NF-1583.04-1

VERIFICATION OF CP&L REFERENCE BWR THERMAL-HYDRAULIC METHODS USING THE FIBWR CODE
AMENDMENT 1

RESPONSE TO NRC QUESTIONS

MARCH 1984

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INTRODUCTION

This amendment to Topical Report NF-1583.04, "Verification of CP&L Reference BWR Thermal-Hydraulic Methods Using the FIBWR Code," is provided in response to requests for additional information as conveyed by the letter to Mr. E. E. Utley, CP&L, from Mr. Domenic B. Vasallo, NRC Division of Licensing, dated January 17, 1984.

Question 1

Please identify any changes or modifications over the Electric Power Research Institute version of the FIBWR code. For these changes, if any, the modified models (e.g., conservation equations, void fraction, and subcooled boiling models, etc.) should be described.

Response

There have been two modifications at CP&L to the EPRI version of the FIBWR code. Both of these modifications involved the addition of capabilities without altering any of the existing models or programming. The first change consisted of the addition of the GEXL correlation as a new option for the critical power/heat flux calculations. The correlation was inserted in the form of a separate subroutine and uses a call statement reserved for a user-supplied correlation. The verification of this change is included as part of the CP&L Topical.

The second modification was made to facilitate the FIBWR interaction with the nodal simulator, PRESTO-B. Equations for determining the effective inlet and exit form loss coefficients for PRESTO are described in Section 4.3.2 of the Carolina Power & Light Company Topical, "A Description and Validation of Steady-State Analysis Methods for Boiling Water Reactors" (CP&L File NF-1583.01, February 1983), currently under review by the Nuclear Regulatory Commission. Values for each of the variables in these equations are either input to or calculated by FIBWR and are, therefore, stored in the FIBWR common arrays at the end of the execution. This modification involved programming Equations 4.3.2 and 4.3.3 of NF-1583.01 into FIBWR so that just prior to the end of execution, the PRESTO loss coefficients would be calculated and printed on the summary page of the FIBWR output.

Question 2

Define the "slow transients" in Item 2 on Page 4. Specify which transients will be analyzed by the FIBWR code for the hot bundle analysis.

Response

The steady-state FIBWR code can be used to analyze the type of transients in which the heat deposition rate in the coolant due to conduction from the clad surface to the moderator can be conservatively represented by the heat generation rate of the fuel. Examples of enthalpy deletion events during which the average core enthalpy changes slowly are the inadvertent HPCI or RCIC pump start and loss of feedwater heater transients.

Question 3

Describe how the FIBWR code is coupled to the neutronics code for power distribution calculations (e.g., the iteration between the neutronic and thermal-hydraulic calculations and the differences in input to FIBWR between the neutronic-type calculations and the CPR-type calculations). Also describe the coupling between the FIBWR code and the plant process computer as mentioned in Item 4 on Page 2.

Response

The FIBWR code calculates the core flow distribution in order to define input for the neutronics code. This is done by either defining a leakage flow/total flow curve from FIBWR to be input into PRESTO or adjusting loss coefficients input to PRESTO to match FIBWR bypass calculations for specific operating conditions. FIBWR flow distributions are also used to calculate effective inlet and exit loss coefficients for PRESTO by weighting the orifice, tie plate, and grid strap losses by the relative flow through each component. A description of the neutronics code coupling with FIBWR is given in Section 4.3.2 of the Carolina Power & Light Company Topical NF-1583.01 specified in Question 1. There are no changes to the FIBWR input when making bypass flow calculations from the input used in the CPR-type calculations.

Future applications may involve calculating input constants for the plant process computer analogous to the effective inlet loss coefficients used by PRESTO. These constants are currently determined by the vendor's hydraulic methods and are supplied for each Brunswick unit at the beginning of the operating cycle. Should CP&L decide to use FIBWR for process computer applications, further documentation will be submitted to the NRC at that time.

Question 5

Explain the applicability of the spacers and lower and upper tie plate form loss coefficients from Reference 3, Table 5.1 (i.e., EPRI-NP-1923) to the Brunswick core thermal-hydraulic analysis (cf. Table 2 of the CP&L Topical).

Response

The EPRI-NP-1923 loss coefficients in Table 2 of the CP&L Topical are all geometry-dependent only. The lower tie plate, upper tie plate, and spacer loss coefficients in EPRI-NP-1923 were derived for fuel assemblies with geometries identical to those at Brunswick and, therefore, should be applicable for Brunswick core thermal-hydraulic analysis. Equal in importance is the fact that the CP&L FIBWR model utilizing these loss coefficients has demonstrated the ability to match Brunswick hydraulic conditions.

Question 6

Explain how the water rods are modeled hydraulically. Are they different from the active flow channels?

Response

The water rods are represented as a separate flow path parallel to the active flow channel. Section 4.7 of EPRI-NP-1924-CCM gives the following description of the water tubes: "FIBWR calculates the water tube flow consistent with the pressure drop of the active coolant parallel to the tube. The entrance and exit elevations are input with reference to the start of the heated region of each channel. The water tube is assumed to start at or above the bottom of the reference length, but may extend up into the upper unheated region. The FIBWR code models the water tube pressure drop in a similar manner to the active coolant, with the exception that the homogeneous void relationship and two-phase local loss multiplier models are used should the water tube experience bulk boiling. No friction loss multiplier or two-phase corrections to the acceleration pressure drop are included; if the flow rate is low enough to allow the water tube to boil, the pressure loss components will not be significant."

Question 7

Only the form loss coefficients for water rod entrances are given in Table 2. Explain why the exit form loss coefficients are not given.

Response

The water rod entrance loss coefficients were empirically derived from the flow data provided by the GE document specified in NRC Question 8. The exit loss coefficients used are those given in EPRI-NP-1923. The method used to develop the FIBWR input for the water rod form loss coefficients is given below:

1. A single-bundle model representing an 8x8 retrofit assembly was established in order to set the power and active flow equal to the nominal conditions specified in the GE document. Initial form loss coefficients input were taken from EPRI-NP-1923.

2. Several FIBWR cases were then run, varying the water rod entrance loss coefficient until the water tube flow matched the value given in the GE document. The entrance loss coefficient was selected over the exit coefficient because flow into the water tube is more restricted than flow at the exit; therefore, most of the water tube pressure drop occurs at the bottom.

3. Loss coefficients derived in Step 2 were verified by running an off-nominal case and confirming the agreement between the FIBWR-calculated water tube flow rate and the GE data for the off-nominal condition.

4. Finally, an approximation was needed for the water tube entrance form loss coefficient of an 8x8 standard fuel assembly. This was done by calculating the ratio K_{8x8}/K_{8x8R} based on the respective water tube geometries, where K is the parallel average of the loss through the water tube entrance holes given by the equation:

$$K = \frac{A_{REF}^{2}}{\left[\sum_{i=1}^{n}\left\{\frac{A_i^2}{K_i}\right\}^{\frac{1}{2}}\right]^{2}}$$

and

A_{REF} = water tube flow area,
n = number of water tube entrance holes,
A_i = area of water tube entrance holes, and
K_i = square-edged water hole loss coefficient (1.5 was used for all holes).

Values for A_{REF}, n, and A_i were obtained from NEDE-24011-P-A-6 (GESTAR II), Section 2. The empirically-derived value for K_{8x8R} from Step 2 was multiplied by the geometry ratio defined above to arrive at an approximation for the standard 8x8 water tube entrance loss coefficient, which corresponds with the empirical method used for the 8x8 retrofit design.

Question 9

Provide the detailed derivation of the leakage coefficients (C1,C2,C3, and C4) for bypass flow paths in Table 4. Justify the parameters needed in the derivations, such as pressure differentials and flow fractions of different paths. If General Electric Company information is used, provide those documents.

Response

The leakage coefficients C1,C2,C3, and C4 used by FIBWR to determine the bypass flow paths were calculated by an iterative method. In the first iteration, the process described in section 5.1.4 of EPRI-NP-1923 was applied to bypass flow fractions from open literature and initial guesses at pressure differentials to calculate an initial set of bypass coefficients. These coefficients were used in FIBWR to generate a new set of pressure drops for the core support plate and channel dependent leakage paths.

During the second iteration, the pressure differentials generated by FIBWR in iteration #1 were used with bypass flow fractions supplied by General Electric in the document described in NRC Questions 8 above, to calculate new values for C1,C2,C3, and C4.

Lower Tie Plate Flow Holes (Path 9)

From GE Report NEDC-21215 (Reference 6 of the CP&L Topical):

$$W = 259.6\rho^{\frac{1}{2}}\Delta P^{\frac{1}{2}} \qquad \text{(Eq. 1)}$$

where W = flow through the two lower tie plate bypass holes (lb/hr)

ρ = density of coolant (lb/ft^3)

ΔP = differential pressure across lower tie plate (LTP) holes (psi)

Equation 1 is of the form $W = C1\,\Delta P^{\frac{1}{2}} + C2\Delta P^{C4} + C3\Delta P^{2}$ with

$$C1 = 259.6\rho^{\frac{1}{2}}$$
$$C2 = C3 = C4 = 0$$

Assuming a density of 47.12 lb/ft^3, the value of C1 for the lower tie plate flow holes (path 9) becomes:

$$C1 = 259.6\ (47.12)^{\frac{1}{2}} = 1782.0$$

The remaining leakage coefficients can be determined from C1 for path 9 and their relative bypass flow fractions.

Control Rod Paths (paths 1,2, and 5)

Flow fraction for path 1:	0.297
Flow fraction for path 2:	0.048
Flow fraction for path 5:	0.003
Total control rod flow fraction:	0.348

Flow fraction for path 9:	0.346
Number of control rod paths:	137
Number of LTP paths:	560
ΔP control rod paths:	24.0 psi
ΔP path 9:	10.0 psi

$$(560)\ \frac{0.348}{0.346}\ Cl_9\, \Delta P_9^{\frac{1}{2}} = (137)\ Cl_{cr}\, \Delta P_{cr}^{\frac{1}{2}}$$

$$(560)\ \frac{.348}{.346}\ (1782)\ (10)^{\frac{1}{2}} = (137)\ Cl_{cr}\ (24)^{\frac{1}{2}}$$

$$Cl_{cr} = 4729.0$$

$$C2 = C3 = C4 = 0$$

Instrument Tube Paths (path 3)

Flow fraction for path 3:	0.002
Flow fraction for path 9:	0.346
Number of Instrument tubes:	31
Number of LTP paths:	560
ΔP path 3:	24 psi
ΔP path 9:	10 psi

$$560\ \frac{0.002}{0.346}\ Cl_9\, \Delta P_9^{\frac{1}{2}} = (31)\ Cl_3\, \Delta P_3^{\frac{1}{2}}$$

$$560\ \frac{0.002}{0.346}\ (1782)\ (10)^{\frac{1}{2}} = (31)\ Cl_3\ (24)^{\frac{1}{2}}$$

$$Cl_3 = 120.1$$

$$C2 = C3 = C4 = 0$$

Shroud Path (path 4)

Flow fraction for path 4:	0.001
Flow fraction for path 9:	0.346
Number of shroud paths:	1
Number of LTP paths:	560
ΔP path 4:	24.0 psi
ΔP path 9:	10.0 psi

$$560\ \frac{0.001}{0.346}\ Cl_9\, \Delta P_9^{\frac{1}{2}} = Cl_4\, \Delta P_4^{\frac{1}{2}}$$

$$560\ \frac{0.001}{0.346}\ 1782\ (10)^{\frac{1}{2}} = Cl_4\ (24)^{\frac{1}{2}}$$

$$Cl_4 = 1861.7$$

$$C2 = C3 = C4 = 0$$

Spring Plug Paths (Path 10)

Flow fraction for path 10:	0.005
Flow fraction for path 9:	0.346
Number of spring plugs:	77
Number of LTP paths:	560
ΔP path 10:	24.0 psi
ΔP path 9:	10.0 psi

$$560 \; \frac{0.005}{0.346} \; Cl_9 \, \Delta P_9^{\frac{1}{2}} = 77 \; Cl_{10} \, \Delta P_{10}^{\frac{1}{2}}$$

$$560 \; \frac{0.005}{0.346} \; 1782 \; (10)^{\frac{1}{2}} = 77 \; Cl_{10} \; (24)^{\frac{1}{2}}$$

$$Cl_{10} = 120.9$$

$$C2 = C3 = C4 = 0$$

Fuel Support Paths (Path 6)

Flow fraction for path 6:	0.015
Flow fraction for path 9:	0.346
Number of fuel support paths:	560
Number of LTP paths:	560
ΔP path 6:	10 psi
ΔP path 9:	10 psi

$$560 \; \frac{0.015}{0.346} \; Cl_9 \, \Delta P_9^{\frac{1}{2}} = 560 \; Cl_6 \, \Delta P_6^{\frac{1}{2}}$$

$$(560) \; \frac{0.015}{0.346} \; 1782 \; (10)^{\frac{1}{2}} = (560) \; Cl_6 \; (10)^{\frac{1}{2}}$$

$$Cl_6 = 77.25$$

$$C2 = C3 = C4 = 0$$

Finger Spring Path (Path 8)

The flow fractions provided by GE represent core average end-of-cycle for Brunswick 2, Cycle 5. A more detailed model of the finger spring leakage path was made using the Cycle 5 fuel mix to represent finger spring deformation in fuel assemblies which have been incore longer than one cycle. The leakage coefficients defined in EPRI-NP-1923, section 5.1.4 1 for path 8 were developed from a flow test using fresh fuel. These s ne coefficients are used in the CP&L model to represent new fuel. The core verage finger spring leakage coefficient can be expressed as a weighted mixture of new and used fuel:

$$(A) \; Cl_{AVG} = (B) \; Cl_{new} + (A-B) \; Cl_{used}$$

where $Cl_{new} \; (\Delta P_{new})^{.5} = 702(\Delta P_{new})^{.7106}$

$$Cl_{new} = 702(\Delta P_{new})^{.2106}$$

$$Cl_{new} = 702 \; (9.1)^{.2106}$$

$$Cl_{new} = 1117.7$$

and Cl_{AVG} is defined as follows:

Flow fraction for path 8:	0.280
Flow fraction for path 9:	0.346
Number of finger spring paths:	560
Number of LTP paths:	560
ΔP path 8:	9.5
ΔP path 9:	10.0

$$560 \frac{0.280}{0.346} Cl_9 \Delta P_9^{\frac{1}{2}} = 560\ Cl_8 \Delta P_8^{\frac{1}{2}}$$

$$560 \frac{0.280}{0.346} 1782\ (10)^{\frac{1}{2}} = 560\ Cl_8\ (9.5)^{\frac{1}{2}}$$

$$Cl_{AVG} = 1479.5$$
$$A = 560$$
$$B = 160$$

$$560\ (1479.5) = 160\ (1117.7) + 400\ (Cl_{used})$$
$$Cl_{used} = 1624.2$$

The final interation involved a verification of the parameters used to generate the leakage coefficients against output from a FIBWR execution. Agreement between pressure drop values shown below eliminated the need for an additional iteration.

ΔP Location	Iteration #1	Iteration #2
Path 6	10.0	9.81
Path 9	10.0	9.81
Path 8 (avg)	9.5	9.36
Path 8 (new fuel)	9.1	9.04
Core Support Plate	24.0	23.93

With the exception of the control rod dependent path, flow fractions calculated from the FIBWR output compared favorably with the GE values. The leakage coefficient for this one path was adjusted downward to produce the results shown in Table 3 of the CP&L Topical.

The leakage coefficients C1,C2,C3, and C4 define the bypass flow as a function of pressure drop for given fuel and lower internal geometry. The bypass flow fractions provided by General Electric represent core average values, and variations for specific bundles are expected. However, we have confidence that the constants calculated with these flow fractions can be used to evaluate bypass flows from pressure drops typical of Unit 2 operating conditions. Because of similar physical characteristics and pressure drop ranges, we also believe that these constants are applicable to earlier cycles of Unit 2 as well as to Brunswick Unit 1.

Question 10

Provide the values of A and B in Equation 4 used for the Brunswick analysis. Also provide the values of ELDE and ELDG, which are inputs to the FIBWR code, as adjustment factors for the upper tie plate and grid spacers, respectively. (See Pages 6 through 15 of EPRI-NP-1924-CCM.) Explain why the modified homogeneous model for the two-phase form loss multiplier (Equation 4-24 of EPRI-NP-1924-CCM versus Equation 5 of the CP&L Topical) is not used.

Response

The values of A and B used in Equation 4 are those given on Page US.B-103 of the General Electric Company Report NEDE-24011-P-A-6-US (GESTAR II). The spacer and upper tie plate two-phase multiplier adjustment factors (ELDG, ELDE, respectively) used in the Brunswick analysis were input as:

$$ELDG = 1.0$$
$$ELDE = 1.0$$

The selected option for the two-phase form loss multiplier was the modified homogeneous model having the form:

$$\phi^2_{\text{2-Phase Local}} = 1 + (E)\,(X)\,(\rho_1/\rho_g - 1)$$

(Equation 4-24 of EPRI-NP-1924-CCM)

where

E = Adjustment Factors (ELDG, ELDE)
X = Equilibrium Flow Quality
ρ_1, ρ_g = Saturated Liquid and Saturated Vapor Densities, Respectively

However, since the value of E was selected as 1.0, this equation reduces to:

$$\phi^2_{\text{2-Phase Local}} = 1 + (X)\,(\rho_1/\rho_g - 1)$$

(Equation 5 of the CP&L Topical)

which is the homogeneous two-phase form loss multiplier.

The use of a 1.0 adjustment factor is consistent with the Yankee Atomic methodology demonstrated in EPRI-1924-CCM and is recommended for design purposes by R. T. Layhey and F. J. Moody in Reference 8 of the CP&L Topical.

Question 11

In benchmarking the core pressure drop and the flow distributions, the plant process computer outputs were used. As explained in the Topical (Page 14), the process computer models use the same iterative calculational techniques as the FIBWR code, but with different and sometimes less detailed models. In light of this, explain the usefulness of these benchmark comparisons.

Response

The plant process computer pressure drop model uses cycle-specific input coefficients determined by the vendor's detailed hydraulic codes to calculate core pressure drop. Although the process computer itself may use less detailed component representations than FIBWR, it is designed to produce the same pressure drop/flow characteristics as its more complicated parent code. The value of benchmarking to the plant process computer is the extension of the FIBWR validation by a comparison of FIBWR-core pressure drops to pressure drops indirectly calculated by the vendor's detailed hydraulic models. These comparisons are also useful in demonstrating the ability of the FIBWR code to effectively reflect changes in the core loading pattern from one cycle to another.

Question 12

Discuss if there were any corrections to the FIBWR calculated pressure drop across the core support plate to match the exact locations where the pressure tap measurements were made (cf. Table 6). If the locations for calculated and measured ΔP's are different, the errors could be large in low-flow situations where the static head becomes more important than that in high-flow situations.

Response

The measured core support plate pressure drop (ΔP_m) is determined from pressure taps located in the Standby Liquid Control System illustrated in FIGURE 12a.

Using the notation in FIGURE 12a,

$$\Delta P_m = P_5 - P_4 \qquad \text{(Eq. 1)}$$

and, assuming constant density,

$$P_5 = P_3 + \rho l \qquad \text{(Eq. 2)}$$
$$P_4 = P_2 + \rho (l + h) \qquad \text{(Eq. 3)}$$

where:
- ρ = density of coolant
- h = elevation of tap 2 above tap 3
- l = elevation of tap 3 above pressure transmitter

As coolant flows from point 3 to point 2 through the core support plate, both static and dynamic pressure losses occur. Therefore, the pressure at point 3 can be expressed as:

$$P_3 = P_2 + \rho h + \Delta P_{flow} \qquad \text{(Eq. 4)}$$

where:
- ρh = static losses between points 3 and 2
- ΔP_{flow} = pressure drop associated with core flow (friction, local, acceleration)

substituting Eq. 4 into Eq.2

$$P_5 = P_2 + \rho h + \rho l + \Delta P_{flow} \qquad \text{(Eq. 5)}$$

ΔP_m can be expressed in terms of its pressure drop components by substituting Eq. 5 and Eq. 3 into Eq. 1:

$$\Delta P_m = P_2 + \rho h + \rho l + \Delta P_{flow} - P_2 - \rho (h+l)$$
$$\Delta P_m = \Delta P_{flow} \qquad \text{(Eq. 6)}$$

The static terms all cancel out leaving only the dynamic losses through the core support plate.

The FIBWR core support plate pressure drop (ΔP_F) is defined on page 5-18 of EPRI-NP-1923 as the differential pressure between the inlet to the orifice

and the top of the core support plate. This involves both static and dynamic losses and can be expressed as:

$$\Delta P_F = \Delta P_{flow} + \rho X \qquad \text{(Eq. 7)}$$

where: ΔP_{flow} = flow losses through core support
X = elevation between the center line of the inlet orifice and the top of the core support plate. (See FIGURE 12a)

The difference between the FIBWR calculated pressure drop and the core support plate pressure drop from the pressure taps is:

$$\begin{aligned} \text{correction term} &= \Delta P_F - \Delta P_m \\ &= \Delta P_{flow} + \rho X - \Delta P_{flow} \\ &= \rho X \end{aligned}$$

FIGURE 12-b gives the location of the core support plate relative to the top of the fuel support piece. The location of the inlet orifice can be scaled from FIGURE 12-c to give a value of X equal to 4 inches.

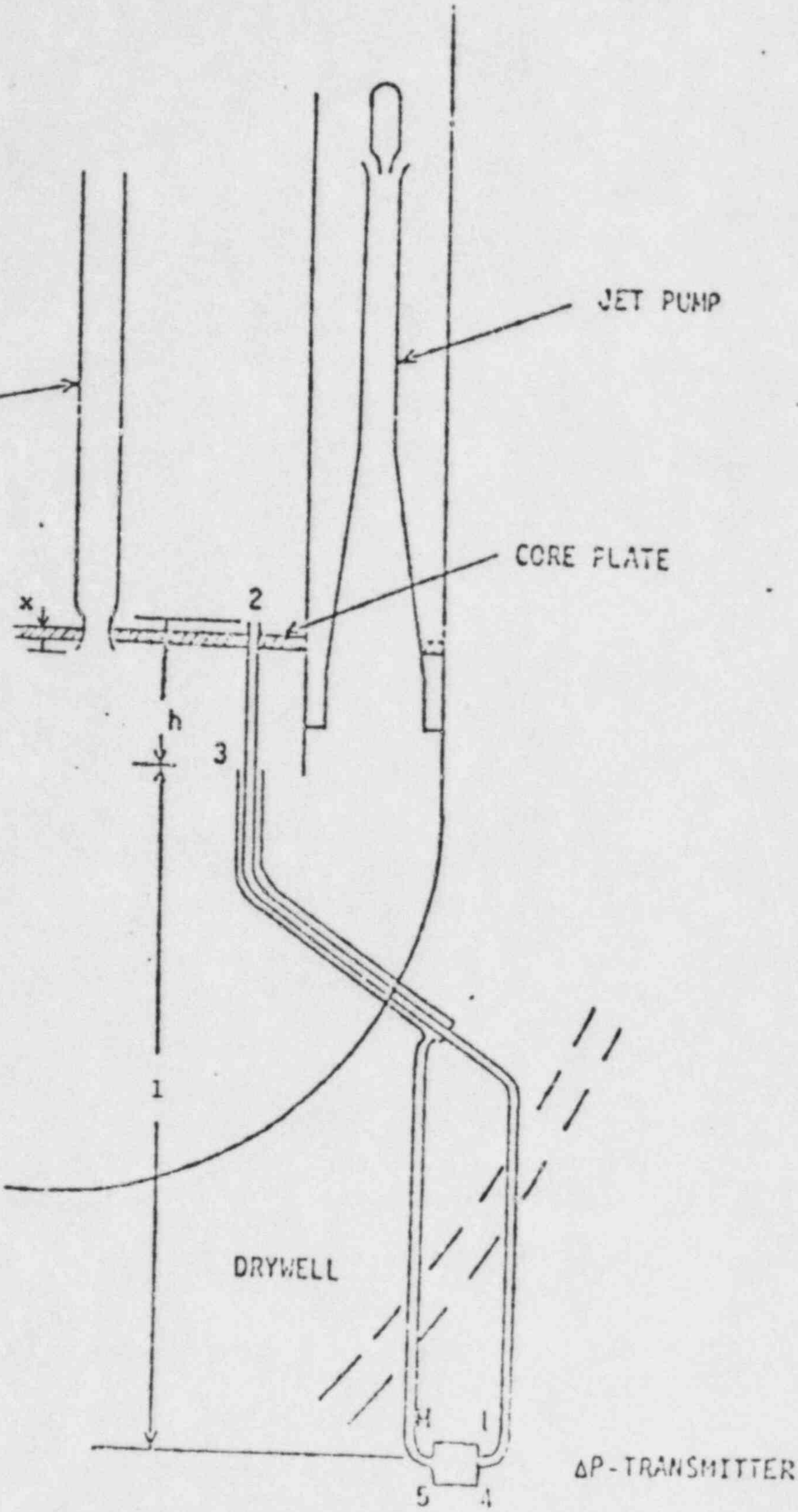
Using typical operating conditions in a BWR, $\rho = 46.17 \text{ lb/ft}^3$.

The correction term ρX becomes

$$\frac{(46.17 \text{ lb./ft}^3)\ (4.0 \text{ in})}{1728 \text{ in}^3/\text{ft}^3}$$

$= 0.11$ psi.

We consider 0.11 psi to be negligible and, therefore, no correction term was applied to the figures in Table 6 of the topical report.



JET PUMP
FUEL
BUNDLE
CORE PLATE
x
2
h
3
VESSEL
1
DRYWELL
H
L
5
4
ΔP-TRANSMITTER

FIGURE-12a

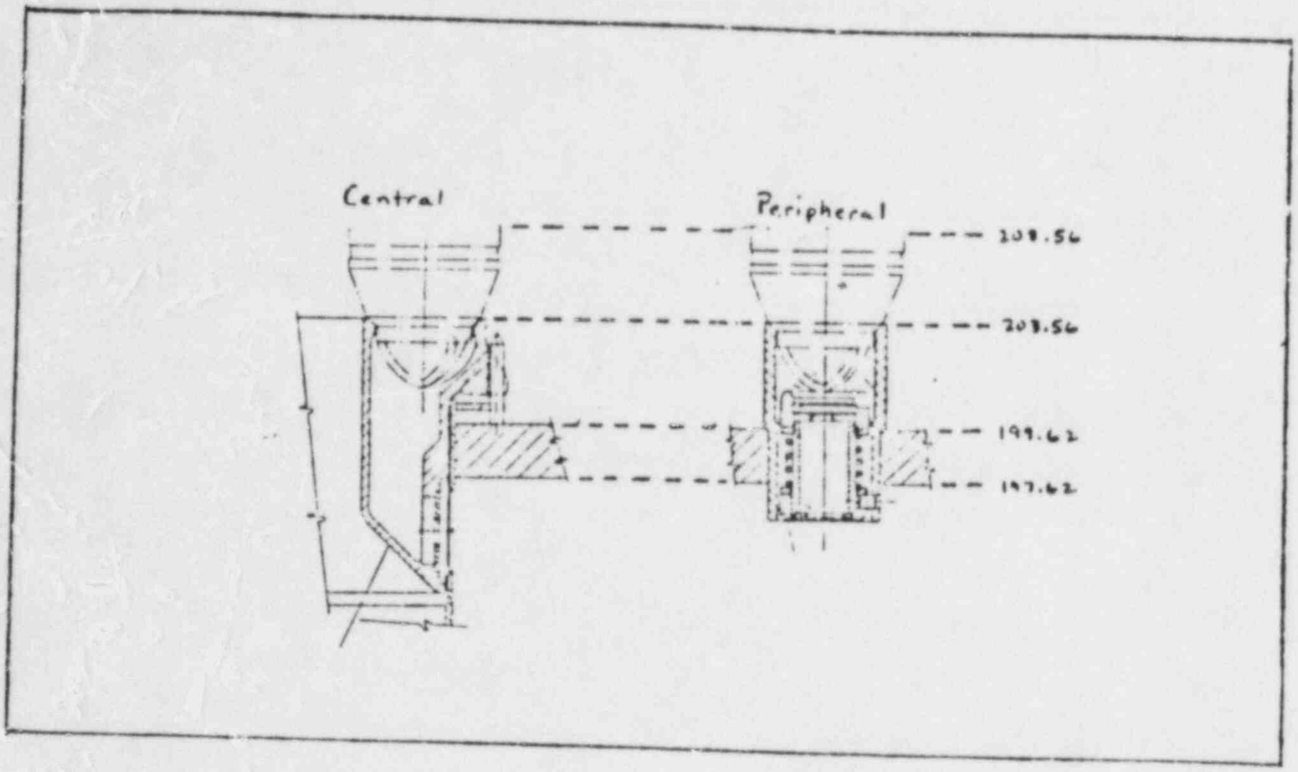


FIGURE-12b

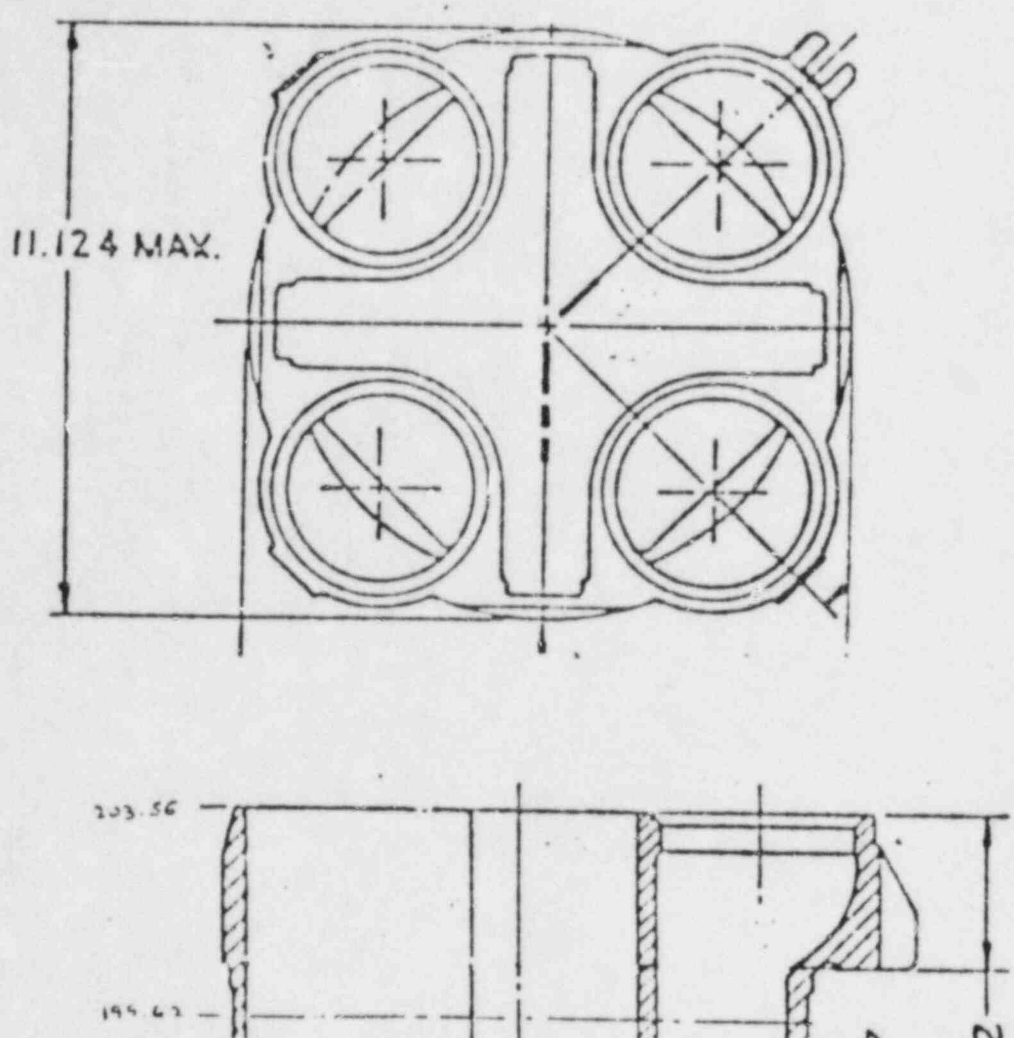


FIGURE-12c

Question 14

Provide the documents that give the vendor-calculated critical power ratios used in Figures 6 and 7 data comparisons.

Response

The local channel conditions and critical power ratios in Figures 6 and 7 of the Topical were obtained from the following documents:

1. General Electric Company: "General Electric Company Boiling Water Reactor Reload 1 Licensing Amendment for Brunswick Steam Electric Plant Unit 2," NEDO-24029, June 1977 (Attachment 14-1)

2. General Electric Company: "Supplemental Reload Licensing Submittal for Brunswick Steam Electric Plant Unit 1, Reload 2"; NEDO-24239; January 1980 (Attachment 14-2)

3. General Electric Company: "Supplemental Reload Licensing Submittal for Brunswick Steam Electric Plant Unit 2, Reload 2"; NEDO-24150; October 1978 (Attachment 14-3)

4. General Electric Company: "Supplemental Reload Licensing Submittal for Brunswick Steam Electric Plant Unit 1, Reload 3 (With Recirculation Pump Trip)"; Y1003J01A52; January 1983 (Attachment 14-4)

5. General Electric Company: "Supplemental Reload Licensing Submittal for Brunswick Steam Electric Plant Unit 1, Reload 3 (Without Recirculation Pump Trip)"; Y1003J01A53; January 1983 (Attachment 14-5)

6. General Electric Company: "Supplemental Reload Licensing Submittal for Brunswick Steam Electric Plant Unit 2, Reload 4 (Without Recirculation Pump Trip)"; Y1003J01A37; June 1982 (Attachment 14-6)

7. General Electric Company: "Supplemental Reload Licensing Submittal for Brunswick Steam Electric Plant Unit 2, Reload 4 (With Recirculation Pump Trip)"; Y1003J01A45; June 1982 (Attachment 14-7)

8. Letter to Mr. J. D. Martin of Carolina Power & Light Company From Mr. L. M. Quintana of General Electric Company; "Additional Brunswick Unit 2, Cycle 5 Confirmatory Load Line Limit Analysis Information"; LMQ83-101; September 27, 1983 (Attachment 14-8)

The cover page from each of the above documents, accompanied by the pages containing the local channel conditions, are provided as attachments. Documents 4 through 7 are draft versions of reload submittals which had not been published at the time that the local conditions were used for benchmarking purposes. Attachment 8 is a formal transmittal of data which was previously supplied to Carolina Power & Light Company by General Electric Company for information purposes only. The data point for the 100%, 7x7 channel case given in the formal transmittal differed from the initial data which was used in the FIBWR benchmark. This new data point has been evaluated using the GEXL

correlation in FIBWR, and the resulting change to Figure 7 of the topical is illustrated in Figures 14-a and 14-b. Figure 14-a is a reproduction of the original Figure 7, and Figure 14-b shows the same figure with the new 7x7 data point.

Attachment 14-1

NEDO-24027

GENERAL ELECTRIC BOILING WATER REACTOR RELOAD-1 LICENSING AMENDMENT FOR BRUNSWICK STEAM ELECTRIC PLANT UNIT 2

GENERAL ELECTRIC

Table 4-2

SUMMARY OF RESULTS OF LIMITING ABNORMAL OPERATIONAL TRANSIENTS

	Maximum ΔCPR	
EOC-2	7x7	8x8
Turbine Trip w/o Bypass	0.17	0.23
Load Rejection w/o Bypass	0.18	0.24
Loss of 100°F FW Heater	0.13	0.14
Inadvertent HPCI Pump Start	0.08	0.10
Rod Withdrawal Error (RBM set at 106%)	0.20	0.17
Feedwater Controller Failure - 135% NBR, run out capacity	0.05	0.06

Table 4-3

GETAB TRANSIENT ANALYSIS INITIAL CONDITION PARAMETERS

	7x7	8x8
Peaking Factors (Local, Radial, Axial)	(1.24, 1.24, 1.40)	(1.22, 1.345, 1.40)
R-Factor	1.100	1.098
Bundle Power (MWt)	5.294	5.738
Nonfuel Power Fraction	0.04	0.04
Core Flow (Mlb/hr)	77.0	77.0
Bundle Flow (10^3 lb/hr)	126.3	117.7
Reactor Pressure (psia)	1035	1035
Inlet Enthalpy (Btu/lb)	526.9	526.9
Initial MCPR	1.26	1.30

Attachment 14-2

NEDO-24239
80NED258
CLASS I
JANUARY 1960

SUPPLEMENTAL RELOAD LICENSING SUBMITTAL FOR BRUNSWICK STEAM ELECTRIC PLANT UNIT 1 RELOAD 2

5. STANDBY LIQUID CONTROL SYSTEM SHUTDOWN CAPABILITY (3.3.2.1.3)

ppm	Shutdown Margin (Δk) (20°C, Xenon Free)
600	0.045

6. RELOAD-UNIQUE TRANSIENT ANALYSIS INPUTS (3.3.2.1.5 AND 5.2)

	EOC3-2000 MWd/t to EOC3	BOC3 to EOC3-2000 MWd/t
Void Coefficient N/A* (¢/% Rg)	-8.50/-10.62	-8.70/-10.87
Void Fraction (%)	41.6	41.6
Doppler Coefficient N/A (¢/°F)	-0.225/-0.214	-0.220/-0.209
Average Fuel Temperature (°F)	1374	1374
Scram Worth N/A ($)	-37.93/-30.34	-36.72/-29.38
Scram Reactivity	Figure 2a	Figure 2b

7. RELOAD UNIQUE GETAB TRANSIENT ANALYSIS INITIAL CONDITION PARAMETERS (5.2)

	8x8		8x8R		P8x8R	
	EOC3-2000 MWd/t to EOC3	BOC3 to EOC3-2000 MWd/t	EOC3-2000 MWd/t to EOC3	BOC3 to EOC3-2000 MWd/t	EOC3-2000 MWd/t to EOC3	BOC3 to EOC3-2000 MWd/t
Peaking factor (local, radial, axial)	1.22, 1.34, 1.40	1.22, 1.40, 1.40	1.20, 1.47, 1.40	1.20, 1.54, 1.40	1.20, 1.45, 1.40	1.20, 1.52, 1.40
R factor	1.051	1.051	1.051	1.051	1.098	1.098
Bundle Power (MWt)	5.703	5.960	6.260	6.558	6.178	6.487
Bundle Flow (10^3 lb/hr)	114.2	112.1	114.3	112.1	115.4	113.0
Initial MCPR	1.30	1.24	1.30	1.24	1.32	1.25

*N - Nuclear input data
A - Used in transient analysis

Attachment 14-3

NEDO-24150
78NED288
CLASS I
OCTOBER 1978

SUPPLEMENTAL RELOAD LICENSING SUBMITTAL FOR BRUNSWICK STEAM ELECTRIC PLANT UNIT 2 RELOAD 2

GENERAL ELECTRIC

6. RELOAD UNIQUE TRANSIENT ANALYSIS INPUTS (3.3.2.1.5 and 5.2)

	EOC
Void Coefficient N/A* (¢/% Rg)	8.10/10.13
Void Fraction (%)	41.76
Doppler Coefficient N/A (¢/%°F)	0.1938/0.1841
Average Fuel Temperature (°F)	1538
Scram Worth N/A ($)	38.85/31.08
Scram Reactivity vs Time	Figure 2

7. RELOAD UNIQUE GETAB TRANSIENT ANALYSIS INITIAL CONDITION PARAMETERS (5.2)

Exposure	7x7 EOC	8x8 EOC	8x8R EOC
Peaking factors (local, radial and axial)	1.24/1.234/1.40	1.22/1.324/1.40	1.22/1.460/1.40
R-Factor	1.100	1.098	1.051
Bundle Power (MWt)	5.267	5.646	6.220
Bundle Flow (10^3 lb/hr)	125.82	116.24	117.03
Initial MCPR	1.25	1.32	1.32

8. SELECTED MARGIN IMPROVEMENT OPTIONS (5.2.2)

None

*N = Nuclear Input Data
A = Used in Transient Analysis

Attachment 14-4

Y1003J01A52
CLASS I
JANUARY 1983

SUPPLEMENTAL RELOAD LICENSING SUBMITTAL FOR BRUNSWICK STEAM ELECTRIC PLANT UNIT 1, RELOAD 3 (WITH RECIRCULATION PUMP TRIP)

GENERAL ELECTRIC

4. CALCULATED CORE EFFECTIVE MULTIPLICATION AND CONTROL SYSTEM WORTH - NO VOIDS, 20°C (3.3.2.1.1 AND 3.3.2.1.2)

Beginning of Cycle, k_{eff}	
Uncontrolled	1.114
Fully Controlled	0.958
Strongest Control Rod Out	0.989
R, Maximum Increase in Cold Core Reactivity with Exposure in Cycle, ΔK	0.0

5. STANDBY LIQUID CONTROL SYSTEM SHUTDOWN CAPABILITY (3.3.2.1.3)

ppm	Shutdown Margin (ΔK) (20°C, Xenon Free)
600	0.031

6. RELOAD UNIQUE TRANSIENT ANALYSIS INPUT (3.3.2.1.5 AND 5.2) (REDY EVENTS ONLY)

	EOC 4
Void Fraction (%)	41.3
Average Fuel Temperature (°F)	1302
Void Coefficient N/A* (¢/% Rg)	-8.33/-10.42
Doppler Coefficient N/A (¢/°F)	-0.232/-0.220
Scram Worth N/A ($)	-46.31/-37.05

7. RELOAD UNIQUE GETAB TRANSIENT ANALYSIS INITIAL CONDITION PARAMETERS (5.2)

Fuel Design	Peaking Factors Local	Peaking Factors Radial	Peaking Factors Axial	R-Factor	Bundle Power (MWt)	Bundle Flow (10^3 lb/hr)	Initial MCPR
BOC 4 to EOC 4							
P8x8R	1.20	1.57	1.40	1.051	6.671	109.9	1.20
8x8R	1.20	1.58	1.40	1.051	6.740	109.4	1.19
8x8	1.22	1.44	1.40	1.098	6.[illegible]	108.7	1.18

*N = Nuclear Input Data

A = Used in Transient Analysis

Attachment 14-5

Y1003J01A53
CLASS I
JANUARY 1983

SUPPLEMENTAL RELOAD LICENSING SUBMITTAL FOR BRUNSWICK STEAM ELECTRIC PLANT UNIT 1, RELOAD 3 (WITHOUT RECIRCULATION PUMP TRIP)

GENERAL ELECTRIC

4. CALCULATED CORE EFFECTIVE MULTIPLICATION AND CONTROL SYSTEM WORTH - NO VOIDS, 20°C (3.3.2.1.1 and 3.3.2.1.2)

BOC k_{eff}	
Uncontrolled	1.114
Fully Controlled	0.958
Strongest Control Rod Out	0.989
R, Maximum Increase in Cold Core Reactivity with Exposure into Cycle, Δk	0.0

5. STANDBY LIQUID CONTROL SYSTEM SHUTDOWN CAPABILITY (3.3.2.1.3)

ppm	Shutdown Margin (Δk) (20°C, Xenon Free)
600	0.031

6. RELOAD UNIQUE TRANSIENT ANALYSIS INPUT (3.3.2.1.5 and 5.2) (REDY EVENTS ONLY)

	EOC 4-2000 MWd/T	EOC 4
Void Fraction (%)	41.3	41.3
Average Fuel Temperature (°F)	1302	1302
Void Coefficient N/A* (¢/% Rg)	-8.12/-10.15	-8.33/-10.42
Doppler Coefficient N/A (¢/°F)	-0.219/-0.208	-0.232/-0.220
Scram Worth N/A ($)	-46.31/-37.05	-46.31/-37.05

7. RELOAD-UNIQUE GETAB TRANSIENT ANALYSIS INITIAL CONDITION PARAMETERS (5.2)

Fuel Design	Peaking Factors (Local, Radial, Axial)			R-Factor	Bundle Power (MWt)	Bundle Flow (10^3 lb/hr)	Initial MCPR
BOC 4 to EOC 4-2000 MWd/T							
P8x8R	1.20	1.53	1.40	1.051	6.516	111.0	1.23
8x8R	1.20	1.55	1.40	1.051	6.614	110.3	1.21
8x8	1.22	1.42	1.40	1.098	6.045	109.6	1.21

*N = Nuclear Input Data
A = Used in Transient Analysis

Attachment 14-6

Y1003J01A37
DRF L12-00306-1
REVISION 2
CLASS I
JUNE 1982

SUPPLEMENTAL RELOAD LICENSING SUBMITTAL FOR BRUNSWICK STEAM ELECTRIC PLANT UNIT 2, RELOAD 4

GENERAL ELECTRIC

7. RELOAD-UNIQUE GETAB TRANSIENT ANALYSIS INITIAL CONDITION PARAMETERS (S.2.2)

Fuel Design	Exposure (MWd/ST)	Peaking Factors (Local, Radial, Axial)	R-Factor	Bundle Power (MWt)	Bundle Flow (10^3 lb/hr)	Initial MCPR
8x8	EOC	(1.22,1.33,1.40)	1.098	5.686	113.5	1.29
	EOC-2000	(1.22,1.45,1.40)	1.098	6.187	110.2	1.18
8x8R	EOC	(1.20,1.46,1.40)	1.051	6.210	114.1	1.30
	EOC-2000	(1.20,1.59,1.40)	1.051	6.757	110.8	1.19
P8x8R	EOC	(1.20,1.43,1.40)	1.051	6.092	115.6	1.33
	EOC-2000	(1.20,1.57,1.40)	1.051	6.664	112.1	1.21

8. SELECTED MARGIN IMPROVEMENT OPTIONS (S.2.2.2)

Transient Recategorization: No

Recirculation Pump Trip: No

Rod Withdrawal Limiter: No

Thermal Power Monitor: Yes

Measured Scram Time: No

Exposure Dependent Limits: Yes

Exposures Analyzed: EOC and EOC-2000 MWd/ST

9. CORE-WIDE TRANSIENT ANALYSIS RESULTS (S.2.2.1)

Transient	Exposure (MWd/ST)	$\dot{\phi}$ (% NBR)	$\dot{Q}/A$ (% NBR)	ΔCPR (Uncorrected) 8x8	8x8R	P8x8R	Plant Response
Load Rejection Without Bypass	EOC	582	128	0.22	0.23	0.26	Figure 2a
	EOC-2000	447	117	0.11	0.12	0.14	Figure 2b
Loss of 100°F Feedwater Heating	BOC-EOC	124	122	0.13	0.13	0.13	Figure 3
Feedwater Controller Failure	EOC	127	110	0.04	0.04	0.05	Figure 4a
	EOC-2000	113	110	0.04	0.05	0.05	Figure 4b

Attachment 14-7

Y1003J01A45
DRF L12-00306-1
REVISION 1
CLASS I
JUNE 1982

SUPPLEMENTAL RELOAD LICENSING SUBMITTAL FOR BRUNSWICK STEAM ELECTRIC PLANT UNIT 2, RELOAD 4

GENERAL ELECTRIC

5. STANDBY LIQUID CONTROL SYSTEM SHUTDOWN CAPABILITY (3.3.2.1.3)

ppm	Shutdown Margin (Δk) (20°C, Xenon Free)
600	0.036

6. RELOAD-UNIQUE TRANSIENT ANALYSIS INPUT (3.3.2.1.5 AND S.2.2)

	EOC
Void Coefficient N/A* (¢/% Rg)	-7.36/-9.92
Void Fraction (%)	41.8
Doppler Coefficient N/A (¢/% °F)	-0.219/-0.208
Average Fuel Temperature (°F)	1312
Scram Worth N/A ($)**	---

7. RELOAD-UNIQUE GETAB TRANSIENT ANALYSIS INITIAL CONDITION PARAMETERS (S.2.2)

Fuel Design	Exposure	Peaking Factors (Local, Radial, Axial)	R-Factor	Bundle Power (MWt)	Bundle Flow (10^3 lb/hr)	Initial MCPR
8x8	EOC5	(1.22, 1.43, 1.40)	1.098	6.114	110.7	1.20
8x8R	EOC5	(1.20, 1.57, 1.40)	1.051	6.689	111.2	1.20
P8x8R	EOC5	(1.20, 1.56, 1.40)	1.051	6.620	112.4	1.22

8. SELECTED MARGIN IMPROVEMENT OPTIONS (S.2.2.2)

Transient Recategorization: No

Recirculation Pump Trip: Yes

Rod Withdrawal Limiter: No

Thermal Power Monitor: Yes

Measured Scram Time: No

Exposure Dependent Limits: No

*N = Nuclear Input Data
A = Used in Transient Analysis

**Generic exposure independent values are used as given in "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A-4, January 1982.

Attachment 14-8

GENERAL ELECTRIC

NUCLEAR ENERGY BUSINESS OPERATIONS

GENERAL ELECTRIC COMPANY • 175 CURTNER AVENUE • SAN JOSE, CALIFORNIA 95125

September 27, 1983
LMQ:83-101

cc: JH Craven
*JS Dietrich
CR Dietz
*JJ Gitnick
RA Hanvelt
*RT Hill
RG Matthews
*JP Rea
*AF Wenger
*EB Wilson

*with attachments

Mr. J. D. Martin
Fuel Department
CAROLINA POWER & LIGHT COMPANY
P. O. Box 1551
Raleigh, NC 27602

SUBJECT: Additional Brunswick 2 Cycle 5 Confirmatory Load Line Limit Analyses Information

REFERENCES: 1) "Load Line Limit Analysis Reverification for Brunswick Steam Electric Plant Unit 2, Reload 4 (Without Recirculation Pump Trip)", NEDO-22088, July 1983.

2) "Load Line Limit Analysis Reverification for Brunswick Steam Electric Plant Unit 2, Reload 4 (With Recirculation Pump Trip)", NEDO-22228, July 1983.

Dear Jack:

Draft copies of the referenced documents were given to CP&L for review on August 9, 1983. These documents report the results of analyses performed by GE to confirm the applicability of the original Brunswick 2 load line limit analysis (LLLA) to Cycle 5. Attached for your information are the GETAB transient analysis initial input condition parameters for the events analyzed to support operation above the 100P/100F load line in Cycle 5. Attached as well are eight figures with the transient plots for those events.

Please let me know if CP&L has any questions or comments about the referenced documents or the attached information. We are delaying final printing of the Cycle 5 LLLA documents pending conclusion of CP&L's review.

Very truly yours,

Louis M. Quintana

L. M. Quintana
Fuel Project Manager
Brunswick 1/2
M/C 174; (408) 925-2026

rem
Attach.

Brunswick 2 Cycle 5

GETAB Transient Analysis Initial

Condition Parameters

Fuel Design	Exposure	Peaking Factors (Local, Radial, Axial)	R-Factor	Bundle Power (MWt)	Bundle Flow (10^3 lb/hr)	Initial MCPR
		100% Power - 94% Flow - with RPT				
		(Events Analyzed: TTNBP, FWCF, LFWH, Inadvertent HPCI)				
P8x8R	EOC5	1.20,1.54,1.40	1.051	6.536	105.4	1.22
8x8R	EOC5	1.20,1.56,1.40	1.051	6.628	104.2	1.20
8x8	EOC5	1.22,1.42,1.40	1.098	6.041	103.9	1.20
7x7	EOC5	1.24,1.28,1.40	1.100	5.446	116.6	1.19
		100% Power - 94% Flow - without RPT				
		(Events Analyzed: LRNBP, FWCF, LFWH, Inadvertent HPCI)				
P8x8R	EOC5	1.20,1.44,1.40	1.051	6.113	107.9	1.31
8x8R	EOC5	1.20,1.46,1.40	1.051	6.229	106.5	1.28
8x8	EOC5	1.22,1.34,1.40	1.098	5.714	106.0	1.27
7x7	EOC5	1.24,1.25,1.40	1.100	5.309	116.7	1.22
		85% Power - 61% Flow				
		(Events Analyzed: LFWH, Inadvertent HPCI)				
P8x8R	EOC5	1.20,1.64,1.40	1.051	5.905	67.53	1.18
8x8R	EOC5	1.20,1.64,1.40	1.051	5.902	67.2	1.18
8x8	EOC5	1.22,1.50,1.40	1.098	5.410	66.8	1.17
7x7	EOC5	1.24,1.36,1.40	1.100	4.928	74.5	1.16

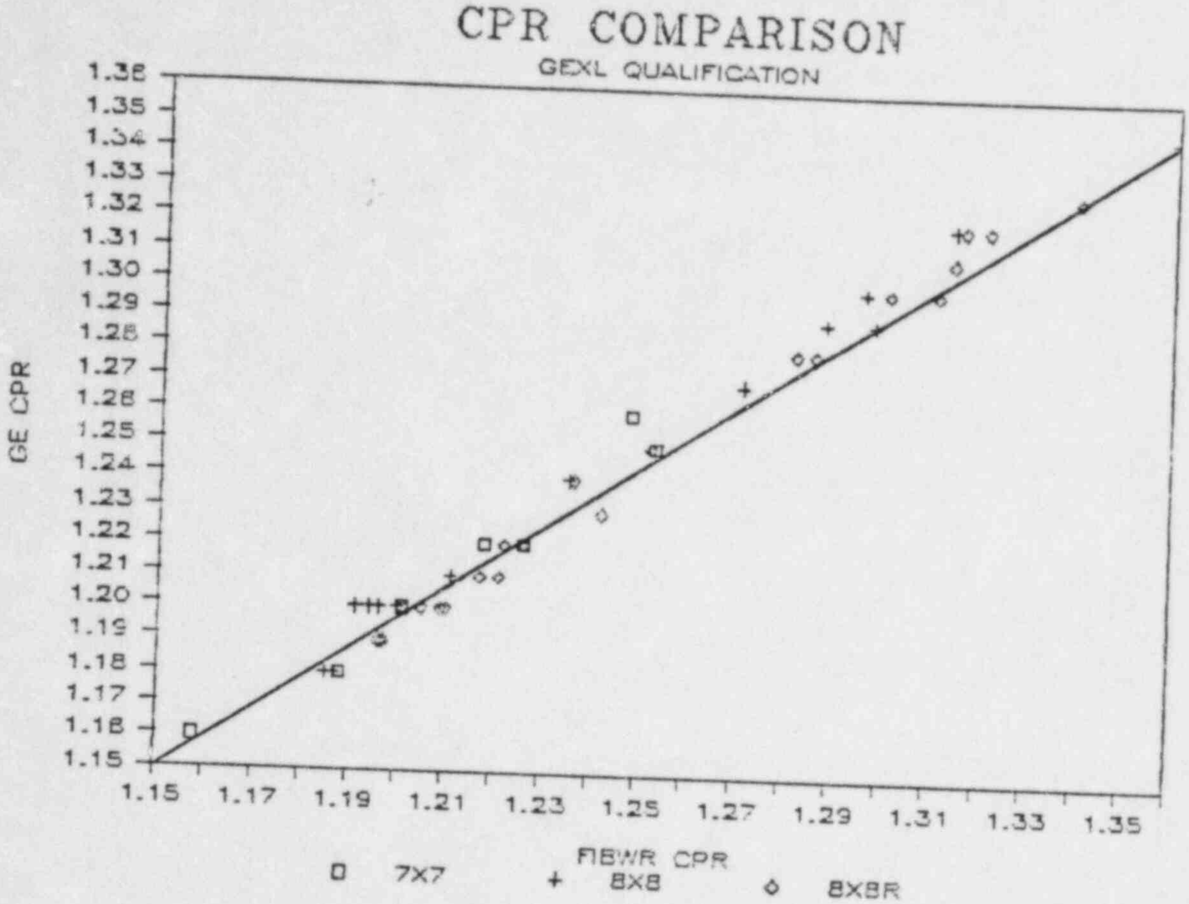


Fig 14-a

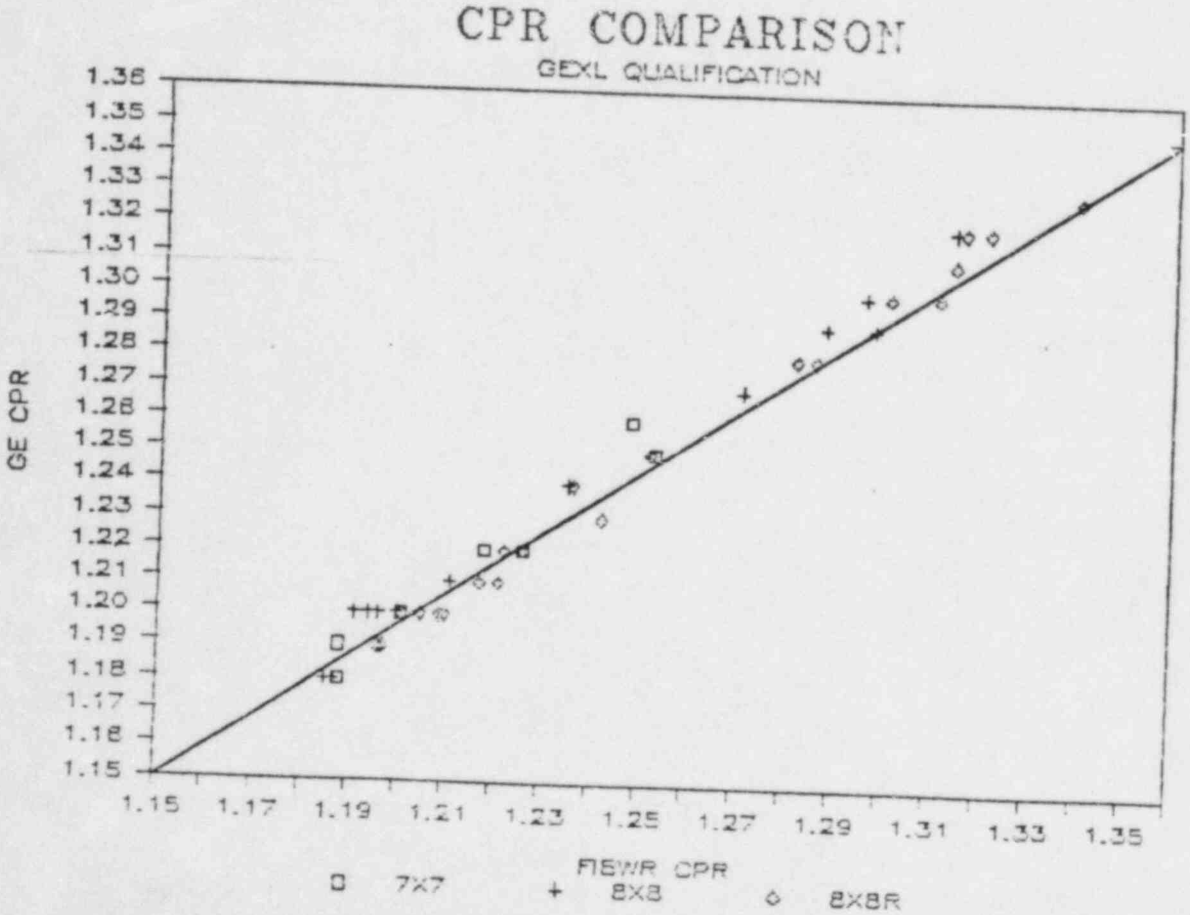


Fig 14-b

Question 15

Discuss the sensitivity of the axial node sizes on the thermal-hydraulic results. The Topical Report presented results using one channel to represent one fuel bundle. If "collapsed," channels (one channel representing several fuel bundles) are intended for future analysis, discuss how it will be approached and the sensitivities on hot bundle parameters.

Response

Sensitivity studies were conducted to determine the effects of changing the axial node size and the number of fuel regions on hot bundle parameters. The 75-channel, high-power, high-flow case from Brunswick Unit 1, Cycle 3 described in the topical (Figure 5C) was used as the base condition for these studies.

The 75-channel case depicts an eighth-core symmetric model, each channel representing eight assemblies, with the exception of those along the diagonal, which represent four assemblies. This model is useful for evaluating the effects of radial peaking factors in adjacent channels, but can be expensive to run. The preferred method for core-wide analysis is to lump channels according to common geometry and orifice type. In doing this, the 75-channel case can be represented as three channels: central 8x8, central 8x8 retrofit, and peripheral 8x8. A fourth channel depicting a single hot bundle can be added for hot channel work. This compressed core representation will be used by Carolina Power & Light Company for most hydraulic applications and produces core-wide and hot channel results very similar to those of the eighth core model as shown in Table 15-A.

A compressed core model with a single hot channel was used to study the effects of changing the axial node size. A 24-node axial power shape was reduced to twelve, eight, and six nodes by averaging over two nodes, three nodes, and four nodes, respectively. A FIBWR model was run, varying only the number of axial nodes; and the results of these executions are shown on Table 15-B. The hydraulic parameters appeared very insensitive to the axial node size with a 0.8 percent change in the hot channel void fraction and only a 0.2 percent change in the core pressure drop and the minimum critical power ratio. CP&L will use a 24-node power shape for most FIBWR applications to remain consistent with the neutronic codes.

A more detailed study was performed on the effects of collapsing channels together. Four cases examined are described below:

Case 1 - An eighth-core symmetric model of Brunswick Unit 1, Cycle 3 with the hot channels representing four assemblies.

Case 2 - A four-channel compressed model of Brunswick Unit 1, Cycle 3 with a single assembly as the hot channel.

Case 3 - A two-channel compressed model of a full-core of 8x8 retrofit assemblies. A hot channel represents 280 assemblies.

Case 4 - A two-channel mod... of a full-core of 8x8 retrofit assemblies. A single assembly is the hot channel, and all remaining assemblies are compressed into one channel.

In all four cases, the core pressure drop and the radial power of the hot channel was fixed and the code solved for core flow and hot channel conditions.

The results shown on Table 15-C indicate that the hot bundle parameters are defined primarily by the power produced in that channel and are independent of how many channels are represented or how the remainder of the core is modelled.

TABLE 15-A

# OF CHANNELS	HOT PEAKING FACTOR	CORE P	HOT CHANNEL FLOW (Mlb/hr)	MCPR
75	1.3952	20.7577	116.58	1.3502
4	1.3952	20.7472	116.42	1.3498

TABLE 15-B

# OF NODES	CORE P	HOT VOID FRACTION	TOTAL BYPASS FRACTION	MCPR
24	20.7472	0.4495	0.1130	1.3498
12	20.7443	0.4482	0.1131	1.3496
8	20.7512	0.4483	0.1132	1.3519
6	20.7800	0.4459	0.1133	1.3528

TABLE 15-C

CASE	HOT PEAKING FACTOR	CORE P	HOT CHANNEL FLOW (Mlb/hr)	MCPR
1	1.3952	20.7577	116.58	1.3502
2	1.3952	20.7577	116.44	1.3498
3	1.3952	20.7577	116.47	1.3499
4	1.3952	20.7577	116.47	1.3499