GE Nuclear Energy

General Electric Company
P. O. Box 780, Wilmington, NC 28402

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Methodology and Uncertainties for Safety Limit MCPR Evaluations

Disclaimer of Responsibility

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

March 11, 1999

MFN-003-99

Mr. Glen A. Watford, Manager
General Electric Company
P.O. Box 780
Wilmington, NC 28402

SUBJECT: ACCEPTANCE FOR REFERENCING OF LICENSING TOPICAL REPORTS NEDC-32601P, METHODOLOGY AND UNCERTAINTIES FOR SAFETY LIMIT MCPR EVALUATIONS; NEDC-32694P, POWER DISTRIBUTION UNCERTAINTIES FOR SAFETY LIMIT MCPR EVALUATION; AND AMENDMENT 25 TO NEDE-24011-P-A ON CYCLE-SPECIFIC SAFETY LIMIT MCPR (TAC NOS. M97490, M99069 AND M97491)

Dear Mr. Watford:

The staff has reviewed the subject reports submitted by GE Nuclear Energy (GENE) by letters dated December 13, 1996, for NEDC-32601P; June 10, 1997, for NEDC-32694P; and December 13, 1996, for Amendment 25 to NEDE-24011P. These submittals provide (1) the description of the procedures used to account for the reload-specific core design and operation in determining the cycle-specific safety limit minimum critical power ratio (SLMCPR) in NEDC-32601P; (2) the power distribution uncertainty for the new GE 3D-MONICORE core surveillance system in NEDC-32694P; and (3) the methodology and uncertainties required for the implementation of cycle-specific SLMCPR in Amendment 25 to NEDE-24011-P-A. The staff has found the subject reports to be acceptable for referencing in license applications to the extent specified and under the limitations stated in the GENE letter dated March 1, 1999, the enclosed report, and the U. S. Nuclear Regulatory Commission (NRC) technical evaluation. The evaluation defines the basis for acceptance of the report.

The staff will not repeat its review of the matters described in the GENE Topical Reports NEDC-32601P, NEDC-32694, and Amendment 25 to NEDE-24011-P-A and found acceptable when this letter request appears as a reference in license applications, except to ensure that the material presented applies to the specific plant involved. NRC acceptance applies only to the matters described in the GENE Topical Reports NEDC-32601P, NEDC-32694P, and Amendment 25 to NEDE-24011-P-A. In accordance with procedures established in NUREG-0390, the NRC requests that GE publish accepted versions of the submittals, proprietary and non-proprietary, within 3 months of receipt of this letter. The accepted versions shall incorporate this letter and the enclosed evaluation between the title page and the abstract and an -A (designating accepted) following the report identification symbol.

If the NRC's criteria or regulations change so that its conclusions that the submittal is acceptable are invalidated, GE and/or the applicant referencing the submittal will be expected to revise and resubmit its respective documentation, or submit justification for the continued applicability of the submittal without revision of the respective documentation.

Sincerely,

Frank Akstulewicz

Frank Akstulewicz, Acting Chief
Generic Issues and Environmental Project Branch
Division of Reactor Program Management
Office of Nuclear Reactor Regulation

Enclosures:
NEDC-32601P, NEDC-32694P, and Amendment 25 to NEDE-24011-P-A Evaluation

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ENCLOSURE 1

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION RELATING TO GENERAL ELECTRIC LICENSING TOPICAL REPORTS NEDC-32601P, NEDC-32694P, AND AMENDMENT 15 to NEDE-24011-P-A

1. INTRODUCTION

By letters dated December 13, 1996, June 10, 1997, and December 13, 1996, from R. J. Reda (GE) to USNRC, General Electric Nuclear Energy (GENE) submitted licensing topical reports: NEDC-32601P, 'Methodology and Uncertainties for Safety Limit MCPR Evaluation" (Reference 1); NEDC-32694P, "Power Distribution Uncertainties for Safety Limit MCPR Evaluation," (Reference 2); and Amendment 25 to NEDE-24011-P-A, Proposed Amendment 25 to GE Licensing Topical Report NEDE-24011-P-A (GESTAR II) on Cycle-Specific Safety Limit MCPR," (Reference 3), respectively. The purpose of the submittal is (1) for NEDC32601P to update values of the CPR correlation uncertainties contained in NEDE-10958-P-A (GETAB, Reference 4) based on the most recent analysis of available data; (2) for NEDC32694P to update values of the power distribution uncertainties contained in NEDE-31152P, Revision 5 based on the most recent analysis of available data, and (3) for Amendment 25 to NEDE-24011-P-A to provide for cycle-specific Safety Limit Minimum Critical Power Ratios (MCPRs).

The NRC staff was assisted in this review by its consultant, Brookhaven National Laboratory (BNL). The NRC staff's evaluation includes those three topical reports and the responses to staff's Request for Additional Information (RAI) dated January 8,1998 (GA W-98-002, MFN-004-98, Reference 5), January 9,1998 (GAW-98-003, MFN-005-98, Reference 6), January 28, 1998 (GAW-98-005, MFN-008-98, Reference 7), April 17, 1998 (GAW-98-009, Reference 8), and July 29, 1998 (GAW-98-012, MFN-017-98, Reference 9). The staff adopted the findings recommended in our consultant's Technical Evaluation Report (Enclosure 2).

2 EVALUATION

This review includes three topical reports involving the Safety Limit Minimum Critical Power Ratio (SLMC PR) methodology and input uncertainties described in NEDC-32601 P, the methodology for constructing the bounding statepoint power distribution described in NEDC-32694P, and the overall procedures for determining the cycle-specific SLMCPR described in Amendment 25 to GESTAR II. The details of the evaluation are provided in Enclosure 2.

2.1 Methodology and Uncertainties for Safety Limit MCPR Evaluation (NEDC-32601P)

The topical report, NEDC-32601 P, provides an update to the Safety Limit MCPR methodology and inputs to be used in the evaluation of the Safety Limit MCPR for BWRs (GETAB, Reference 4) including plant surveillance measurement uncertainties and local R-Factor uncertainties. The plant surveillance component uncertainties include the reactor pressure, feedwater temperature and flow, core inlet temperature and flow, and channel flow area and friction factors. The plant surveillance uncertainty revisions are based on current BWR practice, and are generally evaluated using the error methodology (Reference 10). The

R-Factor provides the critical power dependence on the local pin power distribution (References 4 and 11) in the GEXL correlation. The R-factor uncertainty analysis includes an allowance for power peaking modeling uncertainty, manufacturing uncertainty and channel bow uncertainty.

Based on the review of the NEDC-32601 P topical report and the responses to the staff's request for additional information (RAI) (References 5, 8, and 9), we find the SLMCPR methodology and associated uncertainties to be acceptable, however, actions should be taken as follows:

(1) The TGBLA fuel rod power calculational uncertainty should be verified when applied to fuel designs not included in the benchmark comparisions of Table 3.1 of Reference 1, since changes in fuel design can have a significant effect on calculation accuracy.

(2) The effect of the correlation of rod power calculation uncertainties should be reevaluated to insure the accuracy of the R-Factor uncertainty when the methodology is applied to a new fuel lattice.

(3) In view of the importance of MIP criterion and its potential sensitivity to changes in fuel bundle designs, core loading and operating strategies, the MIP criterion should be reviewed periodically as part of the procedural review process to insure that the specific value recommended in NEDC-32601p is applicable to future designs and operating strategies.

2.2 Power Distribution Uncertainties for Safety Limit MCPR Evaluations (NEDC-32694P)

The power distribution uncertainty topical report NEDC-32694P provides a description of the 3D-MONICORE core surveillance system and the determination of the associated bundle power uncertainty for use in SLMCPR calculation. The 3D-MONCORE system uses three-dimensional coarse-mesh diffusion theory methods, together with models for interfacing with the incore TIP and LPRM instrumentation, to determine the detailed core statepoint. The uncertainty in the 3D-MON ICORE prediction of bundle power was determined by comparisions of measured and calculated TIP integrals and gamma scanned bundle powers.

Based on the review of Reference 2 and the responses to the staff's RAI (References 6 and 8) we have found that the 3D-MONICORE power distribution uncertainties are acceptable for determining the SLMCPR, MAPLHGR and LHGR core limits, however, the 3D-MONICORE bundle power calculational uncertainty should be verified when applied to fuel and core designs not included in the benchmark comparisons of Tables 3.1 and 3.2 of Reference 2.

2.3 Amendment 25 to GE Licensing Topical Report NEDE-24011-P-A (GESTAR II) on Cycle - Specific Safety Limit MCPR

Amendment 25 to GESTAR II provides the methodology and uncertainties required for the implementation of cycle-specific Safety Limit MCPRs that replace the former generic, bounding SLMCPR. General procedures are given describing the analysis to be used in determining the cycle-specific SLMCPR. These procedures require that the analysis be performed for the specific fuel bundle design and core loading used in the cycle reload design.

Based on the review of References 3 and 7, we have found that the proposed methodology to be acceptable for performing cycle-specific SLMCPR analyses.

3 CONCLUSION

Based on our review of Topical Reports NEDC-32601P, NEDC-32694P, and Amendment 25 to NEDE-2401 1-P-A (GESTAR II), the staff concludes that the input plant system uncertainties, the power distribution uncertainties associated with the application of 3D-MON ICORE, and the proposed cycle-specific determination of the SLMCPR are acceptable. In letter dated PM TO SUPPLY, GENE has stated that they will take the following actions whenever a new fuel design is introduced.

(1) The TGBLA fuel rod power calculational uncertainty should be verified when applied to fuel designs not included in the benchmark comparisions of Table 3.1 of NEDC-32601P, since changes in fuel design can have a significant effect on calculation accuracy.

(2) The effect of the correlation of rod power calculation uncertainties should be reevaluated to insure the accuracy of R-Factor uncertainty when the methodology is applied to a new fuel lattice.

(3) In view of the importance of MIP criterion and its potential sensitivity to changes in fuel bundle designs, core loading and operating strategies, the MIP criterion should be reviewed periodically as part of the procedural review process to insure that the specific value recommended in NEDC-32601 P is applicable to future designs and operating strategies.

(4) The 3D-MON ICORE bundle power calculational uncertainty should be verified when applied to fuel and core designs not included in the benchmark comparisions in Tables 3.1 and 3.2 of NEDC-32694P.

4 REFERENCES

1. GE Letter RJR-96-139 MFN-185-96 dated December 13, 1996 from R. J. Reda to USNRC transmitting a topical report, NEDC-32601 P, "Methodology and Uncertainties for Safety Limit MCPR Evaluations," December 1996.

2. GE Letter RJR-97-074 MFN-022-97 dated June 10, 1997 form R. J. Reda to USNRC transmitting a topical report, NEDC-32694P, "Power Distribution Uncertainties for Safety Limit MCPR Evaluations," January 1997.

3. GE Letter RJR-96-133 MFN-179-96 from R. J. Reda to USNRC, "Proposed Amendment 25 to GE Licensing Topical Report NEDE-2401 1-P-A (GESTAR II) on Cycle Specific Safety Limit MCPR," December 13, 1996.

4. General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation and Design Application, NEDE-10958-PA, January 1977.

5. GE Letter GAW-98-002 MFN-004-98 from Glen A. Watford to USNRC, Responses to Request for Additional Information for GE Topical Report NEDC-32601 P, Methodology and Uncertainties for Safety Limit MCPR Evaluations, January 8,1998.

6. GE Letter GAW-98-003 MFN-005-98 from Glen A. Watford to USNRC, Responses to Request for Additional Information for GE Topical Report NEDC-32694P, Power Distribution Uncertainties for Safety Limit MCPR Evaluations, January 9,1998.

7. GE Letter GAW-98-005 MFN-008-98 from G. A. Waterford to USNRC, Responses to Request for Additional Information for Amendment 25 to GE Topical Report NEDE-24011-*P-A* (GESTAR II) on Cycle-Specific Safety Limit MCPR (TAC No. M97491), January 28, 1998.

8. GE Letter GAW-98-009 MFN-014-98 from Glen A. Watford to USNRC, Responses to NRC Request for Additional Information associated with SLMCPR Methodology and Uncertainty Topical Reports NEDC-32601P and NEDC-32694P, April 17, 1998.

8. GE Letter GAW-98-012 MFN-017-98 from Glen A. Watford to USNRC, Additional Information Associated with SLMCPR Methodology and Uncertainty Topical Report NEDC-32601P, July 29, 1998.

10. "Recommended Practice - Setpoint Methodologies," Part II, ISA-RP 67.04, Instrument Society of America, September 1994.

11. NEDC-32505, Revision 1, "R-Factor Calculation Method for GEII, GEI2, and GEI3 Fuel," June 1997.

ENCLOSURE 2

TECHNICAL EVALUATION REPORT

Report Titles:	1)	Power Distribution Uncertainties for Safety Limit MCPR Evaluations
	2)	Methodology and Uncertainties for Safety Limit MCPR
	3)	Amendment 25 to GE Licensing Topical Report NEDE-24011-P-A (GESTAR) on Cycle-Specific Safety Limit MCPR
Report Numbers:	1)	NEDC-32694P
	2)	NEDC-32601P
	3)	NEDE-24011-P-A
Report Dates:	1)	January 1997
	2)	December 1996
	3)	December 1996

Originating Organization: General Electric Company

1.0 INTRODUCTION

In Reference-1, the General Electric Company (GE) has submitted the proposed GESTAR modifications for including the cycle-specific Safety Limit MCPR (SLMCPR), replacing the generic bounding SLMCPR methodology included in GESTAR, for NRC review and approval. These modifications provide the licensing methods to be used in determining the cycle-specific SLMCPR for each plant reload. In support of these modifications, GE has submitted the two additional licensing topical reports: (1) NEDC-32601P (Reference-2), "Methodology and Uncertainties for Safety Limit MCPR," and (2) NEDC-32694P (Reference-3), "Power Distribution Uncertainties for Safety Limit MCPR Evaluations." The NEDC-32601P Topical Report describes the procedures used to account for the reload-specific core design and operation in determining the cycle-specific

SLMCPR. In this topical report, the values of the plant monitoring uncertainties and local R-Factor uncertainty used in the SLMCPR determination are also reviewed and updated to reflect current recommended practices.

The NEDC-32694P Topical Report provides the power distribution uncertainty for the new GE 3D-MONICORE core surveillance system. The 3D-MONICORE power distribution uncertainties are determined based on an uncertainty propagation analysis and on comparisons with benchmark measurements. The resulting 3D-MONICORE uncertainties are used in the determination of the SLMCPR for the plants employing the 3D-MONJCORE system.

The review of the GE core monitoring and SLMCPR analysis was included in the NRC vendor inspections (Nos. 99900003/95-01 and 99900003/96-01) at the General Electric Nuclear Energy Facility in Wilmington, NC during the weeks of August 14 through September 1, 1995 and May 6 through May 10, 1996. Several important concerns were identified during these reviews including: (1) the level of conservatism in the operating state assumed in the cycle-specific determination of the SLMCPR and (2) the effect of the 3D-MONICORE uncertainties on the SLMCPR uncertainty analysis. These concerns are addressed in the safety limit methodology and uncertainty analysis Topical Report NEDC-32694P and the power distribution uncertainty Topical Report NEDC32694P, respectively.

The purpose of this review was to evaluate these methodology modifications and updates to insure that the changes in the monitoring uncertainties are acceptable and that adequate margin is included in the determination of the SLMCPR. The methodology changes are summarized in Section 2, and the evaluation of the important technical issues raised during this review is presented in Section 3. The Technical Position is given in Section 4.

2.0 SUMMARY OF THE REVISED SLMCPR METHODOLOGY

2.1 Power Distribution Uncertainties for Safety Limit MCPR Evaluations (NEDC-32694P)

The power distribution uncertainty Topical Report NEDC-32694P provides (1) a description of the 3D-MONICORE core surveillance system and (2) the determination ofthe associated bundle power uncertainty for use in SLMCPR calculations. The 3D-MONICORE system uses three-dimensional

coarse-mesh diffusion theory methods, together with models for interfacing with the incore TIP and LPRM instrumentation, to determine the detailed core statepoint. The physics methods used in 3D-MONICORE are identical to those used in BWR fuel design calculations and core management evaluations. 3D-MONICORE solves a modified diffusion theory equation in order to allow the local normalization of the power distribution to the TIP and LPRM incore measurements. However, prior to this normalization, the TIP/LPRM measurements are compared to the instrument responses predicted by 3D-MONICORE. If these comparisons indicate that certain measurements are suspect, this data is rejected and the normalization is performed with the remaining reliable TIP/LPRM measurements.

The uncertainty in the 3D-MONICORE prediction of bundle power was determined by comparisons of measured and calculated TIP integrals and gamma scanned bundle powers. These comparisons included a wide range of fuel enrichments, poison loadings and operating conditions. The increased uncertainty between TIP measurements was determined by comparing LPRM-updated TIPs and TIP measurements taken immediately following the LPRM update. The uncertainty analysis also accounts for TIP and LPRM failures (i.e., measurement rejection). The NEDC-32694P uncertainty analysis indicates that the 3D-MONICORE power distribution uncertainty is less than the value presently used in the GETAB SLMCPR determination.

2.2 Methodology and Uncertainties for Safety Limit MCPR (NEDC-32601P)

The NEDC-32601P Topical Report documents the latest updates to the GETAB (Reference-4) (1) plant surveillance measurement uncertainties, (2) local R-Factor uncertainties and (3) SLMCPR methodology. The plant surveillance component uncertainties include the reactor pressure, feedwater temperature and flow, core inlet temperature and flow, and channel flow area and friction factors. The plant surveillance uncertainty revisions are based on current BWR practice, and are generally evaluated using the error methodology of Reference-5. The uncertainty analysis accounts for the overall instrument channel accuracy, drift, calibration, process uncertainty, and plant environmental effects. In most cases, a simple sum-of-the-squares combination of the contributing uncertainties is employed, however, the uncertainty in the inlet subcooling (i.e., core inlet temperature) is determined using the process computer heat balance to propagate the uncertainties.

In most cases, the reevaluation of the plant surveillance uncertainties concluded that the presently accepted GETAB uncertainty values are conservative. A detailed analysis is provided to support the revised values in the cases where the reevaluation results in a reduction in the component uncertainty.

In the GEXL correlation, the R-Factor provides the critical power dependence on the local pin power distribution (References 4 and 6). The R-Factor uncertainty analysis includes an allowance for power peaking modeling uncertainty, manufacturing uncertainty and channel bow uncertainty. The TGBLA (Reference-7) power peaking modeling uncertainty is determined by comparisons of TGBLA with MCNP (Reference-8) and quarter-core benchmark calculations for a range of BWR fuel bundle and core designs. The power peaking uncertainty determined by this analysis was confirmed with gamma scan measurements taken following Cycle-8 of the Duane Arnold Plant (Reference-9).

The uncertainty in the power peaking resulting from channel bow is determined using the procedures of Reference- 10, and the uncertainty introduced by the manufacturing process is baseci on estimated fuel enrichment and density measurements. The R-Factor uncertainty is determined by propagating the resulting local power peaking uncertainty using the R-Factor dependence on peaking factor.

The revised methodology includes updates to the calculation process used to determine the SLMCPR. The operating core statepoint is determined using the PANACEA (Reference-7) 3D-simulator program. The statepoint information used in the SLMCPR calculation includes the channel flows, bundle powers, local void fraction and the TIP detector responses. In addition, the bundle and exposure dependent R-Factors are obtained from the PANACEA statepoint data and used to determine the critical power. The SLMCPR is determined by randomizing the statepoint surveillance input and correlation data to determine the MCPR margin required to insure that 99.9 % of the rods avoid boiling transition.

The SLMCPR is sensitive to the assumed statepoint radial power distribution. In the cycle-specific methodology, the power distribution is selected to provide a reasonable bound on the number of rods expected to experience boiling transition. This selection is made subject to the condition that the core is critical and within thermal limits. For current BWR reload designs the limiting radial power distribution includes a centrally located high powered region which is either circular or annular in

shape. Control rod patterns which provide these limiting power distributions are described and recommended. In order to quantify the severity of power distributions with respect to the number of rods in boiling transition a core weighting parameter is defined. The frequency distribution of this parameter is used to compare and select the limiting power distribution.

The determination of the SLMCPR using the revised methodology and input uncertainties is compared to the presently accepted GETAB methodology for several plants. For the cases evaluated, the effect of the changes in methodology and uncertainties is small ~.01 ΔSLMCPR.

2.3 GESTAR II Amendment 25 on Cycle-Specific Safety Limit MCPR (NEDE-24011-P-A)

Amendment 25 to GESTAR II provides the methodology and uncertainties required for the implementation of the cycle-specific Safety Limit MCPR. A set of general procedures are given describing the analysis to be used in determining the cycle-specific SLMCPR. These procedures require that the analysis be performed for the specific fuel bundle design and core loading used in the cycle reload design. The core radial power distribution must represent a reasonable bound on the number of fuel bundles at or near thermal limits, and the fuel assembly local power distribution must be based on the actual bundle design. The cycle-specific analysis is performed at multiple exposure points throughout the cycle, and either the most limiting or an exposure-dependent SLMCPR is used in determining the Operating Limit MCPR (OLMCPR). The cycle-specific procedures require that the SLMCPR be recalculated or' reconfirmed for each plant operating cycle.

In the reload process, the final core loading plan is evaluated relative to the reference design criteria including the OLMCPR. If the cycle-specific determination results in an increased SLMCPR, the final core loading plan may fail to satisfy the specified acceptance criteria. In this case, calculations of the sensitivity of the OLMCPR to changes in the SLMCPR are used to determine the acceptability of the calculated cycle-specific SLMCPR.

While Amendment 25 provides the overall procedures for determining the cycle-specific SLMCPR, the detailed SLMCPR methodology and input uncertainties are described in NEDC-32601P and the methodology for constructing the bounding statepoint power distribution is described in NEDC-32694P.

3.0 SUMMARY OF THE TECHNICAL EVALUATION

The GE Topical Reports NEDC-32694P, NEDC-32601P and Amendment 25 to NEDE-24011-P-l (GESTAR II) provide the basis for the cycle-specific determination of the SLMCPR, input plant system uncertainties and the power distribution uncertainties associated with the application of 3D-MONICORE. The review of the GE methodology focused on: (1) the assumptions made in the cycle-specific SLMCPR methodology and the changes relative to the presently approved generic SLMCPR approach and (2) the basis for the changes in the SLMCPR uncertainty values. As a result of the review of the methodology, several important technical issues were identified which required additional information and clarification from GE. This information was requested in References-11 and 12 and was provided in the GE responses included in References 13-16. This evaluation is based on the material presented in the topical reports (References 1-3) and in References 13-19. The evaluation of the major issues raised during this review are summarized in the following.

3.1 Power Distribution Uncertainties for Safety Limit MCPR Evaluations (NEDC-32694P)

The 3D-MONICORE system is used to perform the steady-state on-line core performance evaluation. The 3D-MONLCORE models are based on accepted BWR calculational methods. The neutronics model is essentially the same as that described in Reference-7 and the thermal-hydraulics model is the same as presently used in the P-1 Process Computer Analysis (Reference- 13, Response 11.4).

The 3D-MONICORE power distribution uncertainties are required for determining the SLMCPR, LHGR and MAPLHGR limits. The (axially integrated) bundle power uncertainty is required for the SLMCPR and the nodal power uncertainty is required for detenmining MAPLHGR and LHGR. The radial bundle power uncertainty is considered to be a statistical combination of: (1) the uncertainty in the four-bundle power associated with the TIP location and (2) the uncertainty in the allocation of the four-bundle power to the individual bundles. The four-bundle power uncertainty is determined by a comparison of the predicted and measured TIP responses, and the uncertainty in the power allocation is determined by comparisons of calculated and measured (gamma-scanned) bundle powers.

While the calculated bundle powers were determined with the "core tracking" system, rather than with 3D-MONICORE, GE has indicated (in Reference- 13, Responses 1.2 and 1.6) that the difference in these codes has no effect on the uncertainty estimates. The TIP comparisons include cores with both part length fuel rods and axially zoned gadolinium, and all current fuel product lines except for GE13. However, in view of the similarity of the GE13, GE11, and GE12 fuel designs, this is considered acceptable (Reference- 13, Response-II.4). In addition, GE has indicated that the core follow calculations employed the same methods to process and accumulate the void-history and fuel exposure as used in the on-line core surveillance (Reference-1 3, Response-II.8). However, it is concluded that since changes in the fuel and core design can have a significant effect on the calculation accuracy, the 3D-MONICORE bundle power calculational uncertainty should be verified when applied to fuel and core designs not included in the benchmark comparisons of Tables-3.1 and 3.2 of NEDC-32694P.

The review of the calculation-to-measurement (C/M) comparisons indicated an increased uncertainty at end-of-cycle. However, the cycle-average four-bundle power uncertainty is considered acceptable since the uncertainty estimate does not take credit for the uncertainty increase due to TIP measurement uncertainty. The nodal power uncertainty is determined by a statistical combination of the 3D-MONICORE bundle power uncertainty and an accepted TIP axial power uncertainty. The TIP uncertainty is measured once per cycle to ensure that it satisfies the specified acceptance criteria.

The 3D-MONICORE system allows rejection of the TIP measurement data based on a specified acceptance criteria. During the review it was noted that the 3D-MONICORE acceptance criteria will, under certain conditions, reject good TIP measurement data. However, in Responses-I.7 and I.10 (Reference- 13), GE has indicated that the rejection of TIP data is very rare. In addition, in most cases TIP rejection is due to poor agreement between measured and calculated data and, when the acceptance criteria results in rejection of measurements which are in good agreement with the calculations, the effect on the core power distribution uncertainty is negligible.

The uncertainty methodology determines the effect of TIP and LPRM rejection and the LPRMupdate of the power distribution using comparisons of calculations and measurements. In these comparisons the recommended value for the rejection criteria parameter is used. It is noted that after ten years of operation, no correlation has been observed between the rejected TIP locations and the

core locations that are difficult to calculate, such as the peripheral core locations, part-length fuel bundles and partially controlled fuel bundles. It is concluded that the TIP rejections are generally a result of erroneous measurement data rather than miscalculation. It is also noted that the TIP rejection only affects the 3D-MONICORE system and the other BWR surveillance systems use the measured TIP/LPRM data.

The process computer monitors kw/ft and LHGR as well as the SLMCPR. The uncertainty analysis for the 3D-MONICORE LHGR evaluation is provided in Response-II.5 (Reference- 14) and accounts for the effect of both the TIP and LPRM update uncertainties on the nodal power calculation.

Based on the review of the NEDC-32694P topical report and supporting documentation provided in References 13 and 14, it is concluded that the 3D-MONICORE power distribution uncertainties are acceptable for determining the SLMCPR, MAPLLIGR and LHGR core limits subject to the condition identified above (in the third paragraph of this section).

3.2 Methodology and Uncertainties for Safety Limit MCPR (NEDC-32601P)

3.2.1 Process Computer Uncertainties

The reevaluation of the process computer uncertainties provided in the NEDC-32601 P topical report were reviewed in detail. The topical report provides a description of both the instrumentation and modeling uncertainties that are required for the SLMCPR analysis. The evaluation of the core inlet subcooling uncertainty employs the heat balance used by the process computer to relate the inlet subcooling to the available instrumentation signals. Using this relation, the inlet subcooling variance is determined by the individual component variances (e.g., feedwater flow and temperature, core flow, steam carry under fraction). While the coefficients that weight the individual uncertainty components depend on the reactor statepoint, the analysis neglects this dependence and assumes constant weighting coefficients. In Response-I. 1 (Reference-15), GE has shown using a Monte Carlo procedure that these constant weighting coefficients are conservative.

The calculation of the bundle critical power is sensitive to the channel flow area and friction factors. The two-phase friction factor is based on measurements made at the full scale ATLAS test facility covering a range of power and flow The uncertainty in the two-phase friction factor is based on the

comparisons with test data. The uncertainty in the single-phase friction factor is determined by comparison of the calculations to total pressure drop measurements made at the ATLAS facility. In Response-II.6 (Reference-15), it is noted that, since the total pressure drop measurement includes both the single-phase and two-phase losses, the inferred single-phase loss coefficient is conservative.

The channel flow area is subjected to random variations due to non-uniform channel bulge and crud/corrosion buildup which result in channel-to-channel variations in flow. The SLMCPR uncertainty analysis accounts for the effect of these variations by increasing the uncertainty in the channel-to-channel friction factor multiplier (Response-I.2, Reference- 15).

3.2.2 R-Factor Uncertainties

The fuel rod power calculational uncertainty determines the GEXL R-Factor uncertainty and is separated into three components; modeling, manufacturing and bowing. The modeling uncertainty is determined by comparison of the TGBLA calculation to MCNP benchmark lattice calculations. The Table-3.1 TGBLAIMCNP rod power comparisons include all GE fuel designs which are currently loaded in operating BWRs (Response-II.1, Reference-iS). A range of gadolinium rods is included in the comparisons in order to simulate the effects of depleted fuel rods (Response-II.2, Reference- *15)*. The fuel rod power peaking uncertainty is determined by weighting the variance for each fuel design by the number of rods in the lattice (Response-II.3, Reference-iS). However it is concluded that since changes in the fuel lattice design can have a significant effect on the calculation accuracy, the TGBLA fuel rod power calculational uncertainty should be verified when applied to fuel designs not included in the benchmark comparisons of Table-3.1 of NEDC-32601P.

In addition to the TGBLA!MCNP comparisons, GE has evaluated the effect of void fraction uncertainty on the fuel rod power calculation (Response-II.10, Reference-15). Estimates of the lattice-average void fraction uncertainty were determined by comparison with measurement. The local void fraction uncertainty was determined by comparison with detailed subchannel calculations. The effect of the lattice-average and local void fraction uncertainties on the fuel rod power calculation was determined by sensitivity calculations and found to be negligible.

The fuel rod manufacturing uncertainty includes the effects of fuel enrichment, density and rod position uncertainty. The uncertainty in fuel enrichment and density was determined from measurements on a large number of fuel rods performed as part of manufacturing studies. The fuel rod position uncertainty was determined from a series of rod spacing measurements performed on a high burnup fuel bundle. In Responses 11.4, 11.5, and 11.9 of References 14 and *15,* GE has shown that the effects of these uncertainties are conservatively included in the R-Factor analysis. In Response-II.10 (Reference- 14), the effect of local fuel bundle exposure uncertainty on rod power is shown to be negligible. It is important to note that the power peaking uncertainty is determined using a components of uncertainty approach and then independently confirmed by a comparison with gamma scan measurements.

In the approved GETAB methodology of Reference-4, the power peaking calculation errors in neighboring fuel rods are assumed to be correlated so that each of the fuel rods has exactly the same calculational error. In the proposed methodology, the modeling errors in neighboring fuel rods are assumed to be uncorrelated. As a result, the uncertainty in the R-Factor is reduced significantly in the proposed methodology. In Response-II.13 of Reference 14 and in References 17-19, GE has evaluated this effect for the 8x8, 9x9 and 10x10 lattices and has indicated that the R-Factor uncertainty will be increased (relative to the presently approved value of Reference-4) to account for the correlation of rod power uncertainties. However, in References-18 and 19 (Table-1), it is noted that the effect of the rod-to-rod correlation has a significant dependence on the fuel lattice (e.g., 9x9 versus 10x10). Therefore, in order to insure the adequacy of the R-Factor uncertainty, the effect of the correlation of rod power calculation uncertainties should be reevaluated when the NEDC-32601P methodology is applied to a new fuel lattice.

3.2.3 SLMCPR Evaluation Methodology

The SLMCPR is sensitive to the "flatness" of the bundle power distribution of the initial reactor statepoint. GE has defined a MCPR Importance Parameter (MIP) to allow a quantitative assessment of the flatness of the power distribution and identify limiting statepoints for SLMCPR analysis. In Response-III.2 (Reference- 15*),* the expression for determining MIP is derived and shown to provide a quantification of the effect of the bundle power distribution on the SLMCPR. In Reference- 15,

GE provides the specific MIP criterion (Response-III.5) and the thermal limits and reactivity constraints (Response-III.6) for selecting the bundle power distribution to be used in the SLMCPR analysis.

The determination of the selected MIP criterion is based on an extensive evaluation of operating reactor statepoints. In view of the importance of this MIP criterion and its potential sensitivity to changes in fuel bundle designs, core loadings and operating strategies, there is a need to insure that the specific value recommended in NEDC-32601P is applicable to future designs and operating strategies. In response to this concern, GE has indicated that the MIP criterion will be reviewed periodically as part of the procedural review process (Response III.6, Reference-15).

In the presently approved GETAB methodology (Reference-4), the bundle power calculation error is assigned to the four bundles surrounding the TIP in a correlated manner so that each of the four bundles is perturbed simultaneously by the same amount. In the proposed methodology, the calculational error is assumed to be uncorrelated and the individual bundle powers are varied independently in the Monte Carlo uncertainty propagation. The increased variability in the proposed methodology results in a (non conservative) reduction in the SLMCPR. In Response-III.1 of Reference-14, GE has revised the NEDC-32601P methodology to allow for the correlation of the bundle power calculation modeling errors.

Based on the review of the NEDC-32601P topical report and supporting documentation provided in References 14 and 15, we find the SLMCPR methodology and associated uncertainties to be acceptable subject to the conditions identified in Sections-3.2.2 and 3.2.3.

3.3 GESTAR II Amendment 25 on Cycle-Specific Safety Limit MCPR (NEDE-24011-P-A)

Amendment 25 to GESTAR II provides the modifications required for performing the cycle-specific SLMCPR analysis. In the cycle-specific analysis, a search is performed to determine the initial reactor statepoint for use in the Monte Carlo statistical analysis. The purpose of this search is to determine a reactor statepoint that satisfies both (1) the operations criteria required for operating statepoints and (2) the MIP flatness criterion to insure that the statepoint provides a bounding SLMCPR. In the information provided in support of Amendment 25 (Reference-1, Corrective

Action-4, Item-3), it is noted that this search may be terminated before all criteria are satisfied. However, in Responses 2 and 3 (Reference- 16), GE has indicated that if the MIP criterion is not initially satisfied the search will be expanded, by relaxing the operations criteria, to insure that the MIP criterion is satisfied.

In the presently approved GETAB methodology, the limiting power shape is assumed to include a centrally located annular ring of high-powered fuel bundles. While the proposed cycle-specific methodology does not require the power distribution to include this high-powered annular zone, it is indicated in Response-4 (Reference- 16) that the control rods are selected so that this power shape is not precluded from the search for the bounding statepoint.

Based on the review of Amendment 25 and the supporting information provided in Reference-16, we find the proposed methodology to be acceptable for performing cycle-specific SLMCPR analyses.

4.0 TECHNICAL POSITION

The Topical Reports NEDC-32694P, NEDC-32601P and Amendment 25 to NEDE-24011-P-A (GESTAR II) and supporting documentation provided in References 13-16 have been reviewed in detail. Based on this review, it is concluded that the proposed cycle-specific determination of the SLMCPR, the input plant system uncertainties, and the power distribution uncertainties associated with the application of 3D-MONICORE are acceptable subject to the conditions stated in Section 3 of this evaluation and summarized in the following.

1) Since changes in the fuel and core design can have a significant effect on the calculation accuracy, the 3D-MONICORE bundle power calculational uncertainty should be verified when applied to fuel and core designs not included in the benchmark comparisons of Tables-3.1 and 3.2 of NEDC-32694P (Section 3.1).

2) Since changes in fuel design can have a significant effect on the calculation accuracy, the TGBLA fuel rod power calculational uncertainty should be verified when applied to fuel designs not included in the benchmark comparisons of Table-3.1 of NEDC-32601P (Section-3.2.2).

3) In order to insure the adequacy of the R-Factor uncertainty, the effect of the correlation of rod power calculation uncertainties should be reevaluated when the NEDC-32601P methodology is applied to a new fuel lattice (Section-3.2.2).

4) In view of the importance of the MIP criterion and its potential sensitivity to changes in fuel bundle designs, core loadings and operating strategies, the MIP criterion should be reviewed periodically as part of the procedural review process to insure that the specific value recommended in NEDC-32601P is applicable to future designs and operating strategies (Section-3.2.3).

5.0 REFERENCES

1. "Proposed Amendment 25 to GE Licensing Topical Report NEDE-24011-P-A (GES TAR II) on Cycle-Specific Safety Limit MCPR," RJR-96-133, Letter R. J. Reda (GE) to U.S. NRC, dated December 13, 1996.

2. "GE Licensing Topical Report, Methodology and Uncertainties for Safety Limit MCPR Evaluations," RJR-96-139, Letter, R. J. Reda (GE) to U.S. NRC, dated December 13, 1996.

3. "GE Licensing Topical Report, Power Distribution Uncertainties for Safety Limit MCPR Evaluations," RJR-97-074, Letter, R. J. Reda (GE) to U.S. NRC, dated June 10, 1997.

4. "General Electric BWR Thermal Analysis Basis (GETAB) Data, Correlation and Design Application," NEDO-10958-A, January 1977.

5. "Recommended Practice - Setpoint Methodologies," Part II, ISA-PP 67.04, Instrument Society of America, September 1994.

6. "R-Factor Calculation Method for GE11, GE12, and GE13 Fuel," NEDC-32505P, November 1995.

7. "Steady State Nuclear Methods," NEDE-30130-P-A, April 1985.

8. "MCNP - A General Monte Carlo N-Particle Transport Code, Version 4a," LA-12625, J. F. Breismeister, Ed., Los Alamos National Laboratory (1993).

9. "Gamma Scan Measurements of the Lead Test Assembly at the Duane Arnold Energy Center Following Cycle-8," NEDC-3 1569-P. April 1988.

10. "Fuel Channel Bow Assessment," GENE Report MFN086-89, Letter, J. S. Charnley (GE) to R. C. Jones (NRC), dated November 15, 1989.

11. "Request for Additional Information for GE Topical Reports NEDC-32601 P and NEDC-32694P," Letter, J. H. Wilson (NRC) to R. J. Reda (GE), dated August 20, 1998.

12. "Request for Additional Information for Amendment 25 to GE Topical Report NEDE-24011-P-A (GESTAR II) on Cycle-Specific Safety Limit MCPR (TAC No. M97491)," Letter, J. H. Wilson (NRC) to R. J. Reda (GE), dated October 21, 1998.

13. "Responses to Request for Additional Information for GE Topical Report NEDC-32694P," "Power Distribution Uncertainties for Safety Limit MCPR Evaluations," GAW-98-003, Letter, G. A. Watford (GE) to U.S. NRC, dated January 9, 1998.

14. "Responses to NRC Request for Additional Information Associated with SLMCPR Methodology and Uncertainty Topical Reports NEDC-32601P and NEDC-32694P," GAW 98-009, Letter, G. A. Watford (GE) to U.S. NRC, dated April 17, 1998.

15. "Responses to Request for Additional Information for GE Topical Report NEDC-32601P, Methodology and Uncertainties for Safety Limit MCPR Evaluations," GAW-98-002, Letter, G. A. Wafford (GE) to U.S. NRC, dated January 8, 1998.

16. "Responses to NRC Request for Additional Information for Amendment 25 to GE Topical Report NEDE-2401 1-P-A (GESTAR II) on Cycle-Specific Safety Limit MCPR (TAC No. M97491)," GAW-98-005, Letter, G. A. Watford (GE) to U.S. NRC, dated January 28, 1998.

17. "Additional Information Associated with SLMCPR Methodology and Uncertainty Topical Report NEDC-32601P," GAW-98-012, Letter, G. A. Watford (GE) to U.S. NRC, dated June 12, 1998.

18. "Additional Information Associated with SLMCPR Methodology and Uncertainty Topical Report NEDC-32601P," GAW-98-012, Letter, G. A. Watford (GE) to U.S. NRC, dated July 29, 1998.

19. "Additional Information Associated with SLMCPR Methodology and Uncertainty Topical Report NEDC-32601P," GAW-98-017, Letter, G. A. Watford (GE) to U.S. NRC, dated September 9, 1998.

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1. Introduction

This report provides an update to the methodology and inputs to be used in the evaluation of the Safety Limit Minimum Critical Power Ratio (SLMCPR) for BWRs. This methodology was previously described in Appendix IV and Appendix VII of the GETAB thermal analysis basis (Reference 4). All uncertainty values used in the determination of the SLMCPR are verified in accordance with current practices (Reference 5). In most cases the values currently approved for GETAB analyses are confirmed to be applicable and generally are found to be conservative. For a few cases, minor changes in values are justified based on updating to reflect current recommended practices. The uncertainties are summarized in Section 2. Section 2.10 introduces a revised method for the existing power distribution and TIP uncertainties.

Section 3 of this report evaluates uncertainties to be used for the bundle R–factor, which determines the margin to thermal limits for a given fuel bundle. The R–factor uncertainty depends on the uncertainty in pin power peaking factor. Justification for the pin power peaking factor consists of comparisons of the lattice design model TGBLA to Monte Carlo benchmark calculations and to rod by rod gamma scan data obtained from a special feature Lead Test Assembly irradiated at the Duane Arnold plant from 1984 to 1987.

Section 4 contains a detailed description of a revised methodology introduced in Section 2.10 for evaluating the SLMCPR and compares it to the currently approved methodology documented in Appendix IV of Reference 4. The revised evaluations represent a more realistic simulation of the true probability of boiling transition in a BWR. This section also presents a comparison of SLMCPR evaluations performed with the current and revised methodologies.

GE will also be submitting another licensing topical report (NEDC–32694P) that will address the power distribution uncertainties consistent with 3D MONICORE shape–adaption.

2. Uncertainty Summary

2.1 Introduction

This section contains a summary of the uncertainty values used in the Safety Limit MCPR analysis. The analysis supporting the R–factor uncertainties is covered in Section 3. Table 2.1 contains a summary of the uncertainties in the SLMCPR analysis, along with the currently approved values as specified in GETAB (Reference 4). Sections 2.2 through 2.6 contain evaluations of uncertainties associated with reactor instrumentation. These uncertainties are evaluated in accordance with standard instrument channel methodologies as prescribed in Reference 5. The error methodology in Reference 5 considers the following elements:

- The entire instrument channel from primary element through computer input.
- Accuracy, calibration, and drift over a realistic 30–month surveillance interval.
- Influences resulting from actual plant environments and process effects.

All error terms are evaluated and combined in accordance with accepted methodologies and channel errors are determined at the one sigma (1σ) level to allow for direct comparison with Reference 4, Appendix IV and VII.

Sections 2.7 through 2.9 contain evaluations of thermal hydraulic parameter uncertainties, and are either based on known dimensional tolerances or comparisons of calculated versus measured pressure drop data. Section 2.10 contains a review of the P1 process computer TIP and power distribution uncertainties.

Table 2.1
Summary of SLMCPR Uncertainties

Uncertainty Parameter	GETAB Uncertainty ±σ (%)	Revised Uncertainty and Procedures ±σ (%)	Reference
Feedwater Flow System Overall Flow Uncertainty	1.76	[[ ]]	Section 2.2
Feedwater Temperature Measurement	0.76	[[ ]]	Section 2.3
Reactor Pressure Measurement	0.50	[[ ]]	Section 2.4
Core Inlet Temperature	0.20	0.2	Section 2.5
Total Core Flow Measurement	2.5 (6.0 for Single Loop Operation)	2.5 (6.0 for Single Loop Operation)	Section 2.6
TIP Reading and Bundle Power	8.6 applied to quarter segment TIP reading (current procedure) 1.2 random uncertainty applied nodally	Current Uncertainties Total Bundle Integrated Power Uncertainty = [[ ]] (Applied to bundle integral) Total TIP Integral Instrument Uncertainty = [[ ]] (applied to quarter segment) 3D MONICORE Uncertainties	Reference 1 Section 2.10 Section 2.10 Reference 10
TIP Reading Random Uncertainty	1.2 (2.85 for Single Loop Operation)	1.2 (2.85 for Single Loop Operation)	Reference 1
Channel Flow Area Variation	3.0	[[ ]]	Section 2.7
Friction Factor Multiplier Uncertainty	10.0	[[ ]]	Section 2.8
Channel Friction Factor Multiplier	5.0	5.0	Section 2.9
R–factor Uncertainty	1.5	[[ ]]	Section 3 & Appendix C
Critical Power Uncertainty	Different for Each Fuel Type	Different for Each Fuel Type	Reference 11

2.2 Feedwater System Flow Uncertainty

The instrument methodology approach focuses mainly on instrumentation–related errors (all random) via a detailed treatment of uncertainties through analog signal–conditioning modules (pressure transmitters through computer input). Primary element (venturi) and process effect errors, while treated separately, are each generalized into one overall term and are treated as random, which is considered reasonable based on 1) ASME Fluid Meters typically considers flow–related uncertainties (called "tolerances" and covering coefficients, diameters, etc.) to be random; and 2) process effects (e.g., density variations) can occur in either direction (pressure, temperature fluctuations).

Reference 4 focuses mainly on flow element errors, identifies several individual terms (calibration, feedwater line differences, pressure fluctuation and effect on density), assigns very conservative values, and initially identifies some as biased–uncertainties prior to randomizing them with instrumentation uncertainties (which are generalized and are considered random).

Incorporation of the Reference 4 flow element error values into the current instrument methodology approach yields a 1σ error as shown below:

Error Term	1σ error (% of rated)

2.3 Feedwater Temperature Measurement

Although errors are allocated somewhat differently in the instrument methodology approach versus the Reference 4 approach, the instrument methodology approach has been modified to essentially agree with the Reference 4 approach, including the incorporation of a process temperature fluctuation error and the consideration of four temperature inputs (two per feedwater line), yielding the following:

Error Term	1σ error (% of rated)

2.4 Reactor Pressure

The instrument methodology approach is essentially the same as that used in Reference 4, except a realistic 30–month calibration interval has been considered:

Error Term	1σ error (% of rated)

2.5 Core Inlet Temperature

The approach and inputs for calculating the total uncertainty in core inlet temperature is defined in Appendix VII of Reference 4 in response to Question 3–8. The basic approach is to calculate the total uncertainty ($\sigma_{\Delta h_0}$) in core inlet enthalpy subcooling (Δh_o) from the heat balance as performed by the process computer and then translate this into a total uncertainty ($\sigma_{\Delta h_0}$) in core inlet temperature (T_0) using the relationship

$$\sigma_{T_0} = \frac{\sigma_{\Delta h_0} \Delta h_0}{T_0 c_P} \qquad (2\text{–}1)$$

For purposes of these discussions the specific heat at constant pressure (c_p) for subcooled water has a value of 1.25 Btu/(lb_m–°F), the nominal core inlet temperature (T_0) is 535°F and the core inlet subcooling (Δh_0) is 21 Btu/lb_m.

Equation 8–1 on page VII–32 of Reference 4 indicates how core inlet enthalpy subcooling is calculated by the process computer. For convenience that equation is presented below as Equation 2–2. All symbols are defined on page VII–32 of Reference 4.

$$\Delta h_0 = \frac{W_{FW}}{W_T}(H_F - H_{FW}) - F_{CU}H_{FG} + \frac{W_{CR}}{W_T}(H_F - H_{CR}) + \frac{Q_{CU}}{W_T} - \frac{Q_P}{W_T}C \qquad (2\text{–}2)$$

Having defined core inlet enthalpy subcooling, core inlet temperature is then "back–calculated" from enthalpy and pressure (assumed to be dome pressure) using the property relationships for subcooled water. All of this is done in the way that it is done in the plant process computer.

The total uncertainty in core inlet enthalpy subcooling (Δh_0) is determined from Equation 2–2 by using a weighted Square–Root–Sum–of–Squares (SRSS) as follows

$$\sigma^2_{\Delta h_0} = \sum_i [\Phi_i \sigma_i]^2 \qquad (2\text{–}3)$$

where the weighting coefficients Φ_i are determined in unnormalized form by

$$\Phi_i = \left(\frac{\partial(\Delta h_0)}{\partial \psi_i}\right)\psi_i \qquad (2\text{–}4)$$

The ψ_i correspond to each uncertain parameter used to evaluate Δh_0. It is typical (and conservative) to normalize the weighting coefficients so that the sum of the component variances is unity thus the unnormalized Φ_i in Equation 2–3 are replaced by $\hat{\Phi}_i$ where

$$\hat{\Phi}_i = \frac{[\Phi_i \sigma_i]^2}{\sum_i [\Phi_i \sigma_i]^2} \qquad (2\text{–}5)$$

Table 3.8–2 in Appendix VII of Reference 4 defines the σ_i values. In the text under Equation 8–1 the relative contributions (Φ_i) assumed in the Reference 4 analysis are indicated for each input to the total uncertainty. The principal contributors as reported in Reference 4 are in order of descending relative contribution: core flow (59%), feedwater flow (22%), feedwater temperature (11%), core pressure (4%), and steam carry–under fraction (3%). All other inputs to the equation contribute less than 1% to the total core inlet subcooling uncertainty. When the relative contribution factors and uncertainties reported in Reference 4 are used the total uncertainty in core inlet enthalpy subcooling is calculated to be [[ ]]which corresponds to a core inlet temperature uncertainty of [[ ]].

The influence that the weighting factors have on total uncertainty were evaluated. Possible plant-specific variations in the relative contributions to the total core inlet enthalpy subcooling uncertainty are due to plant-specific variations in the values of the quantities (ψ_i) used to evaluate Δh_0. Different conservative approaches were used to calculate the weighting coefficients in order to conservatively calculate the total uncertainty in core inlet enthalpy subcooling. The different calculated values for the total core inlet enthalpy subcooling uncertainty ranged from [[ ]] which correspond to total uncertainties in core inlet temperatures in the range [[ ]]. Note that all these values are well below the [[ ]] uncertainty in core inlet enthalpy subcooling necessary to yield a [[ ]] uncertainty in core inlet temperature.

It is concluded that the core inlet temperature uncertainty of 0.2% specified on page VII-32 of Reference 4 is adequately conservative to accommodate as much as a factor of [[ ]] increase in either the uncertainties or the relative contribution coefficients stated in that reference. This level of conservatism is judged to be adequate to accommodate all plant specific variations.

2.6 Total Core Flow Measurement

The instrument methodology approach focuses mainly on instrumentation-related errors (all random) via a detailed treatment of uncertainties through analog signal-conditioning modules (pressure transmitters through computer input). Primary element (jet-pump) and process effect errors, while treated separately, are each generalized into one overall term and are treated as random. This approach is consistent with the ASME methodology for treating fluid meters which considers flow-related uncertainties to be random. Process effects (e.g., density variations) are also assumed to be random since they can occur in either direction due to random pressure and temperature fluctuations.

Reference 4 focuses mainly on jet-pump errors, identifies several individual terms (variance in calibration coefficients, jet-pump sampling, and density), assigns very conservative values, and treats them as bias-type errors (additive). Instrumentation errors are generalized and are considered random.

The errors are presented in terms of 100% power and 100% core flow. Incorporation of the conservative Reference 4 jet-pump error values into the current instrument methodology approach yields 1σ channel errors as shown below:

Error Term	1σ error (% of rated)

The overall core flow uncertainty for single loop operation remains at 6.0% which is conservative per Reference 11.

2.7 Channel Flow Area

The uncertainty in the channel flow area can be determined from the manufacturing tolerances on the inner dimensions of the channel and the outer diameter of the fuel and water rods. The GE12 10x10 lattice is used to determine the flow area uncertainty because it has the largest number of fuel rods, yielding the largest sensitivity to variations in fuel rod diameter. The flow area uncertainty is given by,

(2-6)

where A_{flow} is the total channel flow area, N_{rod} is the number of fuel rods, N_{wrod} is the number of water rods and D_{rod} and D_{wrod} are the fuel and water rod diameters. W_c is the inside width of the channel. It is conservatively assumed that the standard deviation, σ, for each quantity is given by the dimensional tolerance divided by two. Substituting the appropriate tolerances and dimensions for the GE12 10x10 design into the formula above yields an area uncertainty of σ_a

2.8 Channel Friction Factor Multiplier Uncertainty

The channel friction factor is used to calculate the two phase friction pressure loss in the BWR channel. The friction factor is determined from full scale tests performed in the ATLAS test loop. These tests cover the full range of bundle power and flow expected during BWR operation. The pressure drop correlation has been compared to the experimental data. The standard deviation between the ATLAS experimental data and the correlation varies with mechanical design, but is less than psi in all cases. In addition to the two phase pressure drop uncertainty there is a single phase component which covers the pressure drop between the side entry orifice and the active channel above the lower tie plate. This pressure drop is also determined from ATLAS tests, but is less certain because the geometry near the side entry orifice is not prototypical for all plants. However, the total calculated pressure drop has been compared to

plant data for a number of operating plants. These data are summarized in Table 2.2. Assuming a psi single phase uncertainty and a psi two phase uncertainty results in an RMS

.

Table 2.2
Comparison of Measured and Calculated Core Pressure Drops

Plant	BWR Type	Measured Pressure Drop (psi)	ΔP (calc–meas, psi)
Average ± σ			

.

2.9 Channel to Channel Friction Factor Multiplier Uncertainty

. It is therefore conservative to assume a 5% uncertainty in the SLMCPR uncertainty analysis.

2.10 TIP and Power Distribution Uncertainty

The bases for the currently approved TIP and power distribution uncertainties are documented in Reference 1. In Reference 1, the TIP uncertainty represented the combined effects of TIP instrument uncertainty and the bundle power model uncertainty associated with the P1 process computer model. The revised methodology described in Section 4 calls for the TIP instrument uncertainty to be separated from

the process computer bundle power model uncertainty. The purpose of this Section is to construct the TIP instrument and bundle power model uncertainty using the approved components documented in Reference 1, which are based on the P1 process computer model.

As stated above, the current method for evaluating the SLMCPR combines the TIP instrument and process computer power distribution model uncertainty into the TIP uncertainty. Further, this uncertainty is applied independently to four axial segments of each TIP

2.11 Nomenclature

Symbol	Definition	Units
Δh_0	Core inlet subcooling	Btu/lb_m
σ_a	Channel flow area uncertainty	%
σ_{rod}	Fuel rod diameter uncertainty	in
σ_{wrod}	Water rod diameter uncertainty	in
σ_w	Channel inner width dimension uncertainty	in
N_{rod}	Number of fuel rods in a bundle	—
N_{wrod}	Number of water rod in a bundle	—
W_c	Channel inner width dimension	in
D_{rod}	Fuel rod diameter	in
D_{wrod}	Water rod diameter	in
A_{flow}	Channel flow area	in^2
T_0	Core inlet temperature	°F
C	Unit conversion factor = 3.413 MBtu/(hr–MW)	as stated
c_p	Specific heat at constant pressure	Btu/lb_m–°F
F_{CU}	Steam carry under fraction	—
H_{CR}	Specific enthalpy of control rod drive flow	Btu/lb_m
H_{FG}	Difference between saturated steam and liquid specific enthalpies	Btu/lb_m
H_F	Saturated liquid specific enthalpy	Btu/lb_m
H_{FW}	Specific enthalpy of feedwater flow	Btu/lb_m
Q_{CU}	Heat loss in cleanup system	MBtu/hr
Q_P	Power input due to recirculation pumps	MW
W_{CR}	Control rod drive flow rate	Mlb_m/hr
W_{FW}	Feedwater flow rate	Mlb_m/hr
W_T	Total core flow rate	Mlb_m/hr
Φ_i	Unnormalized weighting coefficient defined by Eq. (2.5–4)	Btu/lb_m
$\hat{\Phi}_i$	Normalized weighting coefficient defined by Eq. (2.5–5)	—
ψ_i	an uncertain input quantity in calculating Δh_0	varies

Symbol	Definition	Units
ΔP	Pressure drop	psi
σ_{TIP}	Total TIP uncertainty	%
σ_{ran}	TIP random error	%
σ_{geom}	TIP geometrical uncertainty	%
σ_{assym}	TIP uncertainty due to P1 extrapolation in asymmetric core conditions	%
σ_{lprm}	TIP uncertainty due to LPRM update process	%
σ_{mdl}	Process computer model contribution to total TIP error	%
σ_B	Bundle integrated power uncertainty due to process computer model	%
$\sigma_{\Delta h_0}$	Total uncertainty of the inlet subcooling	%

3. R–factor Uncertainty

The bundle R–factor is a key parameter in determining the critical power ratio and margin to thermal limits for a given BWR fuel bundle. It represents the influence of fuel rod power peaking on the critical power. The bundle R–factor represented by the symbol R, is an independent input to the GEXL correlation (see Reference 7). The procedure for evaluating the bundle R–factor is documented in Reference 6 for GE11 and later designs and in Reference 7 for GE10 and earlier designs. The formula for the R–factor is given by the following equation

(3–1)

The uncertainty in r_i is evaluated in Section 3.1 and the influence of this uncertainty on the R–factor is evaluated in Section 3.2.

3.1 Uncertainty in the Pin Power Peaking Factor

(3–2)

3.1.1 Model Uncertainty

Fuel pin power peaking factors are computed by the lattice physics code TGBLA (Reference 3). The accuracy of the TGBLA model has been established by comparing its peaking factor distributions with Monte Carlo benchmark results obtained with the MCNP program. The MCNP program was developed at Los Alamos National Laboratory and has been adapted by GE for simulation of BWR fuel bundles. TGBLA power peaking results have been compared for 27 different configurations, which cover eight different 8x8, 9x9 and 10x10 lattice designs. The RMS differences between the MCNP and TGBLA rod powers are summarized in Table 3.1.

Pin power peaking factors are influenced by the presence of neighboring bundles which may have a different enrichment or burnup. This factor has been evaluated by computing quarter core pin power peaking distributions with three group fine mesh diffusion theory. The fine mesh peaking results are then compared to infinite lattice peaking factors for all bundles . The results are summarized in Table 3.2. This table compares the RMS difference between the infinite lattice, which is used to calculate the R–factors, and the fine mesh diffusions theory results. The total uncertainty averaged represents the uncertainty in the local peaking due to the influence of neighboring bundles on the local peaking distribution.

σ_{mdl}

Table 3.1
Summary of TGBLA/MCNP Pin Power Comparisons

Lattice Design	Condition	RMS difference in rod power

Table 3.2
Summary of Fine Mesh Quarter Core Benchmark Comparisons

Quarter Core Model	Condition	RMS Difference in rod peaking distribution between infinite lattice and fine mesh solution

3.1.2 Manufacturing Uncertainty

The principal contributor to the manufacturing uncertainty component is the enrichment uncertainty.

3.1.3 Channel Bow Uncertainty

Procedures for accounting for the effects of channel bow on critical power are documented in Reference 9 and were approved by the NRC staff in 1991. In this procedure the bias in peaking factor due to channel bow is accounted for by adjustments to the lattice peaking factors. The uncertainty in the channel bow, however contributes to the overall peaking uncertainty. In this analysis it is conservatively assumed that the peaking sensitivity to channel bow is given by the corner rod sensitivity. In Reference 9, the uncertainty in the average bow around a particular control blade location is

3.1.4 Confirmation of the Pin Power Peaking Factor Uncertainty With Gamma Scan Data

The total rod peaking uncertainty can be computed by taking the RMS sum of the total model uncertainty,

This uncertainty has been confirmed by comparing the TGBLA local peaking calculations with gamma scan data. The gamma scan measurements have been obtained from a program to determine the relative power shape in Lead Test Assemblies and a few neighboring bundles irradiated in the Duane Arnold Reactor. The measurement program was carried out during the refueling outage at the end of Cycle 8 during the period March 23 to May 18, 1987. This program focused on obtaining the local peaking distribution in the lead test assembly to determine the adequacy of the methods to accurately predict the nuclear and thermal hydraulic characteristics of the bundle during irradiation. Especially important were the perturbations to local power distributions due to the introduction of the large central water rod and the part length rods. The experiment and the detailed comparisons to the TGBLA lattice physics code calculations are described in Reference 8. Standard deviations between the gamma scans and TGBLA predictions were obtained for six elevations in the bundle. The results are summarized in Table 3.3 and are obtained from Table 3–1 in Reference 8.

Table 3.3
RMS Difference Between Gamma Scan and TGBLA Lattice Calculations for the Duane Arnold Special LTA Bundle

Elevation (inches)	RMS difference between gamma scan and TGBLA calculations

3.2 Conversion of Peaking Uncertainty to R-factor Uncertainty

The relationship between local peaking factor and R-factor is given by equation 3-1. The R-factor involves the square root of not one but five different local peaking values, and therefore the variance in the R-factor should be less than one half the variance in the local peaking. This relationship has been verified by picking a representative rod integrated peaking and R-factor distribution shown in Tables 3.4a and 3.4b. This peaking distribution, taken from an actual bundle design for a C-lattice BWR/4 application

Table 3.4a
Peaking Distribution For R-factor Uncertainty Trials

Table 3.4b
Base R–factor Distribution For R–factor Uncertainty Trials

Table 3.4c
R–factor Uncertainty Distribution

3.3 Nomenclature

Symbol	Definition
R_i	R–factor for fuel rod i.
T	Total number of lattice positions (81 for 9x9, 100 for 10x10).
F	Number of fueled rods in a lattice.
r_i	Fuel pin power peaking factor for rod i.
W_i	Weighting factor for rod i in R–factor calculation.
n_i	Number of rods in position i in R–factor calculation.
l_i	Additive constant for fuel rod position i.
σ_{peak}	Total fuel pin power peaking factor uncertainty.
σ_{mdl}	Fuel pin power peaking factor uncertainty due to nuclear model.
σ_{manuf}	Fuel pin power peaking factor uncertainty due to manufacturing tolerances.
σ_{bow}	Fuel pin power peaking factor uncertainty due to channel bow.
$\sigma_{R\text{-}factor}$	Total R–factor uncertainty.

4. Safety Limit MCPR Evaluation Methodology

4.1 Calculation Process

The revised calculation process is summarized in Figure 4.1, which is a flow chart describing the steps in the evaluation of the SLMCPR. This flow chart is identical to Figure IV–4 in Reference 4, with the exception of those processes and inputs which are shaded in Figure 4.1. The shaded boxes represent changes to the current NRC approved methodology. A general description of the process follows, with an explanation of the changes to the current procedure.

The process starts with a nominal reactor state calculation, performed with the 3D Simulator code PANACEA (Reference 3). Traversing in–core probe (TIP) readings are calculated using the bundle and nodal powers combined with TIP correlations. As a result of the PANACEA state calculation, the flow, void fraction, and power distribution are known for each bundle in the core. The R–factors for each bundle are known as a function of bundle exposure for each bundle and are accessed by the 3D Simulator along with the appropriate GEXL correlation constants to calculate the critical power ratio for each fuel rod in the core.

The reason for this methodology change is that it more realistically models the actual uncertainty in the integrated bundle power. The part of the uncertainty associated with the prediction of the TIP response and LPRM update is separated from the random power allocation uncertainty. In this way the correct amount of error correlation is obtained.

Figure 4.1
Calculation Procedure for SLMCPR Evaluations

4.2 Effect of Power Distribution on Safety Limit MCPR Evaluations

The first step in the statistical procedure is the simulation of the initial condition of the reactor. This initial condition is determined by the 3D Simulator code for which power, flow, inlet subcooling, pressure, control blade pattern and exposure distribution are input.

For the case of cycle specific SLMCPR evaluations, the exposure distribution is fixed by the projected operating strategy and the given state point in the cycle. The bundle R–factor distribution is also fixed by the bundle rod peaking distribution at a given exposure point. The only variables remaining are the power level and bundle power distribution.

The bundle power distribution is a function of the control blade pattern for a particular operating state. For a given state point, there are a variety of control rod patterns which produce a critical reactor and still satisfy thermal margin constraints. The initial condition is restricted to those conditions where the limiting bundles are close to MCPR limits while avoiding unreasonable power distributions and violation of LHGR limits. The objective in establishing the initial condition power distribution is to satisfy the total power and local limits and to reasonably bound the total rods expected to experience boiling transition. Therefore the control blade patterns chosen are those which reasonably bound the number of bundles which have critical power ratios close to the Operating Limit MCPR (OLMCPR). For reload core applications, which constitute all of the current applications, the control blade patterns which reasonably bound the number of bundles close to limits are those which result in an annular or cylindrical high power region at the center of the core. In a reload core, it is generally the fresh, or lowest exposure bundles which contribute to the population of high power bundles.

At the beginning of the cycle, the fresh or recently loaded bundles will have low powers because of gadolinium poison. However, the poison quickly burns away and the fresh bundles represent the majority of high power bundles from the middle to the end of the cycle. It has been found in general that the time in the operating cycle yielding the highest SLMCPR is usually either at the time of maximum core reactivity (peak hot excess reactivity) or at end of cycle, where the fresh bundles have the highest power. Most modern reload designs attempt to minimize the amount of neutron leakage out of the core by loading as many fresh high power bundles as possible near the center of the core. Therefore the highest density of fresh bundles will occur at the center, making the cylindrical high power region the one most likely to result in a larger number of bundles close to the operating limit.

Figure 4.2
Recommended Control Blade Patterns

Figure 4.3
Weighting Parameter, MIP

Figure 4.4
Frequency Distribution for Weighted CPR Parameter

4.3 Impact of Revised Uncertainties and Methods on SLMCPR Evaluations

The uncertainties and methods derived in this document have been used to evaluate cycle specific SLMCPRs for a number of plants. SLMCPR values are listed for two cases; the currently approved methodology and uncertainties, and the revised SLMCPR evaluation method combined with the current uncertainties. The results are summarized in Table 4.1. Note that the change in application of the power distribution uncertainties results in a lower calculated SLMCPR. This reduction is due to the fact that the bundle power distribution uncertainties are applied randomly for each bundle. Note that the reduction in SLMCPR for the revised methodology combined with the revised uncertainties is about 0.01 for the larger values of SLMCPR but is smaller for the cases with lower calculated Safety Limits.

Table 4.1
Summary of SLMCPR Values Using GETAB and Revised Methodology

Plant	Cycle	SLMCPR	
		GETAB Methodology	Revised Methodology

4.4 Nomenclature

Symbol	Definition
P_B	Probability of boiling transition in bundle B
P_n	Probability of boiling transition in fuel rod n
P_C	Probability of boiling transition in the core
N_{rod}	Number of fuel rods in a bundle
MIP	Weighted sum of fuel bundles near boiling transition
Δ_n	Difference in CPR ratio between bundle n and operating limit CPR
σ_{Gn}	GEXL uncertainty for bundle n

5. References

1. J .F. Carew, *Process Computer Performance Evaluation Accuracy*, NEDO–20340, June 1974.

2. *Process Computer Performance Evaluation Accuracy—Amendment 1*, NEDO–20340–3, Revision 2, August 1991.

3. *Steady State Nuclear Methods*, NEDE–30130–P–A, April 1985.

4. *General Electric BWR Thermal Analysis Basis (GETAB) Data, Correlation and Design Application*, NEDO–10958–A, January 1977.

5. Instrument Society of America, ISA–RP67.04, Part II, *Recommended Practice—Setpoint Methodologies*, September 1994.

6. *R–factor Calculation Method for GE11, GE12, and GE13 Fuel*, NEDC–32505P, November 1995.

7. *General Electric BWR Thermal Analysis Basis (GETAB) Data, Correlation and Design Application*, NEDE–10958–PA, January 1977.

8. L. M. Shiraishi, *Gamma Scan Measurements of the Lead Test Assembly at The Duane Arnold Energy Center Following Cycle 8*, NEDC–31569–P, April 1988.

9. Letter, J. S. Charnley to R. C. Jones, *Fuel Channel Bow Assessment* GENE Report MFN086–89, November 15, 1989.

10. *Power Distribution Uncertainties for Safety Limit MCPR Evaluations*, NEDC–32694P, December 1996.

11. *General Electric Fuel Bundle Designs*, NEDE–31152P, Revision 5, June 1996.

A. Appendix A

Responses to Request for Additional Information

January 8, 1998 (MFN-004-98)

I. Process Computer Uncertainties

1. **What is the variation of the weighing coefficients, used in evaluating the uncertainty in the Equation (2-3) inlet subcooling, over reactor statepoint (e.g., flow, power, subcooling, pressure) and in what sense are these coefficients conservative?**

The analysis is based on the heat balance relationship from Equation (2-2). In general this equation applies for any operational state where the core flow is positive. The fact that the component values that are used in Equation (2-2) can vary over a wide range is not the issue since the propagation of error due to uncertainties in the components that are used in evaluating Equation (2-2) depends on the relative magnitude of these uncertainties not the absolute value of the components. The fact that the weighting coefficients can also vary as the component values change is the fundamental aspect of the question.

The weighting coefficients used in evaluating the uncertainty in Equation (2-3) are the nominal values indicated in the last sentence at the bottom of page 2-5. For convenience these values are summarized here in Table I.1-1. It is the coefficients used in GETAB that lead to the nominal [[ ]] uncertainty in core inlet subcooling which in turn translates into a nominal core inlet temperature uncertainty of [[ ]] as indicated in Table I.1-1.

It is important to note that for purposes of calculating the Safety Limit a conservative core inlet temperature uncertainty of **0.20%** will be used as stated in the last paragraph of Section 2.5. This larger value corresponds to a core inlet subcooling uncertainty of [[ ]]. Relative to the weightings and component uncertainties previously used in GETAB this means that a factor of four margin is provided for on the nominal products of weighting factor times component uncertainty as indicated in the two rightmost columns of Table I.1-1.

Rather than use the weighting coefficients used previously in GETAB one can derive analytic expressions for the weighting coefficients by determining the partial derivatives of Equation (2-2) with respect to each quantity in accordance with Equation (2-4). Monte Carlo simulation was used to calculate how changes in the uncertainty of the component elements of Equation (2-2) cause the values for these partial derivative expressions to vary. When this is done there is no need to presuppose how the partial derivatives should be weighted since in fact the weighting varies according to how the terms combine in the expressions for the partial derivatives.

Table L1-1
Weighting Coefficients Used in Equation (2-3)

		Using GETAB Coefficients		Using GETAB Coefficients Multiplied by a Factor of 4.0	
Quantity	%sigma	weight	weighted variances	weight	weighted variances
W_{FW}	1.76	0.22	0.1499	[[	]]
W_T	2.50	0.59	2.1756	[[	]]
W_{CR}	5.00	0.01	0.0025	[[	]]
H_F	0.14	0.04	0.0000	[[	]]
H_{FW}	0.76	0.11	0.0070	[[	]]
H_{FG}	0.16	0.01	0.0000	[[	]]
F_{CU}	10.00	0.03	0.0900	[[	]]
Q_{CU}	1.00	0.01	0.0001	[[	]]
Q_P	2.00	0.01	0.0004	[[	]]
C	0.00	0.01	0.0000	[[	]]
H_{CR}	0.00	0.01	0.0000	[[	]]
Sum for Column		1.05	2.4256	[[	]]
%σ for Δh_0			[[		]]
absolute Δh_0 uncertainty			[[		]]
absolute temperature uncertainty			[[		]]
%σ for Temperature			[[ ]]		**0.20%**

One thousand Monte Carlo trials were used to evaluate Equation (2-2) whereby each uncertainty term was independently varied according to its standard deviation. The result is a total uncertainty in Δh_o of [[ ]] which corresponds to [[ ]] uncertainty in the core inlet temperature. Using this same approach it has been determined that to achieve the assumed 0.2% uncertainty in the core inlet temperature requires that all weighting factors or all uncertainties be simultaneously increased by a factor of [[ ]] Statistically this is interpreted as meaning that the acceptable level of 0.2% is [[ ]] standard deviations above the calculated mean value of [[ ]] which implies that at least [[ ]] of the GE BWR fleet is expected to be within the range covered by this evaluation when the suggested 0.20% uncertainty in the core inlet temperature is used.

The application range for the weighting coefficients can be characterized by considering how an allowable increase by an average factor of [[ ]] in the component uncertainties impacts the values obtained from the analytic evaluations of the partial derivatives that define the weighting coefficients. This [[ ]] shift factor in the underlying component uncertainties is intended primarily to account for the fact that the mean value of the coefficients used for this evaluation represents a sample from a range of possible values. Use of such a factor is consistent with statistical control theory practices where it is acknowledged that even an unbiased process will tend over time to drift such that lot

sample means commonly vary on the average in the range of ±1.5 standard deviations about the true mean.

For consistency the same set of random deviates used in the Monte Carlo simulation described in the previous paragraph are used in simulating the impact on the expressions for the partial derivatives. The mean values for the weighting coefficients and their percentage standard deviation for each weighting coefficient was determined from the results of this simulation and are presented in columns 2 and 3 of Table I.1-2. Note that the mean values are not impacted by [[ ]] shift factor on the component standard deviations since this factor serves simply to amplify the uncertainty in the population. On the average the standard deviations for the weighting coefficients are also increased by approximately this factor. The increase is not exact since the underlying component uncertainties may combine in a nonlinear way in defining the coefficient uncertainties (as determined by the expressions for the partial derivatives of Equation (2-2).

The tolerance bands established for the weighting coefficients assumes a $\pm 3\sigma$ band about the mean as is typically done in establishing tolerances. The standard deviations of the weighting coefficients already have imbedded in them the [[ ]] factor on the underlying component uncertainties, thus in Table I.1-2, the tolerances for the weighting coefficients are described as those based on "[[ ]] Perturbations in Component Uncertainties". The fact that this results in coefficients that sum in the absolute sense to be significantly less than unity for the lower tolerances and significantly greater than unity for the upper tolerances is accounted for by the normalization suggested in Equation (2-5). The final values for the coefficients obtained in this way are indicated in Table I.1-2. For convenience the coefficients have been normalized in terms of their fractional contribution to the total variance as suggested in Equation (2-5). This allows the values of the coefficients to be compared on a consistent basis with the values used previously in GETAB.

Notice that the GETAB weighting coefficients result in a total weighted variance that is larger than that obtained using the analytic derivative approach even when all the coefficients are evaluated at their upper tolerance values. This fact is indicated by the two columns in the lower half of Table I.1-2. The conclusion is that the original weighting used in GETAB leads to a conservative assessment of the core inlet subcooling and temperature uncertainties that is adequate to allow for uncertainties in how the weighting coefficients are determined.

Again it is worthy to reiterate that for purposes of SLMCPR evaluations the $\%\sigma$ for core inlet temperature uncertainty (0.2%) is a factor of [[ ]] greater than the [[ ]] value that is obtained from the analysis using GETAB uncertainties and weighting coefficients.

Table I.1-2

Demonstration that GETAB Weighting Coefficients Yield Conservative Results Relative to the Expected Values even when Liberal Tolerances are Assumed

[[

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2. **How does the channel flow uncertainty account for channel bulge and non-uniform crud/corrosion build up on the fuel rods?**

The [[ ]] standard deviation applied to the channel flow area is derived from manufacturing tolerances on the mechanical components which define the flow path as indicated in Section 2.7 of NEDC-32601P. This amount does not include the channel flow uncertainty to account for channel bulge and non-uniform crud/corrosion build up on the fuel rods. These latter effects are accounted for by the 5% uncertainty that is applied on the channel to channel friction factor multiplier as is explained in Section 2.9 of NEDC-32601P. The basis is the same as given in Section 3.2.2.1 of NEDO-20340 and Section IV-3-4-2 of GETAB.

3. **Provide justification for neglecting the bias in the core pressure drop calculation in Table 2.2**

The mean of the [[ ]] samples for measured total pressure drop is [[ ]] psi. The corresponding mean for the calculated total pressure drop for these same [[ ]] cases is [[ ]] psi. The average calculational bias in total core pressure drop is thus [[ ]]. This corresponds to an overestimation in the core flow of roughly [[ ]]. The change in the calculated SLMCPR for such a small increase in flow is negligible.

4. **In Table 2.2, the fact that only the BWR6 data is negative suggests that the uncertainty is plant dependent. Also, the fact that 50% of the data is outside the one-sigma interval suggests that the data is not normal. Provide justification for treating this uncertainty as normally distributed in the SLMCPR analysis.**

The "data" in Table 2.2 that is modeled in the simulation is the total measured pressure drop data in column 3. The mean of this data is [[ ]] psi and its standard deviation is [[ ]] psi. Because these values were taken from different plants they are samples from different populations and thus the standard deviation of the composite population would be expected to be greater than the standard deviations associated with each population individually. Nevertheless, samples taken randomly from multiple populations are generally expected to be normally distributed if the individual populations are normal. A statistical hypothesis test for the normality of the composite population was conducted. According to the Anderson-Darling statistical test, normality cannot be rejected even at the [[ ]] significance level as is indicated in Figure I.4-1. (Typically a significance level less than 0.05 to 0.10 is required to accept the null hypothesis that the data is not normal.)

When one considers the comparisons between the calculated and measured data as shown in column 4 of Table 2.2, one is accounting for the fact that each calculated value is paired with a particular measured value. The fact that each pairing comes from a different

population is not relevant since what is being quantified is the fidelity of the calculations relative to their respective measurements. The calculated-measured data shows that on the average the calculations are not significantly biased relative to the measurements. It is interesting to note that the calculations in all [[ ]] cases predict the measurement within [[ ]] psi which is indeed within the [[ ]] psi standard deviation indicated by the composite set of measured data. The mean of this data is [[ ]] psi and its standard deviation is [[ ]] psi. According to the Anderson-Darling statistical test, normality cannot be rejected even at the [[ ]] significance level as is indicated in Figure I.4-2.

The [[ ]] psi value referred to in the text may be a source of confusion. The value in the text has no relationship with the [[ ]] in the last row of column 4 of Table 2.2. The [[ ]] psi value in the text originates from the ATLAS data as initially cited in the fourth sentence of the first paragraph of Section 2.8 on page 2-7.

[[

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Figure I.4-1 Normality Test for the Absolute Core Pressure Drop Values

[[

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Figure I.4-2 Normality Test for ΔP (calculation - measurement) Values

5. How is the uncertainty in the bypass flow included in the uncertainty analysis?

This question was previously addressed in response to Request No. 5 in *Process Computer Performance Evaluation Accuracy, Amendment 2* (NEDO-20340-2, Class I, September 1975). That response is still accurate and is included by its reference here.

II. R-Factor Uncertainty

1. What specific fuel designs were the TGBLA-to-MCNP pin power comparisons performed and how do these comparisons cover the intended range of GE BWR fuel designs?

The specific fuel designs are listed in Table 3.1 of the Reference document. They cover all GE designs currently operating in BWR's. Both the original 8x8 and the GE9 8x8 with a large central water rod are included, as well as the 9x9 and 10x10 designs. One competitor design, the SVEA96 design is also included. The RMS differences between TGBLA and MCNP for the SVEA design are not significantly different than those obtained for the GE designs.

2. **How is the uncertainty in the TGBLA exposure calculation accounted for in the determination of the local pin power peaking factor uncertainty?**

[[

]]

3. **Provide justification for the weight used to combine the standard deviations of Table 3.1 and determine the local peaking model uncertainty.**

The weighting is based on standard methods for obtaining the overall variance of a group of observations, i.e.,

$$\sigma = \sqrt{\sum_{i=1}^{k}(n_i - 1)\sigma_i^2 \ / (\sum_{i=1}^{k} n_i - k)} \quad \text{where } n_i = \text{sample size for sample I}$$

Here the sample size is the number of pins in the lattice.

4. **Is the enrichment tolerance for rods with enrichment less than 3.5% greater than for rods with enrichment greater than 3.5% and, if so, how is the resulting increased uncertainty accounted for in the local peaking model uncertainty?**

[[

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5. **Provide justification for the assumption that the effect of rod position (e.g., due to rod bowing) has a negligible effect on the local power peaking.**

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6. **Describe the measurements of Table 2.2 and how they determine the single-phase friction factor uncertainty.**

   The measured pressure drop values from Table 2.2 were obtained from plant instrumentation. Because these values represent samples from different populations as discussed in response I.4, the variance in the measured values cannot be used directly to determine the uncertainty in the friction factor because it consists of at least two separate components. The total variance of the measured values consists of (1) variance in the pressure drops from plant type to plant type and plant to plant, and (2) variance in the core pressure drop values for a particular plant. For plant/cycle specific analyses, the variance in pressure drop values from plant to plant is not relevant since each analyses is performed for a particular plant/cycle for which the nominal core pressure drop value is calculated. It is the variance in the measurements relative to this particular calculated value that is important.

   For each plant in Table 2.2 of NEDC-32601P, the difference in the calculated value minus the measured value should be expressed as a percentage of the calculated value since it is the amount of uncertainty to be applied to the calculated value that is sought. These percent differences are shown in the last column of Table III.6-1 below. Since these differences are based on total core pressure drop values, they already implicitly include the variances due to the two-phase pressure losses; nevertheless, the values described in the text of Section 2.8 of NEDC-32601P take no credit for this fact.

Table III.6-1
Determination of %Uncertainties in
Calculated Core Pressure Drops

[[

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7. **The comparisons of Table 3.3 indicate that the local power peaking factor uncertainty is larger at the top of the fuel bundle. How is this apparent spatial dependence of the peaking factor uncertainty accounted for in the SLMCPR evaluation?**

The peaking factors which are used in Equation 3-1 are axially integrated values, according to the procedures specified in the R-factor topical report (NEDC-32505P). Since the peaking factors are integrated values, an equal axial weighting was deemed appropriate for the overall uncertainty in the pin power peaking. [[

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Table II.7-1
Void Fraction Weighting

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8. **Do the TGBLA-to-MCNP comparisons of Table 3.1 indicate a larger uncertainty in the high powered fuel rods and, if so, how is this accounted for in the SLMCPR?**

[[

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Table II.8-1
TGBLA - MCNP Bias as a Function of Pin Power Peaking

[[

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Figure II.8-1
8x8 Pin Power Error vs Pin Power

[[

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Figure II.8-2
9x9 Pin Power Error vs Pin Power

[[

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9. How is the uncertainty in the fuel density accounted for in determining the manufacturing uncertainty?

A number of manufacturing process capability studies have been carried out to determine the fuel density variations from rod to rod. These studies were carried out on approximately [[

]] Statistically combining this small uncertainty with the enrichment and other manufacturing uncertainties is documented in the response to Question II.4 above.

10. How is the uncertainty in the local voids and exposure accounted for in the local peaking factor uncertainty?

The variation in void fraction in a lattice can be due to two effects. The first is due to the uncertainty in lattice void fraction. The second is variation in void fraction as a function of position in the lattice. We have evaluated the sensitivity of pin power peaking due to average lattice void fraction [[

]]

11. **In the improved R-Factor method, how is the uncertainty in the bundle-average void and exposure distributions used in performing the integration of the local R-Factor accounted for in the determination of the R-Factor uncertainty?**

In the improved R-factor method, the axial integration of the pin power is carried out with generic axial power shapes. The radial pin power distributions for each axial height are obtained from generic void and exposure distributions. [[

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Table II.11-1
Difference in Bundle R-factor (Outlet peak - Generic Uncontrolled)

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12. **The local power peaking modeling uncertainty provided on p.3-2 is greater than the value given in the first sentence of Section 3.1.4. Please explain this apparent inconsistency.**

    [[ ]]

13. **The power peaking uncertainties in neighboring fuel rods are generally correlated and, consequently, cannot be taken to be independent and random as assumed in the calculations of Section 3.2 Provide an estimate of the R-Factor uncertainty that recognizes the correlation of the uncertainties in neighboring fuel rods.**

    The pin power peaking errors for an 8x8 and 9x9 lattice developed in the answer to question number 8 have also been analyzed for possible correlation of model errors due to position in the lattice. We have found that the pin power error [[ ]] These positions are illustrated in Figure II.13-1.

Figure II.13-1

[[

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The average pin power bias and standard deviation are summarized in Table II.13-1 for the 8x8 and 9x9 lattice.

Table II.13-1
TGBLA - MCNP Bias as a Function of Pin Power Peaking

[[

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[[

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III. SLMCPR Evaluation Methodology

1. **In Section 4.3, it is stated that the SLMCPR values of Table 4.1 are for the revised methodology and the present (larger) uncertainties, while Table 4.1 indicates that the revised methodology is evaluated with the revised uncertainties. Please explain this apparent inconsistency.**

   Table 4.1 is confusing. The "revised uncertainties" are those identified in Table 2.1. Those uncertainties that have been revised as indicated in Table 2.1 have negligible impact on the calculated values, thus the differences shown in Table 4.1 are due entirely to the change in methodology since the power distribution uncertainties are unchanged from the (larger) power distribution values presented in GETAB. The label in Table 4.1 will be changed to make this clarification. Differences due to reducing the power distribution uncertainties are not addressed until NEDC-32694P.

2. **Provide the basic mathematical definition of Wcore (rather than the mathematical result of the integration given in Section 4.2).**

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[[

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[[

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[[

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3. Provide the definition of the MCPR Importance Parameter (MIP) of Figure 4.4.

[[

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4. What pin power distribution is assumed in the definition of Wcore and what is the effect of this assumption?

[[

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5. Describe the selected 100 nominal control rod pattern cases of Figure 4.4. How do these cases and the Figure 4.4 comparisons of the nominal and limiting MIPs accommodate off-nominal operating statepoints. Provide justification for the assumption that these 100 patterns bound the operating statepoints.

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[[

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[[

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[[

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[[

Figure III.5-1 Updated Version of Figure 4.4 from NEDC-32601P

Figure III.5-2 How MIP Ratio is Related to CPR Margin

Figure III.5-3

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Table III.5-1
Details for the Nominal-to-Limiting MIP Comparisons
Used to Construct Figure 4.4 of NEDC-32601P

[[

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6. **Provide specific details describing how the limiting control rod pattern will be selected in a typical reload core determination of the SLMCPR. What quantitative criteria will be used to select the limiting pattern? What statepoints (e.g., power, flow, exposure) will be included in the determination of the limiting pattern? How will the Figure 4.4 comparison be used to confirm the limiting pattern selection? Will the data base of these 100 nominal cases be expanded or updated?**

[[

Table III.6-1
Thermal Margin and Reactivity Constraints for Determining Limiting SLMCPR Power Distribution

]]

7. **Will the Wcore parameter be monitored by the process computer to insure that the design SLMCPR limiting power distribution bounds the operating power distribution?**

[[

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[[

Figure III.7-1

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8. **Do all the MIP calculations of Figure 4.4 assume the same uncertainties? If not, provide the nominal-to-limiting MIP comparisons separately for each set of uncertainties and explain the effect of this inconsistency on the conclusions drawn from the Figure 4.4 comparisons.**

[[

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9. **The elimination of fuel bundles from the SLMCPR calculation using the criteria $P_b < \lambda P_c$ results in an underestimate of the number of rods in boiling transition. Provide an estimate of the effect of this nonconservatism.**

Consider the largest BWR core in the United States consisting of 800 fuel bundles. Next consider a 10 by 10 BWR bundle with 92 fuel rods. Conservatively assume that all the bundles have a probability of contributing rods susceptible to boiling transition that is equal to the rejection threshold of $P_c \times 10^{-9}$. Also conservatively assume that all 92 rods in these fuel bundles will participate at the same level as the lead rod. For this extremely conservative scenario the relative error in calculating the total probability for boiling transition in the core is bounded by

$$100\% \cdot \left[P_c \cdot 10^{-9} \cdot N_{bundles} \cdot N_{rods/bundle}\right] / P_c \leq 0.00736\% \qquad \text{(III.9-1)}$$

The equation above reduces to the equality in the limit as all rods are rejected and the calculated probability P_c approaches zero. For the case where the calculated number of rods susceptible to boiling transition approaches 0.1% of the rods in the core, the total population of rods that are rejected is reduced by the factor (1-0.001)=0.999 and the relative error goes down by the same factor. For even higher percentages of rods calculated to be susceptible to boiling transition, the maximum relative error goes down even more. Thus the total relative error is always less than 0.00736% for even the largest core and goes down for cores with fewer fuel rods.

Since this aspect of the calculations has not changed, Section IV-3-6 of Reference 1 can be consulted for additional details.

10. **The proposed SLMCPR methodology differs from the presently approved generic methodology. The new method appears to be less conservative with respect to: (1) the selection of the initial CPR distribution, (2) determination of the limiting control rod pattern, (3) termination of the search for maximum SLMCPR and (4) the use of an equilibrium rather than a xenon-free xenon distribution. These specific concerns were identified and described in the NRC Inspection Report No. 99900003196-01 (Letter U.S. NRC to C. P. Kipp (GE), dated September 10, 1996). Provide justification for these changes that have been included in the proposed SLMCPR methodology.**

The response to this question is contained in the submittal accompanying GESTAR II Amendment 25 (See Reference 2).

11. **The revised methodology in which the power distribution model uncertainty is assigned on an individual rather than a four-bundle basis results in a (non conservative) decrease in the SLMCPR. This revision is based on the assumption that the modeling uncertainty in neighboring fuel bundles is uncorrelated. In order to justify this revision, provide benchmark comparisons for the nodal bundle powers demonstrate that the modeling error in adjacent fuel bundles is not correlated.**

The bundle power allocation factors generated by the PANACEA BWR simulator have been compared to Gamma scan data in Table 3.2 of NEDC-32601P. These errors follow a normal distribution as illustrated in Figure III.11-1 below. [[

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Figure III.11-1
Summary of Power Allocation Error Statistics

[[

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Figure III.11-2
Power Allocation Error vs Power Allocation Factor

[[

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References

1. General Electric BWR Thermal Basis (GETAB): Data, Correlation and Design Application, NEDO-10958-A, Class I, January 1977.

2. Amendment 25 Supporting Information, provided as an attachment to the GESTAR II Amendment 25 submittal.

Attachment A
BWR Fuel Void Fraction

B. Appendix B

Responses to NRC Request for Additional Information

April 17, 1998 (MFN-014-98)

II.4 The Table 2.2 comparisons of calculated and measured pressure drop indicate a nonconservative 0.75% overprediction of the core flow. Noting that the SLMCPR accounts for random variations rather than systematic biases, how is this flow bias accounted for in the analysis?

On the average, the over-prediction in core flow cannot be shown to be statistically significant. This fact will be established in the first part of the response that follows. Nevertheless, it appears that the concern behind the question is that specific over-predictions in core flow may lead to potentially non-conservative calculated SLMCPR values. This concern will be addressed in the latter part of this response.

Table 2.2 of NEDC-32601P and Table II.6-1 in response to question II.6 from the previous request for additional information (RAI) both show that the average bias in calculated core pressure drop versus measured pressure drop is [[ ]] psi. On the average this corresponds to an over-prediction in the core pressure drop by [[ ]] relative to the measurements as indicated below in Table II.4-1. Note that this is slightly higher than the [[ ]] indicated in the text of response I.3 or the [[ ]] average indicated in Table II.6-1 in the responses to the previous RAI. The previous values were given relative to the slightly higher calculated values instead of with respect to the slightly lower measured values. Also, the calculated values provided previously were erroneously tabulated and these erroneous values had been used in determining the percentage differences. The correct calculated core ΔP values are tabulated below.

Table II.4-1

(corrected and augmented Table II.6-1 from previous RAI)

Determination of %Uncertainties in Calculated Core Pressure Drops and Corresponding % Errors in Core Flow

[[

]]

Quantities reported in the original document and the associated conclusions do not change. The previous responses to questions I.3, I.4 and II.6 are supplemented by the additional information provided below.

A two-sample "T-test" was performed to show that the difference in the average calculated core ΔP versus the average measured core ΔP is not statistically significant. The results from the test are given in Table II.4-2.

Table II.4-2
Two Sample T-Test and Confidence Interval for
Calculated versus Measured ΔPs

[[

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As indicated by the high "P value" of 0.80, one should **not** assert that the calculated values are biased relative to the measured values. Such an assertion is not meaningful in view of the standard deviation in the two populations. The difference in the two means would have to be [[ ]] before such an assertion would be warranted. Incidentally, the standard deviations of the two populations calculated individually are in remarkable agreement so it is appropriate that the T-test was performed using the pooled standard deviation of [[ ]].

For the [[ ]] comparisons of calculated total core ΔP versus measured ΔP indicated in Table II.4-1, the error in the total core pressure drop ($\delta\Delta P\%$) is used to estimate error in core flow ($\delta W\%$) as follows:

$$\delta W\% = 100*(\sqrt{1+\frac{\delta\Delta P\%}{100\%}}-1) = 100\%*(\sqrt{\frac{\Delta P_{calc}}{\Delta P_{meas}}}-1)$$

The specific core flow errors for the [[ ]] samples are shown in the rightmost column of Table II.4-1. On the average the calculated core flow is [[ ]] as shown near the bottom of the last column. This amount of difference is not statistically significant since it is derived from the insignificant "bias" in calculated ΔPs versus measured ΔPs. Also note that the average of [[ ]] is even lower than the [[ ]] core flow "bias" reported previously because the [[ ]] was derived from an average [[ ]] ΔP "bias" (should have been [[ ]]) whereas the new lower

value is based on the average of the last column in Table II.4-1. The standard deviation for the [[ ]] core flows is [[ ]] and is bounded by the total 2.5% uncertainty that is used in calculating SLMCPRs (see Section 2.6 of NEDC-32601P).

The anticipated range for specific core flow deviations for 95% probability at 95% confidence is[[ ]]. The range is fairly broad [[ ]]. Previously the concern with how SLMCPR values may change as a function of core flow were addressed to support the practice of calculating SLMCPR values [[ ]]. The fact that the calculated SLMCPR is [[ ]]. An example of results obtained earlier for this scenario is provided in Figure II.4-1.

[[

Figure II.4-1 Example of Calculated SLMCPR Sensitivity to Core Flow

]]

The results from this sensitivity study and other similar calculations support the conclusion [[

SLMCPR would be negligible for a very wide range of flows]]. Therefore, the practice of [[ ]] in the SLMCPR calculations is justified as is the practice of performing the calculations [[ ]].

II.5 The effect of the fuel rod bowing displacement on the pin power is based on the rod to rod power gradient. It is not evident that this approach accounts for the change in the pin power distribution resulting from the bowing displacement. For example, is the gradient calculated for the situation in which the fuel rod is displaced.

The effect of rod bow has been evaluated using the Monte Carlo transport method. These evaluations show that for the majority of fuel rods, the change in rod power due to displacement of a rod is [[ ]]. The rods analyzed are shown in Figure II.5-1 for the directions shown in the Figure. These calculations show that if a given rod is displaced toward a rod with higher power [[ ]]. If a rod is displaced toward a rod with lower power, [[ ]]. For example, the Monte Carlo studies show that if rod **a** is displaced toward rod **b**, then the change in rod **a** power is given by

[[ ]]

[[

]]

Figure II.5-1

Rod Bow Perturbations

[[

]]

Figure II.5-1

Rod Bow Power Uncertainties

[[

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II.4/II.9 The evaluation of the effect of uncertainty in fuel rod density assumes that a 1% error in fuel density results in a 1% error in fuel rod power. A similar assumption is made in the evaluation of the effect of uncertainty in fuel rod enrichment. Additional justification should be provided to support these assumptions.

The fission density, which is the majority of the power in a given fuel rod, is evaluated by the following equation:

$$\mathrm{FD} = \sum_{i} \mathrm{N}_i \sum_{g} \sigma_{fi}^{g} \phi^{g}$$

where the summation over i is for all the nuclides in a region, and the summation over g is for energy groups. The uncertainty evaluation assumes [[

]]

II.10 The response to this question does not indicate how the uncertainty in local fuel exposure is accounted for in the local peaking uncertainty.

Local fuel exposure can affect pin power peaking. [[

]] The results are summarized in Figure II.10-1.

Figure II.10-1

Rod Power Uncertainty as a Function of Bundle Exposure

[[

]]

II.13 The response to this question indicates that the errors in neighboring fuel rod powers are correlated. While the response discusses the effect on the rods adjacent to the water gap, it is noted that the R-factor is required for all rods in the fuel bundle. The R-factor uncertainty should account for the fact that the modeling errors in the calculation of the neighboring fuel rod powers is correlated.

[[

]] Table II.13-1 is reproduced below:

Table II.13-1

TGBLA - MCNP Bias as a Function of Pin Power Peaking

[[

]]

The resultant average bias and uncertainties for each fuel rod are summarized in Table II.13-2 below. [[
]]

Table II.13-2

Average R-factor Biases and Uncertainties as a Function of Fuel Rod Position

[[

]]

[[

]]

Figure II.13-1

Frequency Distribution for R-factor Uncertainties

[[

]]

III.11 The present GETAB methodology assigns the bundle power error to the four bundles surrounding the TIP in a correlated manner so that each of the four bundles is perturbed simultaneously by the same amount. In the proposed methodology, the modeling error in these four bundles is assumed to be uncorrelated and the individual bundle powers are varied independently during the uncertainty propagation. Because of the increased random variability in the proposed methodology the SLMCPR is reduced. Additional justification is required to support the assumption that the modeling errors in these four bundles is uncorrelated.

[[

]]

The reason for this methodology change is that it more realistically models the actual uncertainty in the integrated bundle power. The part of the uncertainty associated with the prediction of the TIP response and LPRM update is separated from the random power allocation uncertainty. In this way the correct amount of error correlation is obtained."

References

1. "Request for Additional Information for GE Topical Reports NEDC-32601P and NEDC-32694P," Letter, J.H. Wilson (NRC) to R. J. Reda (GE), dated August 20, 1998.
2. "Responses to Request for Additional Information for GE Topical Report NEDC-32694P, Power Distribution Uncertainties for Safety Limit MCPR Evaluations," GAW-98-003, Letter, G. A. Watford (GE) to U.S. NRC, dated January 9, 1998.
3. "Responses to Request for Additional Information for GE Topical Report NEDC-32601P, Methodology and Uncertainties for Safety Limit MCPR Evaluations," GAW-98-002, Letter, G. A. Watford (GE) to U.S. NRC, dated January 8, 1998.

C. Appendix C

Responses to Request for Additional Information

July 29, 1998 (MFN-017-98)

The Reference 1 response to RAI 11.13 provides an evaluation of the adequacy of the proposed R-Factor uncertainty for the 9x9 fuel lattice. Since the results of this analysis depend on the specific fuel lattice design, an additional evaluation is required for the 10x10 lattice design.

The rod power modeling uncertainty includes two components: (1) the uncertainty due to the infinite lattice TGBLA calculation and (2) the uncertainty introduced by assuming the fuel bundle is in an infinite lattice of identical fuel bundles, rather than in a core location surrounded by different fuel bundles. The correlation of the TGBLA calculational uncertainties is determined by comparison with MCNP, and is accounted for in the R-Factor uncertainty analysis of Reference-11. However, the Reference-1 analysis provided in RAI 11.13 does not account for the correlation of the calculational uncertainties due to the infinite lattice assumption. The calculational uncertainties introduced by assuming the fuel bundle is in an infinite lattice are expected to have a rod-to-rod correlation, since an adjacent Del bundle' will have a similar effect on all rods facing the bundle For example, if the adjacent bundle has a higher power, the outside rods facing this bundle will generally be underpredicted by the TGBLA calculation.

Therefore, provide justification for neglecting the effect of the correlation of the modeling uncertainties introduced by the infinite lattice assumption, or account for this effect in the R-Factor uncertainty.

Response

It is agreed that the uncertainty due to the infinite lattice assumption is correlated with position in the lattice. Therefore the overall R-factor uncertainty has been re-evaluated accounting for this correlation. In the process of the evaluation, it was recognized that in addition to the infinite lattice uncertainty, the uncertainty due to channel bow is also correlated with lattice position, and this fact has also been accounted for in the new evaluation. The new evaluation has been carried out for both 9x9 and 10x10 lattice configurations.

In the new evaluation, the infinite lattice factors and manufacturing uncertainties are assumed to vary independent of fuel rod position. The Monte Carlo to diffusion theory bias is accounted for by row, where row 1 is next to the channel, row 2 is the next row in from the channel and row 3 is the interior rods. In the original R-factor uncertainty analysis, an

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1 Responses to NRC Request for Additional Information Associated with SLMCPR Methodology and Uncertainty Topical Reports NEDC-32601P and NEDC-32694P," GAW-98-009, Letter, G. A. Watford (GE) to U.S. NRC, dated April 17

uncertainty of [[ ]] was used for the gradient effect and [[ ]] was used for the uncertainty in channel bow. In the revised analysis, these two effects are combined in a manner that accounts for the correlation with lattice position.

The lattice position related uncertainties in each row are determined as follows.

Gradient Effect First the relative variation for each row is determined by running the lattice physics code TGBLA. [[

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Figure 1

Relative Power Change for Incoming Leakage

[[

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In this case, the boundary condition represents a net number of neutrons entering the bundle on the left, and therefore an increase in power for pins near that boundary.

]] This function can now be used to determine the relative amount of pin power change in each row of fuel rods due to neutrons leaking in or out of a given bundle.

Let L_j represent the average of the leakage perturbation in row j evaluated from the function in Figure 1. [[

]]

Channel Bow Effect The channel bow distribution is determined in a manner similar to the method used for the gradient effect used above. In this case the lattice model is used to calculate the power peaking ratio due to a shift in the lattice position relative to a control blade. GE already computes the change in peaking due to channel bow effects and so this distribution is readily available. The average relative channel bow impact B_j for row j has the same definition as L_j above. [[

]] (2)

Table 1 summarizes the row-by-row uncertainties for the flux gradient and channel bow effect for 9x9 and 10x10 lattices.

Table 1

Row Pin Power Uncertainties For Channel Bow and Flux Gradient Effects

[[

]]

<u>Effect on R-factor</u>

The effect on R-factor uncertainty is determined by first evaluating the impact on pin powers. The detailed process for the variation of the pin powers is represented as

[[(3)

]] (4)

[[

]]. The standard deviation for the three rows has been summarized in Table 1.

Figure 1

Application of Pin Power Uncertainties for Gradient and Channel Bow Effects

[[

]]

A large number of trials are then performed where in each trial, the pin-by-pin powers are perturbed according to equation (4). After each perturbation, the pin powers must be renormalized to preserve the average pin power equal to 1.0. Next, for each trial, the pin powers are used to calculate the R-factor for each pin in the lattice. These R-factors are then divided by the base R-factors calculated for the nominal lattice configuration. Statistics on this resultant distribution have been determined.

The standard deviation results are summarized in Table 2 for the 9x9 lattice and in Table 3 for the 10x10 lattice. [[

]]

Table 2

9x9 R-factor Biases & Uncertainties

[[

]]

Table 3

10x10 Biases & Uncertainties

[[

]]

Figure 3

9x9 Uncertainty Distribution

[[

]]

Figure 4

10x10 Uncertainty Distribution

[[

]]

D. Appendix D

Responses to Request for Additional Information
September 9, 1998 (MFN-038-98)

The R-factor uncertainty analysis has been completed for 9x9 and 10x10 fuel and documented in the Letter from GA Watford to J. H. Wilson "Additional Information Associated with SLMCPR Methodology and Uncertainty Topical Report NEDC-32601P" July 29, 1998. In this R-factor uncertainty analysis, uncertainty values were evaluated for each row of fuel rods beginning with the rods next to the water gap and ending with the interior rods. We have now evaluated these same uncertainty factors for 8x8 fuel, using the same process as the 9x9 and 10x10 lattice as documented in the above reference.

The uncertainty factors are summarized in Table 1, which has the same format as Table 1 in the above reference, but includes the 8x8 uncertainties. [[

]]

Table 1

Row Pin Power Uncertainties For Channel Bow and Flux Gradient Effects

[[

]]

E. Appendix E

Responses to Requests for Additional Information

March 1, 1999 (MFN-002-99)

GE Nuclear Energy

General Electric Company
P. O. Box 780, Wilmington, NC 28402

March 1, 1999

MFN-002-99
GAW-99-001

Document Control Desk
Nuclear Regulatory Commission
Washington, D. C. 20555-0001

Attention: M. J. Davis/T. L. Huang

Subject: Additional Information Associated with SLMCPR Methodology and Uncertainty Topical Reports NEDC-32601P and NEDC-32694P

This letter transmits additional information requested by the staff in a phone conversation of February 22, 1999 with Dr. Huang.

For the introduction of new fuel designs, the following will be performed:

1. An evaluation of fuel rod power uncertainties and their impact on the SLMCPR process
2. An evaluation of the effects of nodal power calculations on R-factor uncertainty
3. An evaluation of the power distribution uncertainties and their impact on the SLMCPR process

In addition, the following SLMCPR procedural steps are currently in place and will be maintained:

1. The MIP criteria is evaluated for each new fuel and core design on a cycle specific basis
2. The applicability of the MIP criteria is periodically reviewed during the Technical Design Procedure Annual Adequacy Review.

If you need further information, please feel free to contact me.

Sincerely,

Glen A. Watford, Manager
Nuclear Fuel Engineering
(910) 675-5446

cc: S. P. Congdon
J. Andersen
A. Lingenfelter