water Nuclear Reactor, American Nuclear Society, Hinsdale, Illinois, 1977 (Chapter 4, Sec. 4.3).

4. M. Silvestri, "On the Burnout Equation and on Location of Burnout Points, Energia Nucleare, V. 3, N. 9, September 1966.

5. RA. DeBertoli, S.J. Green, B.W. LeTourneau, M. Troy, and A. Weiss, "Forced-Convection Heat Transfer Burnout Studies for Water in Rectangular Channels and Round Tubes at Pressures Above 500 PSIA," WAPD-188, Oct. 1958.

6. D.H. Lee, "An Experimental Investigation of Forced Convection Burnout in High Pressure Water; Part IV, Large Diameter Tubes at About 1600 P.S.I." AEEW-R479, November 1966.

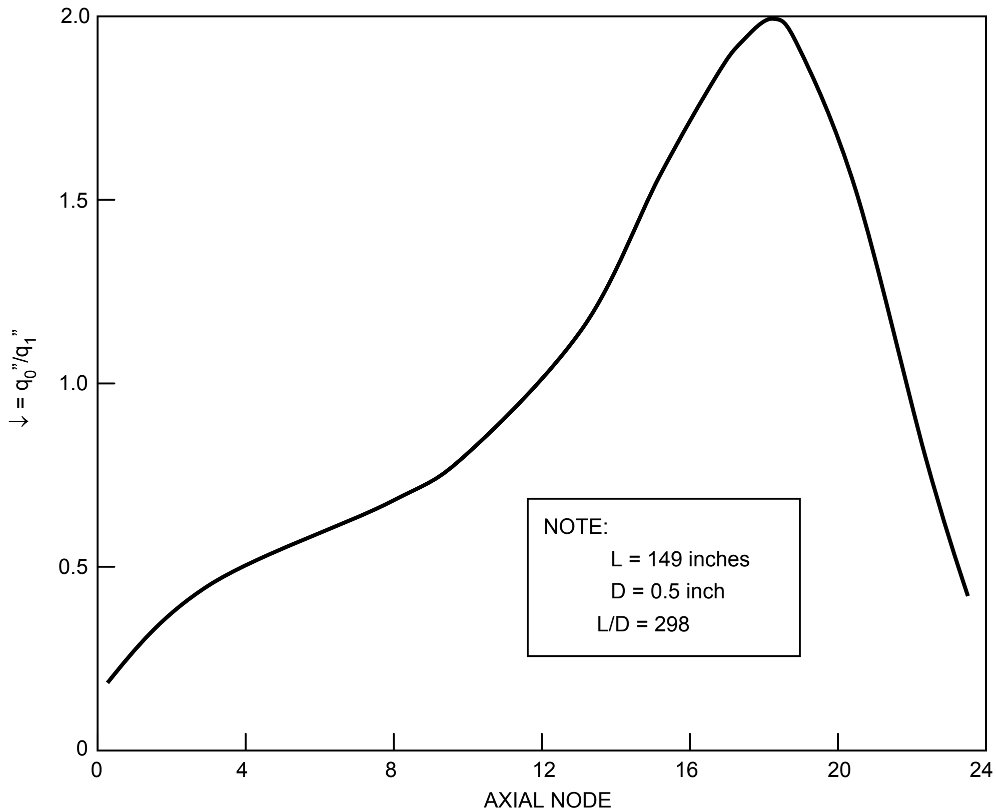

Figure 5.2.3-1. Extreme Profile

REQUEST: Subsection 5.3 – ASME Code Compliance for Vessel Pressure

Provide the assumed fast closure rate (position vs. time) of the MSIV for the event.

Indicate the reference where the MSIV fast closure event is demonstrated to be more severe than the Turbine Trip without Bypass when credit is not taken for the direct scram.

The initial core power levels indicated in Table 5-6 for the BWR/3 plants do not include an increment to account for the core thermal power measurement uncertainty. This is unacceptable for the analyses of this (MSIV closure) limiting pressurization event. Revise Table 5-6 so that the assumed initial steady-state core thermal power level is at least 102% of the licensed power level for this limiting event for all plants (in accordance with the requirements of Standard Review Plan 5.2.2).

Provide a statement regarding when in the cycle the event is most limiting.

Provide or reference the model used to calculate the peak pressure at the bottom of vessel from the transient model output.

Provide a statement as to the assumed performance of the control systems.

Discuss the methods used, assumptions and effects of ATWS, RPT for those plants which have it. List these plants in a table.

RESPONSE:

The MSIV closure event has been demonstrated in the past to be more severe than the turbine or generator trip without bypass when credit is not taken for direct scram. An illustration of the typical relationship between these events is shown in Figure 5.3-1 (attached). Limiting events which are affected by a scram trip are generally most limiting at the end of the cycle due to the dominating effect of the scram degradation in this portion of the cycle because the control blades are withdrawn from the core.

The model used in transient analyses is described in Reference 5-2 of the report. The flow control systems are considered for transient analyses in the manual mode unless otherwise specified. The manual mode presents the most severe conditions for events effected by core flow. Other control systems, such as the feedwater and pressure regulator, are considered in their normal controlling mode.

The power levels indicated in Table 5-6 are consistent with the power levels given in plant technical specifications and used as the basis for the safety analysis. The General Electric basis for code overpressure protection analysis has been documented in Reference 5-14. As noted in the response to the request on Subsection 5.2.1, GE specifically addressed the 2% adder to the initial power, to all transient analysis in Reference 1. Figure 5.3-2 shows the assumed closure rate used in this analysis.

The method of applying ATWS RPT, corresponding to the current design, is to trip the recirculation system after dome pressure has reached the specified setpoint. The assumptions considered are that: (1) trip occurs at the normal setpoint; (2) a 300-msec delay to account for breaker and logic systems delays exists; and (3) the minimum recirculation system inertias are used. The effects of ATWS RPT on transients analyzed for reload licensing generally result in an increase in peak vessel pressure and a reduction in peak neutron and average surface heat flux. No credit is taken for the mitigating effects of ATWS RPT system in the establishment of thermal limits. All BWR/4 plants, except Vermont Yankee, have installed the ATWS RPT. This information is documented in new Table 5-17.

REFERENCES:

1. Letter, R. L. Gridley (GE) to D.G. Eisenhut (NRC), MSIV Closure with Flux Scram Sensitivity of Peak Vessel Pressure to Initial Power Level, September 12, 1977.

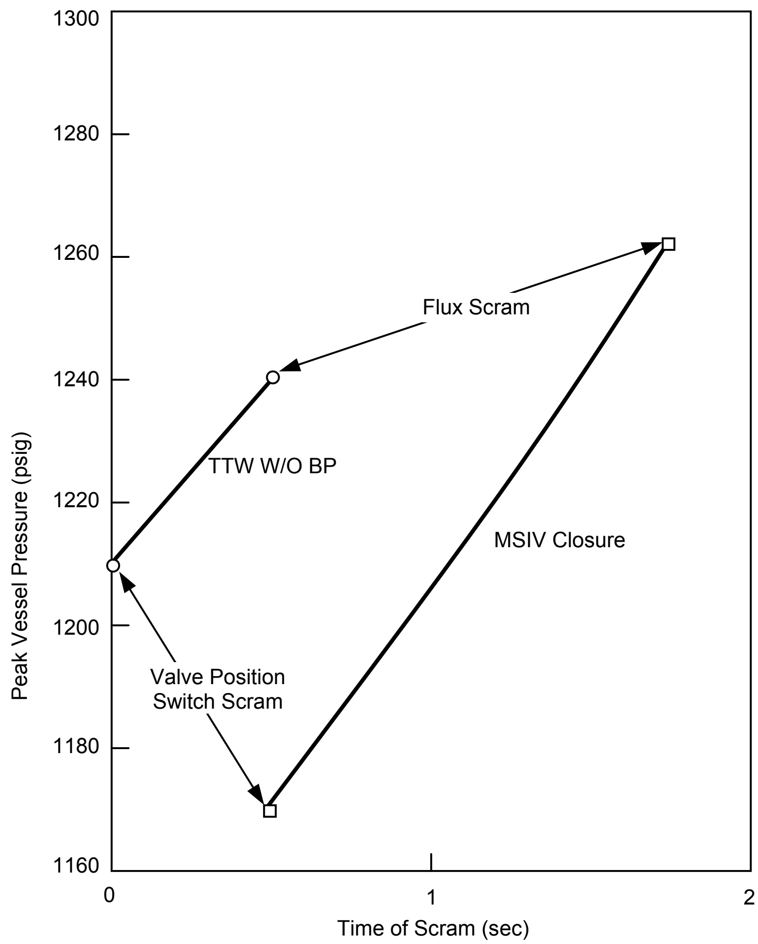

Figure 5.3-1. Effect of Scram Time on Peak Vessel Pressure

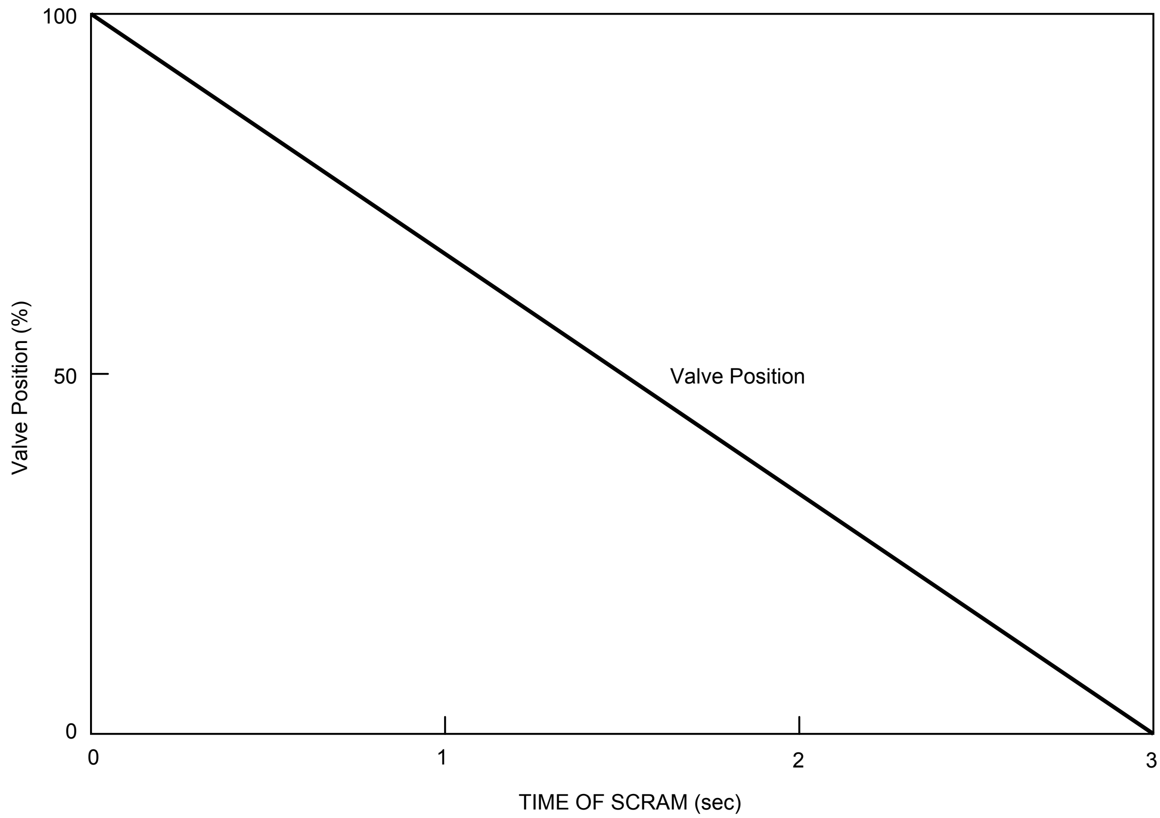

Figure 5.3-2. MSIV Closure

REQUEST: SubsectIon 5.4 – Stability Limits

Indicate by block diagram the important reactivity mechanisms that affect the thermal-hydraulic stability characteristics of a BWR.

Provide a comparison of the new two-water-rod 8x8R fuel design to the old single-water-rod 8x8 fuel design as it relates to total core and channel stability performance. Include the effects of the differing fuel length and diameter (time constant as well as the void and Doppler coefficients.

Discuss the methods used to determine the required nuclear parameter inputs for the stability codes.

Provide a discussion of the basis (in terms of fuel thermal-mechanical integrity) for the selection of a decay ratio of 1.0 as the "Acceptable Performance Limit."

The discussion in Section 5.4 implies (by omission) that reload cores will no longer be evaluated from the standpoint of an "operational design guide decay ratio." Either justify the elimination of this stability criteria for reloads or include a discussion and basis of the operational decay ratio requirements.

Provide a general discussion of the location of the least stable state and why some plants may have it located on the rod block line, while, for other plants, it is located on the 105% rod line.

Provide a set of representative figures which quantitatively maps the limiting lines on the power flow map (natural circulation, minimum recirculation flow and rod block/105% rod lines) onto the decay ratio vs. percent power plot. Provide a general discussion of the mapping. Include the most limiting points and their respective limits.

RESPONSE:

The purpose of the stability analysis is to demonstrate analytically that the acceptable performance limit (i.e., the decay ratio is less than 1.0 or the damping coefficient is greater than 0) is satisfied. This limit was selected because it was considered prudent not to operate a plant in a potentially oscillatory mode. In Reference 1, it is demonstrated that there is considerable margin between plant operation with a limit cycle (decay ratio equal to 1.0 and previously established safety limits which are in themselves conservative.

The purpose of the reload stability analysis is to demonstrate the capability for safe plant operation. This is done by demonstrating for all fuel types that the acceptable performance limit is satisfied for each type of stability analyzed. Because all fuel types are analyzed for each reload, no comparison between 8x8 and 8x8R is considered necessary. Also, it is not considered necessary to provide an evaluation of the automatic flow control range where the operational design guide decay ratio is less than 0.25 because it is only a guide used for operation with automatic flow control.

For reload cores, the stability analysis with a linearized analytical model is described in Reference 2. A block diagram of the important reactivity mechanisms which affect thermal-hydraulic stability is shown in Figure 5-6 of Reference 2. The methods used to develop the nuclear feedback characteristics are given in Reference 3.

Stability analyses are performed along the rod line corresponding to the highest power level for which the safety analyses are performed. This power level is documented in the plant technical specifications. A power flow map (natural circulation, minimum circulation flow and rod block and safety analysis rod lines) is given in each plant final safety analysis report. A typical power flow map is shown on Figure l-5 of Reference 2. The location of the least stable point flow and the highest attainable power level is shown.

REFERENCES:

1. Letter, R.E. Engel to D.G. Eisenhut, “Boiling Water Reactor Stability Margins,” April 4, 1977.

2. “Stability and Dynamic Performance of the General Electric Boiling Water Reactor,” January 1977 (NEDO-21506).

3. “Generation of Void and Doppler Reactivity Feedback for Application to BWR Design,” December 1975 (NEDO-20964).

REQUEST: Subsection 5.5.1 – Control Rod Drop

According to Section 15.4.9 of the Standard Review Plan, one of the CRD fuel failure thresholds to be considered is fuel cladding dryout (MCPR <1.06) when the event occurs from an initial rated power condition. Provide a bounding assessment of the number of 8x8R failed fuel rods which would occur as a result of a CRD from rated conditions. Show that a CRD from the hot full-power operating condition gives less severe radiological consequences than one from the hot or cold shutdown conditions for all the plants listed in Table 1-1. Show that the calculated doses for this situation are also well within the 10CFR Part 100 exposure guidelines.

Provide a statement that the tabulated scram times used in the CRD analysis equal or exceed the technical specifications scram times for the plants considered.

Reference or provide a generic sensitivity study which identifies the key input parameters to the CRD accident (e.g., provide a table of all of the input parameters considered and their effects on the consequences of the CRD accident).

In Equation 5-14 the "+" should be replaced by an "=".

In the definition of BP, "ith" should be replaced with "jth".

Discuss the analyses which conclude that the closest approach of actual plants operating parameters to the 280 cal/gm is represented by Figure 2-25.

A maximum interassembly local peaking factor, P_F, of 1.3 forms the basis for the maximum allowable rod worth of 1.3% ΔK and is cited in the Technical Specifications of several operating BWRs. Describe the procedure (analysis and/or Tech Spec change) which will be used in connection with a reload submittal for one of these plants for the case in which a specific plant's accident reactivity, Doppler coefficient and scram reactivity conservatively fall within the bounding CRDA analysis, but the maximum interassembly local peaking factor, P_F, is greater than 1.3.

With regard to the interassembly local power peaking factor, P_F:

(a) Describe in more detail the methods used to calculate P_F as outlined in Equation 5-14.

(b) Is P_F to be calculated on a plant-cycle specific basis and provided with each reload submittal?

(c) If not, justify the statement that a P_F greater than 1.30 would not be expected to occur at any plant.

In order to base a reload license application on the bounding rather than a plant-specific control rod drop analysis, it is necessary that the plant-specific scram reactivity be within the bounding limit up to a total of 0.02 ΔK (pages 5-46 and 5-47). Do you propose that plant specific analyses are not necessary for cases where the above criteria is met but for which the plant specific scram is outside the bounding limit above ΔK = 0.02?

Justify the adequacy of global values for Doppler Coefficients and delayed neutron fractions in analyzing the control rod drop accident for reload cores containing highly exposed fuel bundles. Take into consideration the possibility that locally the delayed neutron fraction of Doppler coefficient may be smaller in magnitude than the global values.

Either justify that ß will never be less than the limiting values assumed for bounding analyses, or state clearly in the text that ß will be calculated and compared to the limiting value for each reload.

There appears to be all inconsistency between the wording of item (2)-(C) on page 5-38 and item (2)-(C) on page 5-39 with regard to the final position of the dropped rod. Please clarify this situation.

Provide a discussion of the bounding analysis methods which will be used in connection with plants using group notch withdrawal sequences.

Provide a discussion of the models methods and assumptions used to demonstrate that the maximum reactor pressure during any portion of the CRD is less than that which would cause the emergency condition stress limits (as defined in the ASME Boiler and Pressure Vessel Code, Section III) to be exceeded. Provide or reference a bounding or a plant-specific analysis (or a previously submitted plant analysis) which demonstrates that the criteria will be met by the reload fuel.

RESPONSE:

The study "Control Rod Drop Accident at Rated Power" by R.C. Stirn and C.J. Paone was submitted under cover of Reference 1. This study, utilizing the Hench-Levy Critical Heat Flux correlation, showed no fuel rods in transition boiling or reaching the 1% plastic strain linear heat generation rate.

The Technical Specifications are assumed to be an integral part of the licensing basis for each plant. The value(s) assumed for any parameter used in licensing calculations will be at least as conservative as the Technical Specifications value. The only exceptions are those cases in which the submittal is intended to justify a request for a technical Specification change.

The key input parameters to the control rod drop accident which influence the resultant peak fuel enthalpy are defined on pages 5-45 and 5-46. These parameters are the accident reactivity shape functions, total control rod worth, assembly local power peaking factor, delayed neutron fraction, Doppler reactivity feedback and rod drop velocity. The sensitivity of the control rod drop accident resultant peak fuel enthalpy is already documented in this report and References 5-17, 5-18 and 5-19.

Equation 5-14 was corrected in Amendment 1. The typographical error in the definition of BP was also corrected. Figure 5-25 shows only the reactivity shape functions for the bounding analyses and a typical not-bounding actual plant case. Calculations of the type used to generate Figures 5-22, 5-23, 5-24 and 5-25 are discussed in detail in References 5-17, 5-18, and 5-19.

Bounding curves for higher P_f were included in Amendment l. This amendment further stated that, because so many parameters are involved in the determination of the resultant peak fuel enthalpy due to a control rod drop accident, it is not realistic to set a specific value of maximum control rod worth that could be used in Technical Specifications. In the past, a local peaking factor of 1.3 was applied as the upper limit and, based on this local peaking value, a "maximum allowable" control rod worth of 1.3T Δk was set. In reality, some reload cores exceeded both the 1.3% Δk value for rod worth and the l.3 local peaking factor value, yet met all the boundary requirements. Therefore, no specific control rod worth requirement will be set other than those described above" (i.e., accident, reactivity, Doppler coefficient, and scram reactivity).

The bounding value for the Doppler reactivity coefficient given in Figure 5-26 was based on the beginning of life BOL or zero exposure values as stated on page 5-46. It is also stated on this page that this is conservative, since the Doppler coefficient will become more negative with increasing exposure. By comparing Figure 5-26 with Figure 3-5, it will clearly be noted that the Doppler coefficient does become more negative with increasing exposure. The bounding values selected for Figure 5-26 were based on 7x7D initial core fuel. Figure 2.3-3 of Reference 5-5 clearly shows that this is the most limiting BWR lattice design with regards to Doppler reactivity. Even at the beginning of a fuel cycle, some of the four fuel bundles surrounding a control blade in a reload core design will be exposed; hence, the composite Doppler coefficient of the four bundles surrounding the blade will be more negative than the bounding values in Figure 5-26.

A bounding value of 0.005 was selected for the delayed neutron fraction. Because high worth control blades occur where at least one fresh fuel bundle has been loaded as one of the four fuel bundles surrounding the control blade, the delayed neutron fraction in a local region surrounding the blade will always be greater than the conservative bounding value of 0.005 even at end of cycle.

The bounding analysis methods are not dependent on the type of plant being analyzed. While the specifics may differ in such areas as rod patterns considered and distance a rod is allowed to drop, the basic calculations are identical. Inconsistencies in documentation in this subsection were corrected in Amendment 1.

Reference 2 discusses the maximum pressure occurring during a CRD relative to applicable stress limits. A copy will be transmitted as soon as it is available (first quarter 1978). Preliminary results indicate a maximum pressure increase of Less than 15 psi for this accident.

REFERENCES:

1. Letter, I.F. Stuart (GE) to V. Stello (NRC), “Analysis of Control Rod Drop Accident at Rated Power,” December 7, 1973.

2. “Transient Pressure Rises Affecting Fracture Toughness Requirements for Boiling Water Reactors,” T.E. Cooke, et al., to be issued.

REQUEST: Subsection 5.5.2 – Loss-of-Coolant Accident

Subsections 5.5.2.1 and 5.5.2.2 are missing; provide these subsections.

RESPONSE:

Documentation of the loss-of-coolant accident analytical methods can now be found in Subsection 5.5.2.

REQUEST: Subsection 5.5.4 – Fuel-Loading Error

The discussion of the fuel-loading error is inadequate. Provide a detailed discussion of the models methods and assumptions which are employed in the FLE analysis. Include a discussion of the procedures used to determine the most severe fuel loading situation vs. location in the reload core. Discuss how the fresh fuel reactivity effect is maximized. Discuss the assumption on the CPR value of the adjacent monitored bundles. Discuss which of the analysis inputs are plant specific and which incorporate conservative or bounding inputs. Discuss the assumptions used to determine the local peaking factors and R-factor; e.g. fuel type (enrichment), water gap thickness. Discuss how fuel densification effects are incorporated into the evaluation of the peak LHGR. Discuss which analysis (misoriented or mislocation) is performed for each reload. Discuss whether separate analyses (where applicable) are performed for a postulated mislocation of a fresh (8x8R) assembly in a location intended for: (a) an exposed 7x7 location; (b) an exposed 8x8 location; and (c) an exposed 8x8R location.

RESPONSE:

The analytical models and assumptions used in the analysis of the fuel assembly loading error are described in References 1 and 2. These descriptions include the various alternative assumptions with suitable bases which can be used to reduce the undue conservatism in the most conservative methods. The above information has been incorporated into Subsection 5.5.4.

REFERENCES:

1. Letter, R.E. Engel (GE) to D.G. Eisenhut (NRC), "Fuel Assembly Loading Error", June 1, 1977.

2. Letter, R.E. Engel (GE) to D.G. Eisenhut (NRC), "Fuel Assembly Loading Error", November 30, 1977.

REQUEST: Appendix A – Supplemental Reload Licensing Submittal

The following additional information should be provided in Appendix A.

2. FUEL MECHANICAL DESIGN

Provide a brief statement to the effect that all of the generic thermal-mechanical design analyses were checked for applicability to the plant-specific reload application, in accordance with the provisions of Subsection 2.4.3. State the results and conclusions of this review in regard to the predicted reload fuel performance during life and the applicability of the generic calculations. Reference should also be made to generic tables appearing in the Topical (see comments relating to Subsection 2.4.2, particularly the last three sentences of the first paragraph.

3. NUCLEAR EVALUATION

Indicate the exposure corresponding to the reported value of "R". Provide the assumed core average exposure increment "window" for the previous cycle.

5. REACTOR LIMITS

This section should provide a summary paragraph giving the <u>Operating Limits</u>, to be followed by the paragraphs containing the <u>Inputs and Results</u> of the accident and transient evaluations. That is, replace (a) operating MCPR Limit, 5.1, by (a) Operating Limits. Include in paragraph (a):

1. Operating MCPR for each fuel type and exposure (if applicable).

2. Maximum Total Peaking Factor for each fuel type or equivalent.

3. MAPLHGR for the (8x8R) fuel type vs. exposure (when applicable).

4. Gross Core Thermal power limit vs. exposure (if applicable).

Include the prompt reactor period, τ_o, (or equivalently the prompt neutron lifetime l*, and delayed neutron friction β) in Table 1.

Additionally, the proposed new format for Appendix A results in a substantially reduced content of the plant-specific portion of the reload licensing submittal. Accordingly, we require that additional core wide transient analysis output parameters be provided in Appendix A. These parameters are: positive (void) reactivity vs. time, Doppler reactivity vs. time, scram reactivity vs. time and total reactivity vs. time. Finally, the plot of percent core power vs. percent core flow may be deleted, since it does not significantly contribute to the interpretation and/or evaluation of the transient results.

RESPONSE:

The goal of the Generic Reload Fuel Application is to reduce the documentation required in the individual plant supplemental submittal by incorporating generic information and bounding analyses in the generic reload licensing topical report. Much of the additional information requested is not required to demonstrate the safety of a plant-specific reload.

The thermal and mechanical fuel design analyses are implicitly generic. In addition, Subsection 2.7 states that the individual plant reload parameters are checked to ascertain that the generic analyses apply. Should these analyses not apply to a particular reload, an appendix documenting the plant specific analysis will be provided. This appendix will be referenced in Section 1 of the plant supplemental submittal.

The previous end-of-cycle core average exposure, both nominal and minimum from cold shutdown considerations, will bc reported in Section 3 of the supplemental submittal.

Note that Appendix A was revised in Amendment 1. Exposure dependent operating MCPR limit is given in Section 11, and MAPLHGR vs. exposure is given in Section 14. Local, radial, and axial peaking factors for each fuel type and exposure are given in Section 7. A discussion of the total peaking factor is given in the response to the request on Subsection 5.2.1.5. Gross core power relative to exposure for the limiting pressure and power increased transient is given in Section 9.

The prompt neutron lifetime, l*, becomes significant in transient analyses only if the reactor nears prompt criticality. Because the transient analyses presently do not approach prompt criticality, neither l* nor the prompt reactor period, to, is required. The delayed neutron fraction, β, is included in all of the reactivity inputs to the transient analyses.

General Electric believes that the Generic Reload Fuel Application results in a substantially increased amount of information for each plant reload. While the supplemental submittal is indeed smaller, the amount of generic information is significantly enhanced. However, the individual plant transient results will include void, Doppler, scram and net reactivity balances in plant supplemental submittals after 1 June 1978.

It should be noted that, thermally and hydraulically, 8x8 and 8x8R fuel are very nearly identical. There are some differences (i.e., active fuel length and R-factor). However, dynamically the thermal response of the two fuel designs are essentially the same. Thus, the ΔCPR differences due to transients are almost indiscernible. Analyses performed on an operating plant for the load rejection w/o bypass event attest to this fact, as shown below:

Fuel Design	Initial CPR	ΔCPR
8x8	1.268	0.213
8x8R	1.267	0.212

Therefore, it has been concluded that both fuel designs will have the same operating limit MCPR and only the results of the retrofit fuel is reported.

REQUEST: Attachment 2-A – Maximum Spike–Subsection 2.4.2.1.1–Power Spiking

The 2.2% power spiking penalty referenced in this subsection has been previously used in connection with 8x8 fuel types having 144-inch and 146-inch fuel stack lengths. Since the 8x8R fuel assemblies for BWR/4's will have a pellet column 150 inches long (which will increase in the maximum gap size), an increase in the power spiking penalty can be expected. Provide an assessment of the maximum spike for the 8x8R design relative to the 8x8 design.

RESPONSE:

Power spiking analysis for the 8x8R fuel design was performed using a gap size consistent with the 150-inch active fuel length. The model used is described in Reference 1. The result, in the LTR, reports that the 2.2% power spiking penalty is applicable to both the 8x8 and the 8x8R designs. Since the time of calculating the power spiking penalty for the 8x8 fuel design, the calculational model used has been improved in order to provide greater stability and increased accuracy in calculated results. The improvements made consist of using continuous cumulative probability density functions rather than discrete stepwise distributions. Based on this model, the maximum calculated power spiking penalty at the top of the active fuel lengths for 8x8 and 8x8R, respectively, are 1.6 and 2.0%.

REFERENCES:

1. "Responses to AEC Questions—NEDM-10735," NEDM-10735 Supplement 1, April 1973.

REQUEST: Attachment 2-B – Normal Operation – Subsection 2.5.3.1.1 – Cladding Creep Collapse

Densification flux spiking is not mentioned in the cladding creep collapse evaluation. Explain why densification spiking does not significantly affect the collapse analysis results.

RESPONSE:

The calculated power spiking penalty is added to the MLHGR in the cladding creep collapse evaluation for the purpose of determining cladding temperature. Additional explanation of this assumption has been added to Subsection 2.5.3.1.1 in response to the request on Figure 2-16.

REQUEST:

Attachment 2-B. Normal Operation—Subsection 2.5.3.1.2 – Stress Evaluations

This section states that the thermal analysis inputs are given in Subsection 2.4. Included in 2.4 are the densification effects, which includes power spiking, as well as "decreased pellet-cladding thermal conductance resulting from increased radial gap size." Although the former consideration is conservative from a clad stress evaluation standpoint, the latter is nonconservative, since it will result in lower and delayed PCI stresses due to rod power changes. The anisotropic radial densification model was established to provide a conservative fuel rod thermal analysis for stored energy determination and is not considered to be appropriate for fuel rod mechanical design evaluations. Therefore, the stress evaluation should be based on power spiking but without including an increase in the pellet-to-clad gap due to densification. Therefore, provide a quantitative assessment of the effect of this new assumption on the limiting fuel rod stresses reported in Table 2-10.

RESPONSE:

An increase in power due to power spikes as a result of densification and axial gap formation is included in the thermal analysis performed and described in Subsection 2.4. No credit is taken for reduced pellet diameter as a result of densification in the thermal analysis used in conjunction with the stress analysis described in Subsection 2.5.3.1.2. Additional clarification regarding the models and assumptions used for the various thermal analyses has been added to Subsection 2.4 in response to Question 2.4.2. Since the thermal analysis in question was performed using the assumption stated, there is no change to the fuel rod stresses reported in Table 2-10.

REQUEST: Attachment 2-B. Normal Operation—Subsection 2.5.3.1.4 – Fatigue Evaluation

This section does not explicitly discuss densification effects in the cladding fatigue evaluation. Reference is made, however, to the application of the cladding stresses determined in Subsection 2.5.3.1.(2) in the fatigue analysis. Therefore, since the fatigue evaluation is based on the stress analysis, which in turn is affected by fuel densification spiking (and radial gap size change) effects, the above comments, relative to Subsection 2.5.3.1.2, apply equally to this subsection and should be addressed in your responses.

RESPONSE:

See the response to the request on Attachment 2, Subsection 2.5.3.1.2.

REQUEST: Attachment 2-C. Anticipated Transients – Subsection 2.4 – Fuel Rod Thermal Analysis

From the discussions in Subsections 2.4 and 2.4.1.1, it is unclear if densification spiking is explicitly considered in the 1% plastic strain evaluations; i.e., for local (RWE, FLE) transients. It is the staff's position that power spiking must be included in the analysis of such localized transients relative to the calculated initial kW/ft and kW/ft increase. Therefore, clearly describe how spiking is included; either in the calculated LHGRs corresponding to 1% strain, or in the peak transient LHGRs from the plant specific analyses of the anticipated local events. Reference these descriptions in Subsections 5.2.1.5 and 5.5.4.

RESPONSE:

Based on the calculated exposure-dependent LHGR values in Subsection 2.4.1.2, it has been determined that the power required to produce 1% plastic strain in the cladding is equal to or greater than 175% of the design maximum steady-state power throughout life for all rod types in the assembly for the 8x8 fuel design and greater than 160% for the 8x8R fuel design. These ratios consider the presence of a calculated power spiking penalty being added to the MLHGR.

The calculated MLHGR during the RWE is compared to that associated with 1% plastic strain in the cladding to ensure that fuel damage is not expected during the event.

Relative to the loading error accident, the 1% cladding plastic strain is used to predict if fuel failures are expected to occur. If failures are predicted, a radiological evaluation of the consequences will be performed.

The above information has been incorporated into Subsections 2,4.1.2, 5.2.1.5, and 5.5.4, respectively.

REQUEST: Attachment 2-D. ACCIDENTS – 5.5.1 – Control Rod Drop Accident

The document referenced (5-20), for the effect of axial gap formation, due to fuel densification, on the control rod drop accident results (99% probability that local power peaking will be less than 5%) does not address the 8x8R fuel design with the longer (150 in. vs. 144 in.) fuel column length. Provide an evaluation of the maximum local peaking factor increase for the 150-in.-long fuel column associated with the BWR/4 8x8R fuel types. Clearly describe how the resulting power spike effect is accommodated by “adjusting” the local peaking. Indicate whether separate power spike penalties will be used for BWR/2’s and 3’s (144-in. fuel column) and BWR/4’s (150-in. fuel column) or whether a bounding BWR/4 spike will be used in the CRD evaluations.

RESPONSE:

An evaluation was performed for the 8x8R fuel regarding potential power spikes in the RDA condition. The maximum gap size associated with 150-in. active fuel length was used and the evaluation was performed for the top of the active fuel length. Results show that there is a 99% probability that the power spike will be less than 5% which is the same result reported for other fuel types in Reference 5-20. The 5% power spike effect is accommodated by simply increasing the maximum local peaking (P1) factor by 5% (i.e., Pe x 1.05). A bounding value (5%) is used for all CRD evaluations.

REQUEST: Attachment 2-D. Accidents – Subsection 5.5.2 – Loss-of-Coolant Accident

Provide a description of how densification spiking effects will be considered in the LOCA analysis.

RESPONSE:

Densification effects considered in the LOCA analyses is documented in Subsection 5.5.2 (see the response to the request on Subsection 5.5.2).

Responses To Questions On The Barrier Fuel Amendment (Amendment 6)

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QUESTION

RESPONSE

REFERENCES:

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QUESTION

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RESPONSE

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QUESTION

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RESPONSE

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REFERENCES:

1. Letter, J.S. Charnley (GE) to F.J. Miraglia (NRC), “Barrier Fuel Amendment to NEDE-24011-P-A-4,” November 19, 1982.

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Responses To GESSAR II Questions Regarding SRP Section 4.2

QUESTION 4.3

GESTAR II (NEDE-24011), which contains the fuel system design safety analysis for GESSAR II, does not contain clearly identifiable design bases for most of the fuel damage, fuel failure, and coolability phenomena listed in Item II.A of Section 4.2 of the Standard Review Plan (SRP). Thus, except for cladding overheating (Item II.A.2(c)) and fuel pellet overheating [Item II.A.2(d)], we have not been able to identify design basis statements in the text of GESTAR-II or in the referenced documents, even with the aid of Appendix A in Amendment 5 to NEDE-24011. While it is possible in certain cases to infer the design bases, it is preferable to have them clearly stated. Therefore, for each of the fuel system phenomena discussed in Section 4.2 of the SRP, except the two cited above, provide a concise design basis statement which indicates the design objective related to that issue. In responding to this question, provide a cross-reference to Question 490.05.

RESPONSE 4.3

The design basis for each of the phenomena listed in Item II.A of SRP 4.2 was discussed in the presentation to the Core Performance Branch of the NRC on January 25, 1983.

QUESTION 4.4

Unless otherwise stated in Section 4.2 of the SRP, you should provide a design limit for each design basis. The design limit should be a numerical value of some parameter which provides assurance that the design basis (i.e., the objective or need) will be met. For all but the following phenomena, adequate design limits have been supplied or adequate explanations have been provided for the lack of design limits: (1) Fretting Wear [Item II.A.l(c)]; (2) External Corrosion and Crud Buildup [Item II.A.l(d)]; (3) Fuel and Burnable Poison Rod Pressures [Item II.A.l(f)]; (4) Fuel Assembly Liftoff [Item II.A.l(g)]. Accordingly, provide design limits for the above listed phenomena. Alternatively, discuss why no limits are required. Design limits for cladding rupture [Item II.A.2(g)], mechanical fracturing [Item II.A.2(h)], ballooning [Item II.A.3(c)] and fuel assembly structural damage [Item II.A.3(e)] are being addressed as part of separate generic reviews and need not be discussed in your FSAR now. When our generic review of these matters is completed, you should incorporate the appropriate resolutions in your FSAR.

RESPONSE 4.4

Each of the four phenomena listed above is discussed separately:

1. Fretting Wear – No design limit for fretting wear is required since testing and experience indicate that fretting wear has been eliminated as an active wear mechanism.
2. Corrosion – No separate design limit is required for corrosion and crud buildup because it is considered in the design analyses. Corrosion and crud buildup impact the calculated cladding temperature and material strength, and the ability of the clad to meet the stress limits prescribed in NEDE-24011-P-A-5. Thus, the amount of corrosion and crud buildup is limited by the stress limits on the clad.
3. Internal Rod Pressure – Limits for internal rod pressures are discussed in the response to Question 4.8.
4. Fuel Liftoff – The GE philosophy with respect to fuel lift is that under worst-case hydrodynamic loading conditions, vertical liftoff forces must not unseat the lower tie plate from the fuel support piece such that the resulting loss of lateral fuel bundle positioning could interfere with control blade insertion.

QUESTION 4.5

The fuel assembly description and drawing contained in GESTAR II are much less comprehensive than called for by Item II.B of Section 4.2 of the SRP. This particular item in the SRP contains a list of the information commensurate with an acceptable fuel system description. Accordingly, provide the information identified in Section 4.2 of the SRP.

RESPONSE 4.5

The GE BWR fuel assembly design is described in Section 2.1 of GESTAR II. Design specification limits are provided in Table 2-1 of that report.

The other information listed in Item II.8 of SRP 4.2 is not provided because it is either not used in the analyses or because enough detail is provided for the specific item to be derived. End plug dimensions are a function of U-235 enrichment, gadolinia concentration, and dependent on whether the end plug is in the upper or lower end of the fuel rod. The many sizes of end plugs possible make the amount of information available voluminous. Cladding inside roughness and pellet roughness data are available only as tolerances and, therefore, are not provided.

Tolerances are considered in the safety evaluation, which is performed and measured against approved safety criteria. The safety evaluation is performed assuming the most limiting combination of tolerances for all critical dimensions. This is discussed in Section 2.4.1 of GESTAR II.

In summary, GE believes that enough information is provided in sufficient detail to provide a reasonably accurate representation of the fuel design, thus satisfying the intent of the SRP.

QUESTION 4.6

In the recently submitted Appendix A to NEDE-24011-P-A-5, you state that the channel deflection analysis is provided in Section 5.3.2 of NEDE-21354-P. However, no such section exists in that topical report. Correct this reference. Furthermore, since the referenced channel box deflection report is relatively old (1976) and more data are available now regarding the magnitudes and rates of channel box deflection as a function of service, indicate whether: (1) the data verify the predictions of the deflection model in NEDE-21354-P; (2) your model adequately addresses channel bowing as well as bulging; and (3) you still recommend the periodic settling friction tests and measurements described in NEDE-21354-P, and if so, on what schedule. If you now recommend some other approach, or if NEDE-21354-P procedures have been revised, describe the changes and discuss their rationale.

RESPONSE 4.6

The reference in NEDE-24011-P-A-5 should be to Section 3.2 of NEDE-21354-P. This reference will be corrected in the next amendment to NEDE-24011-P-A-5.

The other parts of this question are addressed below:

1. Additional channel bulge data to higher exposures have been obtained since the publication of NEDE-21354-P. These data confirm the adequacy of the model used by General Electric to predict channel bulge.

2. The model in NEDE-21354-P addresses only channel bulge. GE has recommended to the utilities an approach to mitigate the effects of channel bowing.

3. The following general guidelines minimize the potential for and detect the onset of channel bowing:

   (a) Channels shall not reside in the outer row of the core for more than two operating cycles.

   (b) Channels that reside in the periphery (outer row) for more than one cycle shall be situated in a core location each successive peripheral cycle which rotates the channel so that a different side faces the core edge.

   (c) At the beginning of each fuel cycle, the combined outer row residence time for any two channels in any control rod cell shall not exceed four peripheral cycles.

After core alterations (i.e., reload) and before reaching 40% thermal power, a control rod drive friction test[2] is recommended for those cells exceeding the above general guidelines. After the technical specification scram speed surveillance test on each rod, as required by BWR/6 Standard Technical Specification 4.1.3.2.a, each control rod meeting the above conditions will be allowed to settle a total of two notches, one notch at a time, from the fully inserted position.

[2] This control rod settling friction test provides an equivalent level of the tests described in NEDO-21354. This test provides adequate assurance of the scram function. The amount of friction detectable by this test is ~250 lb. Control rod drive tests indicate that the CRD will tolerate a relatively large increase in drive line friction (350 lb) while still remaining within technical specification limits. The control blade is in its most constrained, highest friction location when it is fully inserted. The ability of the blade to settle from this position demonstrates that the total drive line friction is less than the weight of the blade (~250 lb).

Total control rod drive friction is acceptable if the rod settles, under its own weight, to the next notch within approximately 10 seconds. If the rod settles too slowly, a rod block alarm will actuate, indicating possible impending channel box-control blade interference. The results of this test will be considered acceptable if no rod block alarm is received. This testing will give an early indication of this interference and will prompt an investigation into the source of the friction. If necessary, corrective action will be completed before startup after the next core alteration.

In lieu of friction testing, fuel channel deflection measurements may be used to identify the remaining amount of channel lifetime.

QUESTION 4.7

In GESTAR II (NEDE-24011), which is the primary support document for the fuel system for your proposed 238 Nuclear Island design, you have not provided a discussion of fuel assembly liftoff for normal operation and "abnormal transients" which are separate and distinct from our concerns regarding the seismic-and-LOCA-loads liftoff. As indicated in Item II.A.l(g) of Section 4.2 of the SRP, however, worst-case hydraulic loads for normal operation should not exceed the holddown capability of the fuel assembly. Although your letter from Gridley to Eisenhut, dated July 11, 1977, addresses this issue for plants and fuel designs of 1977 vintage, it is not evident that assembly liftoff will be precluded for normal operation, including anticipated operational occurrences or abnormal transients in your proposed 238 Nuclear Island. Accordingly, provide a discussion of your analysis of this issue. The design basis and limits aspects of this issue should be addressed as part of your response to Questions 490.0l and 490.02.

RESPONSE 4.7

Design changes which have occurred in the GE BWR fuel bundle since the July 11, 1977 letter from Gridley to Eisenhut have not changed the conclusions reached in that letter. Therefore, it is considered applicable to the fuel to be used in the 238 Nuclear Island design.

GE performs two analyses for the effect of hydrodynamic loads on our fuel bundles. The results of the analysis are provided in the plant-specific New Loads Report. The first analysis is for loads on the bundle resulting from upset conditions (i.e., loads from the combination of normal operation plus OBE plus scram plus SRV actuation). For these operating conditions, including anticipated operational occurrences, GE does not calculate any fuel bundle separation from the core support piece. The second analysis is for loads on the bundle resulting from faulted conditions (i.e., loads from the combination of normal operation, LOCA, SSE, SRV Actuation and Mark III containment hydrodynamic conditions). The method of analysis used for these faulted condition loads is provided in NEDE-21175-3-P, "BWR Fuel Assembly Evaluation of Combined SSE and LOCA Loadings". These faulted loads bound the loads calculated for upset conditions. NEDE-21175-3-P provides a discussion of the margin between the calculated fuel movement as a result of the faulted condition loadings and that movement required to disengage the bundle lower tie plate from the fuel support piece.

QUESTION 4.8

You state in Section A.4.2.1.1.6 of GESTAR II that there is no limit for internal gas pressure. The internal pressure is used in conjunction with other loads on the fuel rod cladding in calculating cladding stresses. The results of such calculations which are provided in Section 2.5.1 of NEDE-24011, show that the calculated cladding stresses can be accommodated. Although this analysis may satisfy our acceptance criteria for cladding stress (Item II.A.l.a of Section 4.2 of the SRP), it does not satisfy our acceptance criterion for rod internal pressure (refer to Item II.A.l.f of SRP Section 4.2 and Question 490.01) because this criterion involves more than stress limits on the cladding. The rod internal pressures used in your cladding stress calculations are well in excess of the nominal coolant system pressure. Accordingly, justify operation under these conditions and explain why the absence of an internal gas pressure limit does not appreciably decrease the margin of safety in calculating fuel system damage.

RESPONSE 4.8

Fuel and poison rod internal pressure increases with increasing burnup and at end-of-life the total internal pressure, due to the combined effects of the initial helium fillgas and the released fission gas, is at a maximum. This maximum internal pressure is used in conjunction with other loads on the fuel rod cladding to calculate cladding stresses.

While there are no limits on internal gas pressure stated in GESTAR II or elsewhere, the maximum internal gas pressure actually is limited by the stress limits in GESTAR II consistent with the following criterion: the stress in the cladding resulting from differential pressure will not exceed the stress limits specified in Section 2.5 of GESTAR II.

GE has performed evaluations for P8x8R and barrier P8x8R using the GESTR(M) mechanical model, in which the fuel rod cladding creepout rate was calculated and compared with the fuel pellet irradiation swelling rate, during late life operation when the fuel rod internal pressure is highest. The results of this evaluation demonstrate that the cladding creepout rate is less than the fuel pellet irradiation swelling rate. This indicates that the fuel cladding gap is not expected to increase under the maximum planned normal operating conditions.

GE has accumulated significant operating experience with all fuel types. This experience includes the operation of lead test assemblies to higher than design exposures. Based on numerous post-irradiation inspections performed, it is concluded that the fuel is adequately designed relative to fuel rod internal pressure since cladding creepout due to fuel rod internal pressure has not been observed.

GE will resolve the internal pressure issue to the satisfaction of the NRC staff before the reference of GESSAR II by the first Applicant.

Responses To Questions On The Fuel Loading Error Analysis Basis Change (Amendment 28)

By letter dated May 17, 2004, Global Nuclear Fuel (GNF) submitted proposed amendment 28 to the GESTAR II. The proposed amendment would revise the GESTAR II based on compliance with the Standard Review Plan, Section15.4.7, "Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position".

The Nuclear Regulatory Commission (NRC) staff has reviewed the information GNF provided that supports proposed amendment 28 to the GESTAR II. In order for the staff to complete its evaluation, the following additional information is requested:

Transient Category

Question

1) In a September 27, 1993 RAI response to the September 30, 1992 application GE stated that from 1981-1993 there were 26 reported cases of rotated fuel bundles and in addition a June 23, 1995 submittal reported that in a 16 plant historical data base evaluation, 133,787 bundles were inserted with 2 mislocations overlooked by the bridge crew. How many more reportable misloaded (for the purpose of the RAIs misloaded fuel will mean mislocated and misoriented fuel errors unless specified otherwise) fuel bundles have occurred in foreign and domestic plants using GE fuel since the 1993 RAIs for rotated fuel and the 1995 submittal for mislocated fuel? What verification process is used to find these loading errors? How many times have plants operated at power with misloaded fuel?

Response

1)(General): The quotes from the September 27, 1993 RAI response and June 23, 1995 submittal quoted above were incomplete from the subject letters. In order to be clear about the meaning of the stated numerical values, the complete statements are provided. The complete statement from the 1993 RAI response is, "Since 1981 when SIL-347 was first implemented, there have been 26 reported cases of misoriented bundles being discovered and corrected by the verification process." The complete statement from the 1995 submittal is, "Of the 16 plants responding to a survey on mislocated bundles, there have been 133,787 bundle insertions to date with 2 mislocated not identified by the refueling bridge crew. However, these mislocated bundles were identified during subsequent core verification prior to power operation."

1)(a): How many more reportable misloaded (for the purpose of the RAIs misloaded fuel will mean mislocated and misoriented fuel errors unless specified otherwise) fuel bundles have occurred in foreign and domestic plants using GE fuel since the 1993 RAIs for rotated fuel and the 1995 submittal for mislocated fuel?

For the purposes of counting occurrences of a bundle not being properly loaded in the core, the mislocated and misoriented bundles may be counted together. However, they are different errors. The misoriented bundle is placed in its proper location, but by a single error is not placed in its correct orientation. The cases reported in the 1993 and 1995 references, in every instance, are misoriented bundles. Mislocating a bundle requires that two bundles be placed in incorrect locations. Two bundles may be switched within the core-loading pattern, or a bundle that should be loaded in the core may be left in the pool and an incorrect bundle loaded in the core. The error rate history supports the assertion that the likelihood of a mislocated bundle is much less than a misoriented bundle.

A survey was performed in July through November 2004 and reconfirmed in February 2005 to cover late 2004 outages. Table 1 summarizes the results of the 1995-2/2005 operating plant history regarding mislocated and misoriented fuel bundles.

Table 1 Summary of Misloaded Fuel Bundle Events 1995-2/2005

Plant	Discovered in loading verification	Operated with	Comments
Browns Ferry 2	None	None	
Browns Ferry 3	None	None	
Brunswick 1	None	None	See 1)(c) Below
Brunswick 2	None	None	
Clinton	None	None	
Cooper	1 misoriented ~2001	None	
Dresden 2	None	None	
Dresden 3	None	None	
Duane Arnold	1 misoriented	None	
Fitzpatrick	3 misoriented – 2000	None	
Grand Gulf	1 misloaded (2 Assemblies) – 1996	None	
Hatch 1	None	None	
Hatch 2	None	None	
Hope Creek	Non-Specific response	None	See 1)(c) Below
LaSalle 1	None	None	
LaSalle 2	None	None	
Limerick 1	None	None	
Limerick 2	None	None	
Monticello	None	None	See 1)(c) Below
Nine Mile 1	None	None	
Nine Mile 2	None	None	
Oyster Creek	None	None	
Peach Bottom 2	None	None	
Peach Bottom 3	None	None	
Pilgrim	None	None	
Quad Cites 1	None	None	
Quad Cities 2	None	None	
River Bend	1 or more misoriented – 1995	None	
Vermont Yankee	1 misoriented – 2001	None	

For the 29 plants that responded to the survey, 9 misloading errors were identified during core verification and corrected. During the 10 years from 1995 to 2005, there have been no misloaded bundles that were undetected by the core loading verification process.

1)(b): What verification process is used to find these loading errors?

Each operating BWR has its own procedure for core verification following fuel and core component movements prior to startup. The procedures follow the recommendations of SIL-347. While the focus of SIL-347 is misoriented bundles, the utility procedures have generally applied the SIL-347 recommendation of "at least 2 independent reviewers" to the each of the three core verification elements: bundle location, orientation, and seating.

During fuel movement, each move (location and orientation) is observed and checked at the time of completion by the operator and a spotter. After completion of the core load, the core is verified by videotaping the core using an underwater camera. The taping may involve two or more videotape runs at different ranges to provide clear resolution of the bundle serial number, and to illustrate the orientation in four bundle clusters. The core verification may take place during the taping process, by viewing of the tapes after taping, or a combination. Verification of the bundle serial number (location) and orientation is performed by independent reviewers. Each independent team records the bundle serial numbers on a core map, which is verified with the design core-loading pattern.

1)(c): How many times have plants operated at power with misloaded fuel?

Respondents to the 2004 survey identified three misoriented bundles in years prior to 1995 that were not detected during verification. The refueling year when the error occurred and plants are as follows:

plants are as follows:	1979	Brunswick Unit 1,
	1993	Monticello, and
	1994	Hope Creek.

These misoriented fuel bundles were operated for a complete cycle with no failures (leakers) occurring in the affected or adjacent fuel bundles. Brunswick Unit 1 operated with a bundle rotated 180 degrees in 1979. The error was discovered after startup by viewing the core verification videotape. Continued operation was justified and NRC approved Brunswick Unit 1 operation with the rotated bundle. Hope Creek and Monticello identified the errors at fuel off-loading at the next outage.

There have been no confirmed mislocated fuel bundles that operated in the core for the entire BWR history.

2) The misloaded fuel bundle event is caused solely by operator error and is currently classified as an anticipated operational occurrence (AOO) and therefore is analyzed every fuel cycle. What has changed since GE's previous submittals that would prove that the probability of a misloaded fuel bundle event has significantly decreased and should be

reclassified as an infrequent incident and should no longer be covered by General Design Criteria 10? Include in your answer how the changes in refueling schedules due to shorter outages and reliance on contractor personnel are included into the probability.

Response

Following is quoted from Section 6.3 of the GESTAR Rev 0 SER May 12, 1978

> The reload topical does not classify the fuel loading error (FLE) as an abnormal operational transient. Actual fuel loading experience at operating BWR's has lead the staff to conclude that the frequency of occurrence of a fuel loading error is an event which can occur approximately once per plant lifetime. A fuel loading error could also result in fuel degradation were it to occur, go undetected, and violate the safety limit MCPR. Accordingly, we have required that the fuel loading error be considered as a transient for reload safety analyses. Moreover, a fuel loading error can give rise to significant increases in bundle R-Factor and bundle power and hence a significant change in CPR.
>
> As discussed in Section 6.2.1, GE has proposed a safety limit of 1.0 for this event. This proposal is being reviewed separately by the staff. In the interim, we will continue to require that the fuel loading error be reanalyzed as a limiting local transient event which does not violate the safety limit MCPR.
>
> In summary, then, we conclude that the events appearing in Section 5.2.1 of Reference 3 together with the fuel loading error, provide a sufficient set of abnormal operational transients to be reanalyzed for each plant-specific BWR reload licensing application.

Therefore, although GE/GNF originally classified the FLE as an infrequent incident, since 1978 it has been analyzed as an AOO.

The data presented in Table 1 supports the classification of the FLE event as an "infrequent incident" per the NRC guidelines in Regulatory Guide 1.70 (RG 1.70), Revision 3, November 1978. Section 15.X.X (page 15-4) of the RG 1.70 defines an Infrequent Incident as one that may occur once in a plant's lifetime. Because RG 1.70 was prepared in a time period when the lifetime was limited by the ASME vessel lifetime of 40 years, the plant lifetime used in the mathematics below is selected as 40 years. In order to compare this frequency specification with the observed number of errors reported in the 1)(c) Response, the number of expected errors for the 29 plants included in Table 1 is determined from the RG 1.70 definition as

1 FLE per Plant per Lifetime * 29 Plants / 40 year Lifetime = 0.725 FLE / year

For the 25-year period from 1980 to 2005, which spans the 3 reported uncorrected fuel loading errors, the RG 1.70 Infrequent Incident definition would allow 0.725 FLE/year*25 years or 18.125 errors. In other words, the 3 uncorrected errors that actually occurred are less than 20% of the threshold of the RG 1.70 definition for the Infrequent Incident.

Using the same numbers, the error rate experience for the 25-year period may be compared to the RG 1.70 definition as follows:

$$\frac{3 \text{ Errors}}{29 \text{ Plants} * 25 \text{ Years}} * \frac{40 \text{ Years}}{\text{Lifetime}} » 0.17 \text{ FLE per Plant per Lifetime}$$

The FLE error rate experience of ≈ 0.17 FLE per Plant per Lifetime is compared to the RG 1.70 defined error rate of 1 FLE per Plant per Lifetime.

From refueling outage data for every operating domestic BWR from 1980 to date, the approximate number of fuel bundles moved in the 1980 to 1995 period, and from 1995 to 2005 were determined. The following table is a summary of number of refueling outages, approximate number of bundle movements, and error rate.

Time Period	Outages	Moves	Uncorrected Loading Errors
1980 to Jun 1995	241	148020	3
Jun 1995 to Jan 2005	179	118684	0

There are several conclusions that can be drawn from this data:

1. The number of fuel bundle moves before and after June 1995 are comparable and, therefore, can be used to compare the effectiveness of the core loading verification processes and procedures for these periods.
2. In the post 1995 period, the core loading verification processes and procedures are being more effectively applied as compared to the period prior to 1995.
3. The industry trend to shorter refueling outages and any change in the mix of employees versus contractors used during refueling outages has not created an adverse trend in fuel loading errors. The industry trend toward longer cycles and fewer outages is a benefit to the overall probability of a FLE.

The error rate for the 25-year period and the trend for the most recent 10 years of refueling outages support the classification of the FLE as an "Infrequent Incident."

Thermal Limits

3) Your submittal states that fuel loading error consequence would be perforations in a small number rods in the misplaced bundle, and the resulting fission product will be detected by the offgas system. However, if a mislocated or misoriented fuel bundle remains in the core, despite the event categorization, any transient event (e.g. FWCF or TTWNBP) response could lead to higher delta CPR and exceeding the SLMCPR. This would result in violating

GDC 10. Explain if the GE methodology uses a multiplier to the OLMCPR in order to incorporate the impact of operating with a misloaded bundle(s). If not, justify why the delta CPR due to fuel loading error should not be included in the calculations of the OLMCPR value based on other limiting transients.

Response

The responses to Transient Category Question 1 and 2 provide the justification for the classification of the FLE as an Infrequent Incident. With the re-classification of the FLE as an Infrequent Incident, the questions pertaining to the event as an AOO and the associated MPCR criteria are not applicable.

4) As an AOO, the fuel loading error (FLE) is analyzed every reload as one of the limiting transients. S.2.2.3.6, "Mislocated bundle Accident," in GESTAR II describes the possible mislocated bundle errors. However, in order to meet the current operating conditions, the core design strategies have changed since the establishments of the mislocated and misoriented fuel bundles analysis methodologies. The following questions assess the mislocated and misoriented fuel bundle analyses methodology as it relates to establishing the transient CPR.

a) Provide an evaluation that demonstrates if the mislocated fuel bundle scenarios used to establish the CPR represent limiting core configurations that are consistent with the current core design strategies. For example, does the current plant and cycle-specific mislocated fuel bundle transient event consider the potential for loading high enrichment bundle in core cells already containing two similarly hot bundles.

b) Identify the important parameters or conditions that would have the most impact on the FLE response. For example, what impact would the following conditions have on the FLE response: high enrichment, high GD loading, cycle exposure, exposure of the bundles in the control cell with the mislocated fuel bundle, the configuration and type of bundles in the surrounding cells.

c) What affects, if any, would a mislocated fuel bundle have on the CPR response for the core designs necessary for achieving the current operating strategies. State if the FLE transient could establish the limiting CPR response.

d) Similarly, discuss questions a, b, and c using the misoriented fuel bundle event.

Response

The responses to Transient Category Question 1 and 2 provide the justification for the classification of the FLE as an Infrequent Incident. With the re-classification of the FLE as an Infrequent Incident, the questions pertaining to the event as an AOO and the associated MPCR criteria are not applicable.

5) Justify why an event that could establish the limiting transient CPR response would be categorized as an accident as proposed in the amendment request.

Response

RG 1.70 categorizes events according to their frequency of occurrence. The definitions from RG 1.70 are as follows:

1. Incidents of Moderate Frequency - these are incidents, any one of which may occur during a calendar year for a particular plant.
2. Infrequent Incidents - these are incidents, any one of which may occur during the lifetime of a particular plant.
3. Limiting Faults - these are occurrences that are not expected to occur, but are postulated because their consequences would include the potential for the release of significant amounts of radioactive material.

The distinction between the Incidents of Moderate Frequency (AOOs) category and Infrequent Incidents category is purely one of frequency of occurrence.

The U.S. NRC Standard Review Plan 15.4.7, Inadvertent Loading And Operation Of A Fuel Assembly In An Improper Position, Draft Rev. 2, 1996 defines the FLE as an infrequent incident. This SRP also defines the acceptance criteria for the consequences of the event as a small fraction, which is characterized as 10%, of 10CFR100. The following quote is extracted from Section II, Acceptance Criteria, of SRP 15.4.7:

> 10 CFR Part 100 is applicable to SRP Section 15.4.7, because it specifies the methodology for calculating radiation exposures at the site boundary for postulated accidents or events that might be caused by a fuel-loading error. For events having a moderate-frequency of occurrence, any release of radioactive material must be such that the calculated doses at the site boundary are a small fraction of the 10 CFR Part 100 guidelines. A small fraction is interpreted to be less than 10% of the 10 CFR Part 100 reference values. For the purpose of this review, the radiological consequences of any fuel-loading error must include consideration of the containment, confinement, and filtering systems. The applicant's source terms and methodologies with respect to gap release fractions, iodine chemical form, and fission product release timing should reflect NRC-approved source terms and methodologies.

6) It is a historical staff position that the fuel loading error (FLE) has the potential to be a significant transient. For this reason the GESTAR II requires the FLE be analyzed as an AOO. The Staff's historical position as found in the safety evaluation report (SER) for GESTAR II, Rev.0, dated May 12, 1978, Section 6.3 "Transient Events Analyzed for Reloads" on page US.C-43 is as follows, "The reload topical does not classify the fuel loading error (FLE) as an abnormal operational transient. Actual fuel loading experience at operating BWR's has lead the staff to conclude that the frequency of occurrence of a fuel

loading error is an event which can occur approximately once per plant lifetime. A fuel loading error could also result in fuel degradation were it to occur, go undetected, and violate the safety limit MCPR. Accordingly, we have required that the fuel loading error be considered as a transient for reload safety analyses. Moreover, a fuel loading error can give rise to significant increases in bundle R-Factor and bundle power and hence a significant change in CPR." The staff strongly agrees with the explanation of the handling of the FLE in the1978 SER for the following reasons:

* Misloaded fuel bundles do occur. Undetected operation with a misloaded fuel bundle has occurred at Hope Creek. The event at Hope Creek was reported voluntarily. Since a FLE event is reported and docketed on a voluntary basis, the staff is unable to verify whether or not other FLE events have occurred. The industry trend towards minimizing outage times, to the shortest time possible, could place more pressure on the reloading personnel. This could lead to the probability of a FLE event increasing due to the potential for the misloaded bundle to go undetected. Thus it is a staff position that operation with a misloaded fuel bundle can occur.

* Core designs and operating conditions have changed such that the consequences of operating with a misloaded fuel bundle is becoming more severe in terms of the MCPR response during steady state operation. Extended power uprates and longer fuel cycles are causing the consequences of the FLE event to increase. In the core designs needed to fulfill the current operating strategies, the fuel enrichment, batch fraction and number of high powered bundles required every cycle have increased. Thus a FLE event could potentially place three hot bundles in a control rod cell. For these reasons the changes in fuel design are causing the CPR response for the FLE transient to become more limiting, thus more likely to establish the cycle specific OLMCPR. Therefore, it is a staff position that the MCPR response of the FLE transient needs to be calculated and analyzed every cycle since it has the potential to become the limiting transient.

* If a FLE goes undetected and the plant operates with a misloaded fuel bundle, the SLMCPR calculations and the SLMCPR value can become invalidated. If the limiting transient were to occur, a properly calculated OLMCPR value will offset the potential change in the SLMCPR calculations and this offset is what ensures the SLMCPR value will remain valid. If the calculation for the limiting transient was no longer performed, the OLMCPR offset for this transient would cease to exist and the SLMCPR calculation and SLMCPR value may not be protected for the transient. Therefore, it is a staff position that the cycle-specific CPR response for the limiting transient (which could be the FLE) must be used in the OLMCPR calculation so that a valid offset exists to ensure the SLMCPR calculations and the SLMCPR value remain valid if the limiting transient were to occur.

* A detailed PRA analyses has never been conducted to assess the probability of a FLE event based on the refueling practices that are currently performed and allowed by the current Technical Specifications.

* A preliminary FSAR survey indicated that a majority of the plants are required to analyze the FLE on a cycle specific bases to establish the CPR response.

After taking these concerns into consideration the staff is unable to find suitable safety findings to support the proposed changes. The staff finds that this amendment request did not provide sufficient bases to reverse the staff's historical position that the FLE should be analyzed on a cycle-specific basis as a limiting reactivity initiated event. In order for the staff to consider your proposed changes for acceptability, the following is needed:

1) Detailed information explaining the hardware, software, mechanical interlocks, etc (administrative procedures don't count) that prevent misloading errors from occurring.

2) Detailed analysis proving that the CPR value for the FLE transient is negligible now and how the value will remain negligible in the future. In your response provide typical CPR values that are used for establishing the OLMCPR.

3) Detailed PRA analysis that proves the FLE is an accident and not an AOO. Include a chart or table showing the historical data used in determining the event probability.

Response

The responses to Transient Category Question 1 and 2 provide the justification for the classification of the FLE as an Infrequent Incident. With the re-classification of the FLE as an Infrequent Incident, the questions pertaining to the event as an AOO and the associated MPCR criteria are not applicable. Responses to specific relevant questions follow.

* A detailed PRA analyses has never been conducted to assess the probability of a FLE event based on the refueling practices that are currently performed and allowed by the current Technical Specifications.

A detailed PRA model was submitted with the September 1992 submittal, MFN-183-92. In the June 1995 submittal, MFN-066-95, this model and the predicted probabilities were updated. However, the extensive period of refueling history as reflected in the responses to Transient Category Questions 1 and 2 makes a PRA model of limited value. In other words, there is no particular information provided by a model that is not reflected in the actual refueling data for the past 25 years.

1) Detailed information explaining the hardware, software, mechanical interlocks, etc (administrative procedures don't count) that prevent misloading errors from occurring.

The response to Transient Category Question 1 presented the loading and verification procedures. The overall error rate for the past 25 years, and in particular, the zero error rate for the past 10 years, demonstrates the effectiveness of the procedures.

The following 2 dose questions were provided by Letter from Andrew A. Lingenfelter, (GNF) to Document Control, (US NRC), Subject: Response to NRC Request for Additional Information Regarding Amendment 28 to GESTAR II (TAC NO. MC3559), FLN-2006-018, May 11, 2006.

NRC Question

1. For accident Scenario 1, GNF assumed that fission products from all fuel rods in five failed fuel assemblies were released to the turbine and condensers based on a power level of 5.75 per bundle. Please provide the fission product source term after applying a safety factor of 1.4 and a radial peaking factor of 2.5 as you proposed. State if you assumed the same source term for accident Scenario 2.

GE Response

The fission product source term (before applying any of the release fractions in Attachment B Table B-1) is provided below:

Table 1 – Scenario 1 and 2 Fission Product Source Term

Isotope	Source Term (Ci)
I-128	3.26E+04
I-129	1.00E-01
I-130	8.61E+04
I-131	2.64E+06
I-132	3.81E+06
I-133	5.42E+06
I-134	5.96E+06
I-135	5.07E+06
Kr-83m	3.37E+05
Kr-85	3.34E+04
Kr-85m	7.18E+05
Kr-87	1.38E+06
Kr-88	1.95E+06
Kr-89	2.39E+06
Kr-90	2.36E+06
Kr-91	1.75E+06
Kr-92	8.47E+05
Xe-131m	2.94E+04
Xe-133	5.43E+06
Xe-133m	1.69E+05
Xe-135	1.84E+06
Xe-135m	1.05E+06
Xe-137	4.73E+06
Xe-138	4.50E+06
Xe-139	3.53E+06

The source term in the table above is the same source term that was used for Scenario 2.

NRC Question

2. For accident Scenario 2, please provide the amount of fission products released from the offgas system, resulting doses, and relevant dose calculations for several representative charcoal holdup times in Figures B-2, B-3, B-5, and B-6 of Attachment B.

GE Response

As discussed in Attachment B Section B.4.2.2, iodine is not considered in Scenario 2 due to the retention in the offgas charcoal beds. The remaining fission products available at the inlet to the offgas system are as follows:

Table 2 – Scenario 2 Fission Product Source Term at Offgas System Inlet

Isotope	Source Term (Ci)
Kr-83m	3.37E+04
Kr-85	3.34E+03
Kr-85m	7.18E+04
Kr-87	1.38E+05
Kr-88	1.95E+05
Kr-89	2.39E+05
Kr-90	2.36E+05
Kr-91	1.75E+05
Kr-92	8.47E+04
Xe-131m	2.94E+03
Xe-133	5.43E+05
Xe-133m	1.69E+04
Xe-135	1.84E+05
Xe-135m	1.05E+05
Xe-137	4.73E+05
Xe-138	4.50E+05
Xe-139	3.53E+05

A sample calculation is provided to show how the curves in Attachment B Figures B-2 and B-3 were generated. The dose conversion factors that were used are as follows:

Table 3 – Scenario 2 Dose Conversion Factors (DCFs)

Isotope	Whole Body DCF ($Rem\text{-}m^3/Ci\text{-}sec$)	TEDE DCF ($Sv\text{-}m^3/Bq\text{-}sec$)
Kr-83m	6.44E-04	1.50E-18
Kr-85	5.58E-04	1.19E-16
Kr-85m	3.94E-02	7.48E-15
Kr-87	1.98E-01	4.12E-14
Kr-88	4.89E-01	1.02E-13
Kr-89	4.59E-01	N/A
Kr-90	3.18E-01	N/A
Kr-91	1.81E-01	N/A
Kr-92	1.88E-01	N/A
Xe-131m	5.03E-03	3.89E-16
Xe-133	1.13E-02	1.56E-15
Xe-133m	1.04E-02	1.37E-15
Xe-135	6.20E-02	1.19E-14
Xe-135m	1.08E-01	2.04E-14
Xe-137	4.69E-02	N/A
Xe-138	2.82E-01	5.77E-14
Xe-139	2.32E-01	N/A

The Attachment B Figure B-2 and B-3 whole body doses are calculated using the following formula:

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The following table provides a sample calculation for the krypton whole body dose, assuming a X/Q of 1.0E-04 sec/m^3 and an offgas holdup time of 40 hours:

Table 4 – Attachment B Figure B-2 Sample Calculation

Isotope	λ (sec^{-1})	Whole Body Dose (Rem)
Kr-83m	1.035E-04	7.284E-10
Kr-85	2.047E-09	1.865E-04
Kr-85m	4.298E-05	5.811E-04
Kr-87	1.520E-04	8.553E-10
Kr-88	6.876E-05	4.771E-04
Kr-89	3.656E-03	0.0
Kr-90	2.146E-02	0.0
Kr-91	7.967E-02	0.0
Kr-92	3.767E-01	0.0
	Total	**1.24E-03**

The resulting Kr whole body dose for a X/Q of 1.0E-04 sec/m^3 and an offgas holdup time of 40 hours is 1.24E-03 Rem, which is consistent with the applicable curve in Attachment B Figure B-2.

The following table provides a sample calculation for the xenon whole body dose, assuming a X/Q of 3.0E-04 sec/m^3 and an offgas holdup time of 40 days:

Table 5 – Attachment B Figure B-3 Sample Calculation

Isotope	λ (sec^{-1})	Whole Body Dose (Rem)
Xe-131m	6.691E-07	4.383E-04
Xe-133	1.517E-06	9.758E-03
Xe-133m	3.598E-06	2.088E-07
Xe-135	2.100E-05	1.044E-31
Xe-135m	7.551E-04	0.0
Xe-137	3.008E-03	0.0
Xe-138	8.136E-04	0.0
Xe-139	1.716E-02	0.0
	Total	**1.02E-02**

The resulting Xe whole body dose for a X/Q of 3.0E-04 sec/m^3 and an offgas holdup time of 40 days is 1.02E-02 Rem, which is consistent with the applicable curve in Attachment B Figure B-3.

The Attachment B Figure B-5 and B-6 TEDE doses are calculated using the following formula:

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The following table provides a sample calculation for the krypton TEDE dose, assuming a X/Q of 1.0E-04 sec/m^3 and an offgas holdup time of 40 hours:

Table 6 – Attachment B Figure B-5 Sample Calculation

Isotope	λ (sec^{-1})	TEDE Dose (Rem)
Kr-83m	1.035E-04	6.280E-12
Kr-85	2.047E-09	1.472E-04
Kr-85m	4.298E-05	4.079E-04
Kr-87	1.520E-04	6.576E-10
Kr-88	6.876E-05	3.684E-04
Kr-89	3.656E-03	0.0
Kr-90	2.146E-02	0.0
Kr-91	7.967E-02	0.0
Kr-92	3.767E-01	0.0
	Total	**9.24E-04**

The resulting Kr TEDE dose for a X/Q of 1.0E-04 sec/m^3 and an offgas holdup time of 40 hours is 9.24E-04 Rem, which is consistent with the applicable curve in Attachment B Figure B-5.

The following table provides a sample calculation for the xenon TEDE dose, assuming a X/Q of 3.0E-04 sec/m^3 and an offgas holdup time of 35 days:

Table 7 – Attachment B Figure B-6 Sample Calculation

Isotope	λ (sec^{-1})	Whole Body Dose (Rem)
Xe-131m	6.691E-07	1.676E-04
Xe-133	1.517E-06	9.578E-03
Xe-133m	3.598E-06	4.832E-07
Xe-135	2.100E-05	6.448E-28
Xe-135m	7.551E-04	0.0
Xe-137	3.008E-03	0.0
Xe-138	8.136E-04	0.0
Xe-139	1.716E-02	0.0
	Total	**9.75E-03**

The resulting Xe TEDE dose for a X/Q of 3.0E-04 sec/m^3 and an offgas holdup time of 35 days is 9.75E-03 Rem, which is consistent with the applicable curve in Attachment B Figure B-6.

RESPONSES TO QUESTIONS ON ADDITIONAL AMENDMENTS (References)

AMENDMENT 7

1. Letter from J.S. Charnley (GE) to C.O. Thomas (NRC), "Response to Request for Additional Information on Proposed Amendment to GE Licensing Topical Report NEDE-24011-P-A," MFN-230-83, December 19, 1983".

2. Letter from J.S. Charnley (GE) to C.O. Thomas (NRC), "Response to Request Number One for Additional Information on NEDE-20411, Revision 6, Amendment 7," MFN-050-84, April 23, 1984.

AMENDMENT 8

1. Letter from H.C. Pfefferlen (GE) to C.O. Thomas (NRC), "Response to Request Number One for Additional Information on NEDE-20411, Revision 6, Amendment 8," MFN-178-84/009-85, January 14, 1985.

AMENDMENT 9

No questions submitted.

AMENDMENT 10

1. Letter from J.S. Charnley (GE) to C.O. Thomas (NRC), "Response to Request Number One for Additional Information on NEDE-24011-P-A-6, Amendment 10," MFN-035-85, March 11, 1985.

2. Letter from J.S. Charnley (GE) to C.O. Thomas (NRC), "Response to Request Number Two for Additional Information on NEDE-24011-P-A-6, Amendment 10," MFN-053-85, April 26, 1985.

3. Letter from J.S. Charnley (GE) to C.O. Thomas (NRC), "Supplementary Information Regarding NEDE-24011-P-A-6, Amendment 10," MFN-060-85, May 2, 1985.

4. Letter from J.S. Charnley (GE) to C.O. Thomas (NRC), "Supplementary Information Regarding Use of SAFE/REFLOOD to High Exposures," MFN-132-85, October 31, 1985.

AMENDMENT 11

1. Letter from J.S. Charnley (GE) to C.O. Thomas (NRC), "Response to Request Number One for Additional Information on NEDE-24011, Revision 6, Amendment 11," MFN-094-85, July 18, 1985.

AMENDMENT 12

No questions submitted.

AMENDMENT 13

No questions submitted.

AMENDMENT 14

1. Letter from J.S. Charnley (GE) to G.C. Lainas (NRC), "Response to NRC Request for Additional Information for Amendment 14 to General Electric Licensing Topical Report NEDE-24011-P-A," MFN-032-86, May 7, 1986.

AMENDMENT 15

1. Letter from J.S. Charnley (GE) to G.C. Lainas (NRC), "Response to NRC Request for Additional Information for Amendment 15 to GE Licensing Topical Report NEDE-24011-P-A," MFN-057-086, dated July 11, 1986.

2. Letter from J.S. Charnley (GE) to G.C. Lainas (NRC), "Response to Second NRC Request for Additional Information for Amendment 15 to GE Licensing Topical Report NEDE-24011-P-A," MFN-008-087, dated January 16, 1987.

AMENDMENT 16

1. Letter from J.S. Charnley (GE) to M.W. Hodges (NRC), "Response to Reviewer Comments on GESTAR Amendment 16," MFN-111-087, dated November 11, 1987.

AMENDMENT 17

No questions submitted.

AMENDMENT 18

1. Letter from J.S. Charnley (GE) to M.W. Hodges (NRC), "Additional Information Pertaining to NRC Review of GESTAR Amendment 18," MFN-055-087, August 4, 1987.

2. Letter from J.S. Charnley (GE) to M.W. Hodges (NRC), "Requested Information for Amendment 18," MFN-115-87, December 10, 1987.

3. Letter (and enclosure) from J.S. Charnley (GE) to G. Lainas (NRC), "Additional Information Pertaining to Proposed Amendment 18 to NEDE-24011-P-A," MFN-003-087, dated January 29, 1987.

4. Letter (and attachment) from J.S. Charnley (GE) to M.W. Hodges (NRC), Pertaining to GESTAR Amendment 18 (GE8x8NB Fuel), MFN-065-087, dated October 14, 1987.

5. Letter from J.S. Charnley (GE) to M.W. Hodges (NRC), "Response to Questions on GESTAR-II Amendment 18," MFN-005-088, dated January 13, 1988.

6. Letter from J.S. Charnley (GE) to M.W. Hodges (NRC), "Additional Information on GESTAR-II Amendment 18," MFN-012-088, dated February 12, 1988.

AMENDMENT 19

1. Letter from J.S. Charnley (GE) to M.W. Hodges (NRC), "Supplementary Information Regarding GE Special Report MFN-106-85," MFN-085-86, dated September 9, 1986.

2. Letter from J.S. Charnley (GE) to M.W. Hodges (NRC), "Response to Request for Additional Information on GE Special Report MFN-106-85," MFN-004-087, dated January 14, 1987.

3. Letter from J.S. Charnley (GE) to M.W. Hodges (NRC), "Revised Response to Question 3 of Request for Additional Information on GE Special Report MFN-106-85," MFN-025-87, dated March 12, 1987.