May 18, 1984

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Mr. E. E. Utley
Executive Vice President
Carolina Power & Light Company
Post Office Box 1551
Raleigh, North Carolina 27602

Dear Mr. Utley:

SUBJECT: BRUNSWICK RELOAD LICENSING METHODOLOGIES

Re: Brunswick Steam Electric Plant, Units 1 and 2

By letter dated March 10, 1983 you submitted for our review three reports dealing with reload analysis methods to be applied to your Brunswick units. The three reports are:

1. "A Description and Validation of Steady-State Analysis Methods for Boiling Water Reactors," CP&L NF-1583.01.

2. "Methods of PRESTO-B, A Three-Dimensional LWR Core Simulator Code," CP&L NF-1583.03. (Proprietary and Non-Proprietary Versions).

3. "Methods of RECORD, An LWR Fuel Assembly Burnup Code," CP&L NF-1583.02 (Proprietary and Non-Proprietary Versions).

On November 1, 1983 we requested additional information and on January 3, 1984 you supplied the additional information.

We have completed our review of these reports and the additional information you provided. We find the reports to be acceptable for reference in licensing submittals related to reload fuel analyses for Brunswick Units 1 and 2. Our evaluation of the three reports are enclosed.

Sincerely,

Original signed by RAHermann for/

Domenic B. Vassallo, Chief
Operating Reactors Branch #2
Division of Licensing

Enclosures:
1. Evaluation of Report NF-1583.02
2. Evaluation of Report NF-1583.03
3. Evaluation of Report NF-1583.01

cc w/enclosures:
See next page

	SM		
DL:ORB#2	DL:ORB#2	DL:ORB#2	DL:ORB#2
SNorris:ajs	SMacKay	MGrotenhuis	DVassallo
05/16/84	05/17/84	05/17/84	05/18/84

Mr. E. E. Utley
Carolina Power & Light Company
Brunswick Steam Electric Plant, Units 1 and 2

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ENCLOSURE 1

Evaluation of Report NF-1583.02

1. Introduction

Carolina Power and Light, in anticipation of the performance of core reload analyses for their boiling water reactors, has presented three reports for review by the NRC. These reports are:

NF-1583.01, "A Description and Validation of Steady-State Analysis Methods for Boiling Water Reactors"

NF-1583.02, "Methods of RECORD: An LWR Fuel Assembly Burnup Code"

NF-1583.03, "Methods of PRESTO-B: A Three-Dimensional BWR Core Simulation Code"

The Core Performance Branch has reviewed these reports. We had the assistance in the review of our consultant, Brookhaven National Laboratory, under Technical Assistance Contract FIN A-3407.

The present evaluation is for report NF-1583.02. The other reports are the subject of separate evaluations. The second section of the evaluation summarizes the report contents and the third presents a summary of our evaluation. A statement of the evaluation procedure and the conclusions (regulatory position) of our evaluation follows.

2. Description of Report

RECORD is a detailed reactor physics code for performing light water reactor fuel assembly (lattice) calculations, taking into account most of the features found in BWR and PWR fuel designs. The code calculates neutron spectra, reaction rates and reactivity as a function of fuel burnup, and generates the few-group data required by full scale core simulators - in particular the PRESTO code which will be used by CP&L. Report NF-1583.02 describes in detail the basic models used and the techniques and procedures employed in RECORD. Since the report is proprietary, what follows is a discussion of the code in general terms only.

The report first presents an overview of the method (Section 2) and then provides details of the separate components. Section 3 describes the calculation of the thermal neutron spectrum and Section 4 describes the epithermal and fast spectra calculations. Section 5 and 6 describe the treatment of burnable poison and control absorbers respectively. Section 7 describes the two-dimensional assembly flux and power distribution calculation and homogenization procedures. Section 8 describes the burnup calculations and Section 9 describes additional calculations (e.g., TIP instrumentation factors) which are required by the core simulator. Finally, Section 10 presents comparisons with experiments which have been used to verify that RECORD successfully calculates the neutronics parameters for light water reactor fuel assemblies.

The calculation procedure employed in RECORD is similar to that used in other lattice physics codes and proceeds through successive stages of assembly homogenization and larger group sizes (fewer groups). The basic cross-section information is that from the ENDF/B-III data files with certain additions and emendations to correct known deficiencies. The essentially infinite group data from the files is first reduced to 50 energy troups - 15 thermal groups covering the range from 0 to 1.84 ev, and 35 epithermal groups covering the range from 1.84 ev to 10 Mev. This group structure is used to obtain the neutron spectra for the pin cell consisting of a fuel rod and its associated clad and moderator. A single average pin cell for the entire assembly is calculated in the epithermal region but several (typically six) diffusion subregions are defined for the thermal calculation in BWR assemblies and a pin cell calculation is performed for each subregion. The pin cell is homogenized and a thermal and epithermal flux spectrum is obtained for each cell type (subregion).

The next step in the process is the calculation of the flux distribution across the fuel assembly and associated water gaps along with the reactivity (k_∞ or k_{eff}) of the assembly. For this purpose the entire assembly is modeled in x-y geometry with each pin cell explicitly represented as well as the additional water, channel box and control rod (if present). The 50 group cross-section set is reduced to five groups (two thermal and three epithermal) for this calculation. The calculated flux distribution is used to homogenize the assembly and the homogenized cross-section set is reduced to a two group representation for use in the simulator (PRESTO-B) code.

Each step of this process is described in detail in the report. In the thermal energy range the neutron transport equation is solved by the point energy approach. The Nelkin scattering model is used for hydrogen bound in water and the Brown-St.John model is used for oxygen. A modified Amouyal-Benoist method is used to obtain the energy dependent flux ratios in the fuel, clad, and moderator of the pin cells. The average fuel pin in each spectral subregion in the assembly is obtained by averaging the isotopic content of all the pins in that sub-region.

For the epithermal spectrum calculation the entire assembly is treated as a single spectral region and the average fuel pin isotopics are obtained by averaging over the whole assembly. The spectrum and group constants are calculated by a multigroup Fourier transform technique whose fundamental equations are derived by applying a Greuling-Goertzel slowing down model to a B-1 or P-1 approximation of the one-dimensional Boltzmann equation. The calculation of resonance absorption is based on the total resonance integral for a single pin which is, in turn, based on the Hellstrand experiments. The individual rod resonance absorptions are further adjusted by the Dancoff factor to account for the differing environments of the rods. The assignment of the fraction of the total absorption that occurs in each group is based on resonance distribution functions which have been precalculated and stored in the code epithermal library as a function of rod radius and fuel temperatures.

The effect of burnable poison (gadolinia in BWRs) is calculated by the THERMOS-GADPOL code system. THERMOS is a transport theory code with about 35 thermal groups and is used to calculate the absorption rate in gadolinia bearing rods. The pin cell is modeled in cylindrical geometry and is surrounded by a cylindrical shell which is the homogenized representation of the eight rods surrounding the burnable poison rod. An additional shell representing the water in the bundle-to-bundle gaps may also be used. The GADPOL code models the same core region in x-y geometry and calculates diffusion theory constants for the burnable poison rod which conserve the reaction rates. These constants are then used by RECORD in the assembly homogenization. A similar technique is also used to treat PWR cluster burnable poison rods.

Control rod absorbers are treated in RECORD as non-diffusion sub-regions defined by boundary conditions at the absorber surfaces. The boundary conditions are current-to-flux ratios calculated from transport theory approximations. Both the traditional rodded BWR control blades and the newer solid blades may be treated as well as PWR clusters. The current-to-flux ratios are used as boundary conditions on the control rod surfaces in the x-y assembly calculation.

Burnup calculations are performed in a straight-forward manner by integrating the differential equations describing the buildup and decay of fission product and actinide concentrations. RECORD uses a fission product model consisting of 12 isotopes in six chains and four pseudo fission products. This model was chosen because it gives good agreement with more exact models. Fuel burnup chains beginning with U-235 and U-238 are used.

RECORD is used to calculate TIP instrumentation factors - the relation between TIP signal and the power in the four surrounding fuel assemblies. These factors are used in the core simulator code to obtain TIP trace values for comparison to measured values.

Verification of the RECORD calculations of criticality and power distributions in critical experiments and operating power reactors is presented in the report along with comparisons of actinide concentrations as a function of burnup to measurements made at Yankee-Rowe. Comparisons were made to cold clean UO_2 criticals performed by Westinghouse, Babcock and Wilcox and in Scandanavia. The effective multiplication factors calculated by RECORD were without significant bias and had a standard deviation of 0.0042. The comparison to mixed oxide criticals performed by Battelle Pacific Northwest Laboratories showed essentially the same standard deviation but had a positive bias (i.e., overpredicted the effective multiplication factor) of 0.0071.

The comparisons of isotopic concentration as a function of burnup showed good agreement between calculation and measurement out to about 25 gigawatt days per ton of uranium. Local power distribution calculations were compared to experimental data from the 1976 Quad Cities Unit 1 gamma scan measurements. The total RMS difference between the RECORD calculations and the measurements was about the same as the measurement uncertainty. No significant trends were noted in the comparisons.

3. Summary of Evaluation

The calculation procedure used in RECORD - that of proceeding from essentially infinite group cross-sections to a two-group representation and from a single pin cell to a full assembly homogenization - is standard industry practice for lattice physics codes and is acceptable.

The point energy approach used in the calculation of the thermal neutron spectrum is a departure from the usual industry practice. However, in response to staff questions, CP&L has presented an analysis which demonstrates that this approach is essentially equivalent to that of the widely used THERMOS code. They also presented comparisons with reference THERMOS results (spectra and reaction rates) obtained in a 54-group calculation which showed essentially identical results. We conclude that the use of the point energy approach is acceptable.

The epithermal spectrum is calculated by methods which are widely used in the industry and are acceptable. The treatment of burnable poison is a standard method of obtaining the diffusion theory equivalent of a transport theory calculation and is acceptable. The treatment of control rod absorbers is also similar to industry practice and is acceptable.

The burnup calculation procedures are straight-forward and acceptable. Likewise the calculation of the TIP factors and LPRM factors are straightforward and acceptable.

The comparisons with experimental data are sufficient to permit the conclusion that the RECORD code is capable of performing its function of preparing input for the nodal code. The use by CP&L of the RECORD code to perform core analysis for boiling water reactors will be addressed in a separate evaluation.

4. Evaluation Procedure

The review of report NF-1583.02 has been conducted within the guidelines provided for analytic methods in the Standard Review Plan, Section 4.3. Sufficient information is provided to permit the conclusion that the calculational model described in the report is state-of-the-art and is acceptable. The uncertainties to be ascribed to the various calculated quantities are discussed in a separate evaluation.

5. Regulatory Position

Based on our review, which is described above, we conclude that report NF-1583.02 is suitable for reference by Carolina Power and Light in licensing actions concerning their boiling water reactors. Such reference may be made for purposes of describing the analysis methods used. The validation of the use of this code for analyses by CP&L is the subject of a separate evaluation.

We encourage the continued monitoring of the performance of this code by CP&L and of upgrading as required. The staff wishes to be kept informed of significant developments.

ENCLOSURE 2

Evaluation of Report NF-1583.03

1. Introduction

Carolina Power and Light, in anticipation of the performance of core reload analyses for their boiling water reactors, has presented three reports for review by the NRC. These reports are:

NF-1583.01, "A Description and Validation of Steady-State Analysis Methods for Boiling Water Reactors"

NF-1583.02, "Methods of RECORD: An LWR Fuel Assembly Burnup Code"

NF-1583.03, "Methods of PRESTO-B: A Three-Dimensional BWR Core Simulation Code"

The Core Performance Branch has reviewed these reports. We had the assistance in the review of our consultant, Brookhaven National Laboratory under Technical Assistance Contract FIN A-3407.

The present evaluation is for report NF-1583.03. The other reports are the subject of separate evaluations. The second section of the evaluation summarizes the report contents and the third presents a summary of our evaluation. A statement of the evaluation procedure and the conclusions (regulatory position) of our evaluation follows.

2. Description of Report

Report NF-1583.03 describes the PRESTO-B code, which is a three-dimensional BWR core simulator with coupled neutronics and thermal-hydraulics models. Also included are data to support a general verification of the code. Verification of its use by CP&L is discussed in a separate evaluation.

PRESTO-B uses the Borreson formulation of the coarse mesh approximation to diffusion theory which has been used by others (e.g., in the SIMULATE code used by Yankee Atomic). The thermal hydraulics formulation is the steady-state version of that used in the RAMONA code.

After an introduction and summary the report describes the core model used in the calculations (Section 3) and discusses the representation of the nuclear data (Section 4) and the neutron diffusion model (Section 5). The thermal-hydraulics model discussed in Section 6 and Section 7 contains a description of the power distribution and burnup calculations. Section 8 describes the models and procedures used to obtain core performance parameters (TIP and LPRM responses, thermal limits, etc.). The xenon dynamics model described in Section 9 and Section 10 contains a discussion of various auxiliary functions (e.g., control rod pattern search) performed by the code. Finally Section 11 presents the results of the general code qualification that has been performed. Each section is discussed below.

The core is divided into essentially cubical volumes called nodes. Each node then represents a cube with six-inch sides. Each fuel assembly in the core is assigned a unique identifier. The data on the input files (prepared by the RECORD code) are listed with the same identifier. A similar procedure is used to identify control rods since PRESTO-B has the capability to monitor control rod depletion.

Neutron cross-section data are supplied in assembly averaged two-group format and are the output of RECORD or other lattice physics codes. These data are then processed by an auxiliary code to generate polynomial fits in exposure and exposure-weighted void fraction for each cross-section. These fits are used in PRESTO-B along with additional algorithms to account for instantaneous voids, power level, xenon and samarium concentration, etc. to obtain the cross-sections for each node. Fits are generated for both controlled and uncontrolled assemblies. The models used to account for each effect are described in detail.

The derivation of the algorithms for the calculation of the two-group flux distribution and core eigenvalue (effective multiplication factor) is presented. The Borresen formulation of the coarse-mesh diffusion equations is used. A central mesh point finite difference formulation is used for the fast flux. The thermal flux may be either obtained from the converged fast flux or it may be iterated in the same manner as the fast flux is calculated. The various models, algorithms and procedures used for the thermal hydraulic calculations are described in detail.

The relative power in each node is calculated in a straight-forward manner by forming the product of the fission cross-section and flux for each group and summing. The power is normalized to an average value of unity for the core. The nodal average linear heat generation rate (APLHGR) is calculated in a straight-forward way and the nodal maximum linear heat generation rate is obtained by multiplying the APLHGR by the nodal pin peaking factor obtained from the RECORD calculation.

If required, a burnup step calculation may follow the power distribution calculation. The nodal fuel exposure and exposure weighted void distribution are integrated through the step along with the Sm-149 and Pm-149 concentrations and the concentration of up to two fission product isotopes used for gamma scanning. The nodal power may be assumed to be constant throughout the step at its initial value or, if a new power distribution is calculated at the end of the step, a weighted average of the initial and final nodal powers may be used. Also, a cycle burnup (Haling) calculation may be performed. The procedures and algorithms used to perform each of these functions are described in the report.

TIP and LPRM readings may be calculated by PRESTO-B using instrument factors for the four surrounding assemblies which are calculated by RECORD. If measured values for these quantities are provided these may be used to obtain the uncertainty in the radial power distribution. The measured and calculated total areas under the curves for all measured locations are normalized to each other after which the areas under individual curves are compared.

The margins to thermal limits - critical heat flux ratio, fraction of limiting power density and average nodal linear heat rate (APLHGR) may also be calculated by PRESTO-B.

The xenon transients which occur during certain plant operations (e.g., control rod pattern exchange maneuvers or load following) may be simulated by PRESTO-B. Analytic solutions of the nodal xenon and iodine concentrations are obtained at user specified time points during the transient. The power level or flow rate required to maintain criticality at all points in the transient may also be calculated along with the maximum rate of change of nodal power for nodes above a specified power. The latter quantity is used for comparison with pellet-clad interaction (PCI) limits.

Certain features of the code such as critical control rod pattern search, maximum stuck rod worth search, reload fuel shuffling and fuel discharge priority listing are briefly described but no detailed description of procedures is given.

An extensive comparison of the results of PRESTO-B calculations against a benchmark problem and measurements is presented to provide a general qualification of the code for BWR calculations. Comparisons of nodal power and eigenvalue were made to a fine mesh diffusion theory calculation of a benchmark problem (IAEA 3D Benchmark). Nodal powers were predicted to within 1.6 percent for the simplified thermal flux model mentioned above and to within 1.3 percent for the more detailed thermal model. Eigenvalues were predicted to within 40 pcm and 30 pcm respectively.

The PRESTO-B thermal-hydraulic model was verified by comparison with the FRIGG void loop experimental data. The overall standard deviation of the difference between calculated and measured void was 2.2 percent void.

Comparisons were made to the results of gamma scans of the Hatch Unit 1 reactor. Reactor operation through the first cycle was simulated with PRESTO-B and the end of cycle La-140 distribution was calculated for comparison with the corresponding measured distributions. A total of approximately 100 nodes were compared to obtain a total standard deviation of 6.4%. A total of 72 comparisons of axially integrated bundle activities were made to obtain a standard deviation of about 2 percent. Essentially no difference was found between rodded and unrodded nodes in the comparisons.

Comparisons of PRESTO-B calculations with measurements that have been done at the Brunswick plant are presented in an additional report and will be the subject of a separate evaluation.

3. Summary of Evaluation

The following discussion summarizes our evaluation of Report NF-1583.03.

The nodal core modeling used is standard for BWR reactors and is acceptable. The manner in which the input cross-section data are prepared for the code (two-group representation) is typical of more recent industry practice and is acceptable. The algorithms used by the code to obtain the values to be used for each node (polynomial fits) is standard industry practice and is acceptable.

The Borresen formulation of the coarse mesh diffusion theory has been used in other recent nodal codes and been shown to produce adequate results. The treatment of boundary nodes by use of albedos is common industry practice and is acceptable.

A detailed fuel performance code is proposed for use to provide average fuel temperatures to PRESTO. However, the COMETHE code has not been reviewed by the staff* and, therefore, does not constitute an acceptable source of fuel temperature data in plant safety analysis.

The use of a detailed fuel performence code (such as COMETHE) is required in several areas of the safety analysis. The applications range from establishing the Doppler coefficient contribution to the overall power defect (as proposed by CP&L) to providing the initial fuel conditions for the loss-of-coolant accident (LOCA) analysis. Because the LOCA analysis is used to determine reactor operating limits, the fuel performance calculations are very important and subject to stringent review. For steady-state BWR analysis, on the other hand, the Doppler coefficient contribution is not critical (although a large error in fuel temperature or Doppler may result in unexpected rod position requirements and a Technical Specification violation in predicting reactivity). For this reason, we agree with the licensee's position, stated in response to staff questions, that a review of COMETHE is not warranted for this application.

For non-LOCA BWR transient analysis, the moderator feedback (which controls most events) is strongly dependent on fuel temperature and gap conductance. The fuel performance analysis is, therefore, more

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* A proposed application of COMETHE for BWR transient analysis has been received by the staff (Ref. 2). This application is currently under review.

critical and some level of technical review is justified (including compliance with Generic Letter 83-11, Ref. 1). CP&L has stated that the subject of fuel performance in BWR transient modeling will be addressed in future submittals. Since the use of COMETHE in BWR transient applications is not being sought at this time, we find this to be acceptable.

The thermal-hydraulics model is the steady-state version of that used in the RAMONA code which has been used for transient calculations. The version used in the PRESTO-B code has been verified by Scandpower against test data and shown to produce acceptably accurate results. The licensee has performed further verification against BWR operating data. We conclude that the thermal-hydraulics model used in PRESTO-B is acceptable.

The calculations of relative nodal power, nodal average linear heat generation rate and maximum linear heat generation rate are performed in a straight-forward manner and are acceptable. The core burnup calculation method is consistent with standard industry practice and is acceptable.

The algorithms used for the calculation of core xenon transients are derived from analytic solutions to the differential equations which describe xenon and iodine behavior during the transient and are acceptable.

The comparisons shown between measured and calculated power distributions, eigenvalues, and void fraction data are sufficient to permit the conclusion that the PRESTO-B code is suitable for the calculation of core characteristics of boiling water reactors in steady-state and xenon transient conditions.

The core flow and pressure drop comparisons provide verification that the thermal hydraulic model in PRESTO is capable of acceptable accuracy in calculating these quantities. The xenon transient and loss of feedwater heater transient comparisons provide verification that slow transients (slow enough to permit the neutron flux and heat flux to remain in phase) may be acceptably calculated by PRESTO. In particular the bias in the eigenvalue was essentially the same as that for steady-state cases and the standard deviation in the LPRM readings was similar to that for the steady-state TIP comparisons.

4. Evaluation Procedure

The review of report NF-1583.01 has been conducted within the guidelines provided for analytic methods in the Standard Review Plan, Section 4.3. Sufficient information is provided to permit the conclusion that the data base used for the verification is sufficient, that the analyses of the results is proper and that the various biases and uncertainties are typical for the methods used and are acceptable.

5. Regulatory Position

Based on our review, which is described above, we conclude that report NF-1583.01 may be used as a reference for the verification by CP&L of the RECORD-PRESTO code system. Such reference may be made in licensing actions for which analyses are performed by CP&L. We further conclude that use of this code system in the areas listed in Section 2 of this evaluation is acceptable.

We encourage the effort at CP&L to monitor the performance of this code system and made improvements as required. The staff wishes to be informed of significant developments.

REFERENCES

1. D. G. Eisenhut (NRC), Generic Letter No. 83-11 to All Operating Reactor Licensees on "License Qualification for Performing Safety Analyses in Support of Licensing Actions" dated September 26, 1983.

2. J. K. Lee and J. F. Burrow, "Validation of COMETHE III-J for Gap Conductance Calculations," Tennessee Valley Authority Report EAS-138 (October 25, 1983). Transmitted by L. M. Mills (TVA) letter to H. R. Denton (NRC) dated November 21, 1983.

ENCLOSURE 3

Evaluation of Report NF-1583.01

1. Introduction

Carolina Power and Light, in anticipation of the performance of core reload analyses for their boiling water reactors, have presented three reports for review by the NRC. These reports are:

NF-1583.01, "A Description and Validation of Steady-State Analysis Methods for Boiling Water Reactors"

NF-1583.02, "Methods of RECORD: An LWR Fuel Assembly Burnup Code"

NF-1583.03, "Methods of PRESTO-B: A Three-Dimensional BWR Core Simulation Code"

The Core Performance Branch has reviewed these reports. We had the assistance in the review of our consultant, Brookhaven National Laboratory, under Technical Assistance Contract FIN A-3407.

The present evaluation is for report NF-1583.01. The other reports are the subject of separate evaluations. The second section of the evaluation summarizes the report contents and the third presents a summary of our evaluation. A statement of the evaluation procedure and the conclusions (regulatory position) of our evaluation follows.

2. Description of Report

Report NF-1583.01 presents brief descriptions of the codes RECORD and PRESTO-B which have been described in detail in the companion reports NF-1583.02 and NF-1583.03, respectively. The major portion of NF-1583.01 describes the extensive qualification of the code combination that has been performed by Carolina Power and Light (CP&L).

CP&L intends to use the code system to perform steady-state BWR analysis including the effect of xenon transients. The specific areas of proposed use are listed and include:

1. Integral core simulation for normal operation and certain transients
2. Fuel management calculations
3. Calculation of reactivities and rod worths for nominal and transient conditions
4. Data base for generation of core effective reactor physics data
5. Evaluation of thermal limits from normal and certain transient conditions.

Section 2 of the report NF-1583.01 presents a brief description of the RECORD report. A detailed description is given in NF-1583.02 along with code verification performed by Scandpower, the developers of the code. That report is the subject of a separate evaluation. The RECORD code calculates the infinite lattice multiplication factor, pin relative power distributions, isotopic densities and two group diffusion parameters as a function of exposure to use in nodal simulator codes. Weighted delayed neutron parameters are also calculated for use in kinetic calculations.

To verify the use of RECORD, CP&L has augmented the results given in NF-1583.02 with comparisons to higher order calculations, isotopic ratios measured in both a BWR and a PWR, and pin power distributions measured in Quad Cities Unit 1. The results of these comparisons are given in Section 3 of the subject report. The following paragraphs summarize the results of the comparisons.

Comparisons of RECORD-calculated assembly reactivity as a function of burnup were made to calculations performed by the CPM (Collision Probability Method) calculations. This code is part of the EPRI ARMP package and is used in the industry as a standard of reference.

Comparisons with the RECORD calculations show generally good agreement with a tendency by RECORD to underpredict k-infinity at larger burnups-particularly at low void concentrations. RECORD also burns the gadolinium at a faster rate than CPM so that it overpredicts the k-infinity early in the assembly lifetime.

Isotopics comparisons were made to measurements performed on three pin samples which had been exposed in Cycles 1 and 2 of H. B. Robinson Unit 2 (PWR) and on eight pins from a mixed oxide assembly exposed in Cycle 2 of Quad Cities Unit 1 (BWR). Exposures ranged from 7 to 30 GWd/MTM. The agreement between calculation and measurement was similar for both the PWR and BWR cases and was quite good except for the higher isotopes. In particular, the production of plutonium-240 was underpredicted by more than ten percent. In the BWR comparisons the agreement with experiment was not a function of the void history of the exposure.

Comparisons were made between RECORD calculations and measurements of the La-140 concentrations in four assemblies that had been exposed in Quad Cities Unit 1. Comparisons were made at four axial elevations and exposures ranged from 7.5 to 19.4 GWd/MTM. 7x7, 8x8 and a mixed oxide assembly were represented. The average standard deviation in the results was 2.7 percent and the assembly local peaking factors were predicted to within 1.6 percent for the UO_2 assemblies. Including the mixed oxide assembly the local peaking factor was overpredicted by 1.7 percent with a standard deviation of 4.1 percent. The standard deviation for gadolinium bearing pins was 2.8 percent - not significantly different from the total.

Section 4 of the report contains a brief description of the PRESTO-B code - a three-dimensional core simulator code for boiling water reactors. A detailed description has been presented in NF-1583.03 which is the subject of a separate evaluation. That report presents a performance verification based on work done by Scandpower using data from European sources and some U. S. data. Section 5 of NF-1583.01 presents data to provide supplementary verification and to demonstrate the ability of CP&L to use the code for analyses of their reactors.

The following data are presented:

A. Comparisons between PRESTO and Fine Mesh Diffusion Theory (PDQ07)

   1. Cold critical eigenvalues for

      a. All rods in
      b. Single rod out
      c. 50 percent rod density
      d. 25 percent rod density

   2. Core power distributions for the above set

   3. Cold critical eigenvalues for cores containing water holes (reshuffling configurations)

   4. Hot critical eigenvalues and power distributions at three axial elevations

B. Comparison of PRESTO with cold critical statepoint data (in-sequence criticals) for Brunswick 1 and 2 and Quad Cities Unit 1.

C. Comparison of PRESTO and Quad Cities Unit 1 "clumped" cold criticals (shutdown margin test)

D. Comparison of PRESTO with hot critical statepoints at various times in several cycles of Brunswick 1 and 2 and Quad Cities Unit 1.

E. Comparison of PRESTO calculated hot power distributions with process computer (TIP) distributions for several cycles of Brunswick 1 and 2 and Quad Cities Unit 1.

F. Comparison of PRESTO calculations to the Quad Cities Cycle 1 and 2 gamma scan measurements.

G. Comparison of PRESTO Calculations of core flow distribution and pressure drop to process computer values.

H. Simulation of a xenon transient in Brunswick 2

I. Simulation of a Brunswick 1 loss of feedwater heater startup test.

The results of the comparisons and simulations are given in the form of tables and figures and are summarized in Section 6 of the report. For the cold critical eigenvalues PRESTO and PDQ07 agreed to within less than 0.001 for the all rods in the configuration and to within about 0.003 for the other configurations including the reshuffling configurations.

The power distribution comparisons for these configurations showed that PRESTO tends to overestimate power peaking factors by about ten percent. The hot eigenvalues were predicted to within about 0.3 percent and hot power peaking to about 2.5 percent.

Comparisons between PRESTO and cold critical statepoint data show that the in-sequence eigenvalue is underestimated by 0.005 with a standard deviation of 0.002. For shutdown margin test configurations the standard deviation is about 0.003. The uncertainties (95/95) to be applied to calculations for the eignevalue are 0.0048 and 0.0064 respectively. The bias (0.005) is the same for both calculations. For hot critical statepoints the bias and uncertainty are essentially the same as for the in-sequence cold criticals.

The comparison of PRESTO calculations with TIP scans from Brunswick 1, Cycles 1, 2, 3, Brunswick 2, Cycle 4, and Quad Cities Unit 1 Cycles 1 and 2 show that the uncertainty (standard deviation) in the difference between measurement and calculation is 8.9 percent for nodal power, 5.1 percent for assembly power and 4.4 percent for peak nodal power. If the measurement uncertainty is removed the results are 4.5, 3.2, and 3.5 percent respectively. Comparisons of PRESTO with assembly gamma scan data from Quad Cities Unit 2 yielded standard deviations of 3.77 percent for nodal power, 2.52 percent for assembly power and 1.49 percent for peak power. Comparison with pin gamma scan data yielded 2.6 and 2.8 percent standard deviation for nodal power and peak power respectively. The combined measurement uncertainty for these data is about 3.0 percent.

The comparisons between PRESTO calculated core flow and pressure drop and process computer values for these quantities show standard deviations of 3.5 and 3.0 percent respectively.

A control rod sequence exchange xenon transient in Brunswick Unit 2 was simulated by PRESTO. Detailed measurements of LPRM readings were made during the exchange to provide a basis for comparison. The LPRM readings were forced to agree at the initial state (by applying correction factors to the PRESTO values which were then held constant). The eigenvalues obtained for the various statepoints analyzed showed an average value of 0.996 with the standard deviation of 0.001. The maximum value of the RMS difference between calculated and measured LPRM values (determined at each statepoint) was 3.5 percent of rated power.

The Brunswick 1 loss of feedwater heater transient, performed as a part of the startup testing was modeled by PRESTO in a fashion similar to that used to model the xenon transient. The LPRM responses and peak reactor power were determined. The ratio of peak to initial power was determined to within 0.6 percent. The peak discrepancy between measured and calculated LPRM readings at peak power was less than 10 percent. The average discrepancy was of the order of one percent.

3. Summary of Evaluation

We have reviewed the experimental data used for the verification of the RECORD - PRESTO code set. The following comments are relevant.

1. The data base used to verify the RECORD code (as a "stand-alone" code) is similar to that customarily used and is acceptable. The results of the comparison show uncertainties that are typical of those usually found and are acceptable.

2. Comparison of nodal calculations with higher order (e.g., fine-mesh diffusion theory) calculations is a standard technique for assessing their performance and is acceptable. The results for PRESTO are typical of the results of such comparisons and are acceptable.

3. Verification of power distribution calculations by comparison with TIP measurements in operating reactors is a standard technique and the experiment set used by CP&L for PRESTO verification is adequate. The results of the comparisons are typical and are acceptable.

4. The Quad Cities gamma scan data are part of a body of such data obtained for the purpose of verifying calculation techniques and is a sufficient data base when combined with other data (e.g., TIP data). The results of the PRESTO comparisons show standard deviations which are less than twice the measurement uncertainty. This is acceptable agreement.

4. Evaluation Procedure

The review of report NF-1583.03 has been conducted within the guidelines provided for analytic methods in the Standard Review Plan, Section 4.3. Sufficient information is provided to permit the conclusion that the calculational model described in this report is state-of-the-art and is acceptable. The uncertainties to be ascribed to the various calculated quantities are discussed in a separate evaluation of the CP&L topical report NF-1583.01.

5. Regulatory Position

Based on our review of report NF-1583.03 we conclude that it is acceptable for reference in boiling water reactor licensing actions by Carolina Power and Light Company. Such reference may be made for the purpose of describing the methods of analysis of steady-state core physics parameters and for analysis of xenon transients.

We encourage the continuing effort at Carolina Power and Light to monitor the performance of this code and to make further refinement as required. The staff wishes to be informed of any significant developments.