Babcock & Wilcox
a McDermott company

Nuclear Power Division
3315 Old Forest Road
P.O. Box 10935
Lynchburg, VA 24506-0935
(804) 385-2000

October 26, 1988
ESC-939

Mr. Joel D. Page
Task Manager, Section B,
Engineering Issues Branch
Office of Nuclear Regulatory Research
Mail Stop NL/S-302
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Page:

Please find attached final responses to the NRC questions regarding Generic Issue No. 79, RV Thermal Stresses. These responses were originally submitted via letter of June 23, 1988 and were inadvertently labeled "draft" responses on the title page. The title page has been corrected with no other changes made to those original responses. The response document is attached hereto, and replaces the original issue in total.

Very truly yours,

JRPaljug

J. R. Paljug
Project Manager
Owners Group Engineering Services

JRP/leh

Attachment

cc: B&WOG Analysis Committee

C. H. Turk - AP&L
P. F. Guill - DPCo
J. E. Burchfield, Jr. - DPCo
E. H. Davidson - FPC
A. Irani - GPUN
J. K. Atwell - SMUD
J. F. Dunne - TED

DF05
1/1

Final Response to NRC Questions

I. General

Question 1 The background section of Reference 1 states that thermal gradients can be created during the transition to decay heat removal system or during a natural circulation cooldown. However, the analysis only addresses the natural circulation cooldown transient. Please provide justification for not considering the transition to decay heat removal system operation in this analysis. Additionally, please provide justification for considering only 20 natural circulation cooldown cycles.

Response 1 - Transition to the decay heat removal system was considered. The pressure vs temperature curve used for the stress analysis input, Reference 2 Page 97, includes transition to the decay heat removal system (DHRS) beginning at approximately 11 hours. Total primary system flow before and after DHRS actuation are estimated to be quite similar. Natural circulation flow is estimated at approximately 8500 GPM, and DHRS flow is approximately 7500 GPM. Consequently, the change in flowrate into the upper head region following actuation of the DHRS would not be significant.

The B&W plants' original design basis was to remain at hot shutdown following an unplanned NSS transient until a normal plant cooldown could be initiated. Natural circulation capability was designed into the plant as a means of removing decay heat following loss of forced flow, but natural circulation cooldowns were not included in the original design basis of the plants. In recent years natural circulation capability has been viewed as a significant contributor to achieving cold safe shutdown following severe plant upsets and has been evaluated at various times to address operational or technical concerns, including RV head stress issues. For these types of evaluations B&W has recommended a relatively small number of cycles (20) for natural circulation cooldowns. This value is consistent with other emergency category cycles specified in the RCS Functional Specifications and the component stress reports in the original design basis. Operating plant experience supports the use of 20 cycles as a reasonable upper limit for the number of natural circulation cooldown events. To date, no natural circulation cooldown has occurred at a B&W plant.

The design basis of 240 cycles (Table 5.2-1 of Oconee FSAR) is only applicable to a normal forced flow cooldown. A natural circulation cooldown is considered an emergency event and is limited to 20 cycles. The total number of cycles for both forced flow cooldown and natural circulation cooldown is 260 cycles.

Sentence number 4 of the Background section of the stress analysis report Reference 1 should be revised as follows: These gradients could develop as a result of non-uniform cooling of the reactor coolant within the reactor vessel that are created during a natural circulation cooldown and the subsequent transient to decay heat removal system operation following the natural circulation cooldown.

Question 2 Section 4 of Reference 1 gives the coolant cooldown rate as 20 F to 100 F/h; however, only a 50 F/h cooldown rate is used in the analysis. Please provide justification for selecting only the cooldown rate of 50 F/h.

Response 2 - A natural circulation cooldown of the NSSS, for all B&W plants excluding the Oconee units, would occur only when offsite power is lost for an extended period of time. The loss of offsite power precludes the use of the turbine bypass valves for cooldown of the primary thus requiring a cooldown with the atmospheric dump valves (ADV's). The total capacity of the ADV's in B&W plants in general does not exceed 10% of the total rated steam flow, thus limiting the long-term primary natural circulation cooldown rate to 50 F/h or less.

Oconee Units 1, 2 and 3 have implemented procedural limits of 50 F/h for natural circulation cooldowns.

II. Heat Transfer Analysis

Question 1 Provide justification for selecting a temperature difference of 15 F and a vertical plate height of 50 inches to compute the void level heat transfer coefficient for Fluid Block 1, transient 3 (pages 40 to 43 of Reference 2). Can the film coefficients calculated for the void level for all the transient conditions be considered as upper limits.

Response 1 - A vertical plate height of 50 inches roughly corresponds to the elevation of the liquid level above the plenum cover. For transient condition 3, the fluid block element temperatures below the steam void level are assumed to equal the core outlet temperature.

The temperature differential between the liquid and the wall was chosen to be 15 F, which is a reasonable approximation based upon the detailed finite difference fluid model which was used to generate the inner surface film heat transfer coefficient (Reference 3).

The film coefficient calculated on pages 39 - 43 of Reference 2 is only applicable to fluid block 1 for transient condition 3. The film coefficient for the voided region is discussed in question 3.

Question 2 Page 5 of Reference 3 deduces a top region (Fluid Block 1) heat transfer coefficient of 115 BTU/ft2-h-F for a "non-venting case," to be used in Reference 2, but Reference 2 considers open vent valves (see Page 55, with Fluid Block 5). Is there a conflict? What is the effect of vent valves on the Fluid Block 1 heat transfer coefficient?

Response 2 - Venting is in reference to the continuous vent line which connects the reactor vessel upper head to the hot leg. Consequently, the "non-venting case" in reference 3 refers to an upper head cooldown analysis which was performed without the vent line attached.

The reactor vessel vent valves are expected to be active during the natural circulation cooldown. The fluid flow into the upper head region (fluid block 1- transient 3), with reactor vessel vent valves operating, is estimated to be approximately 10% higher than with no vent valves. Vent valve operation would result in a slightly higher heat transfer coefficient for fluid block 1.

Question 3 Section 2 of Reference 1 refers to a "void" forming in the upper head. Is this superheated steam or saturated vapor? Figure 3 of Reference 3 gives the film coefficient as being for subcooled water. Why was this value chosen?

Response 3 - The void in the upper head is assumed to be saturated steam at 600 F. The film coefficient for subcooled liquid was mistakenly applied to the steam region. However, there is high resistance to heat flow between the head and the vessel. The only paths are: from the head, through an air gap to the studs, and then back to the vessel; and from the head to the vessel through 2 small "O" rings. Therefore, the artificially high film coefficient on the inside of the head will not change the stresses, up or down, significantly.

Question 4 The overall U values prepared in Reference 3 for use in Reference 2 were inadvertently misquoted in Reference 1 (page 4-3) and Reference 2 (Page 11), which should read:

RV head inner surface 115 BTU/ft2-h-F;

RV head outer surface .23 BTU/ft2-h-F; (upper region)

RV head outer surface 2.83 BTU/ft2-h-F: (lower region)

Response 4 - That is correct. The heat transfer coefficients are:

RV head inner surface film heat transfer 115 BTU/ft2-h-F (subcooled liquid)

RV head outer surface .23 BTU/ft2-h-F (upper region, which includes an air gap between the metal and the insulation)

RV head outer surface 2.83 BTU/ft2-h-F (lower region, which does not include an air gap between the metal and the insulation)

The RV head inner surface heat transfer coefficient is applied with respect to the temperature difference between the upper head fluid and the metal wall inside surface temperature. The RV head outer surface coefficients are overall heat transfer coefficients with respect to the difference between the average metal temperature and containment temperature.

Question 5 The outer heat flows used in Reference 3 to deduce overall effective U's are varying spatially. What is the effect of the averaging done here? One would expect that the thermal insulation is the main thermal resistance, and since that is apparently 3 inches thick everywhere (Table 4-3 and Figure 4-4 of Reference 4) one would expect it not to vary by more than one order of magnitude between top and sides. What is the insulation material? What are its thermal conductivity and emissivity?

Response 5 - The representative U's for the lower region and upper region were obtained by adding the representative heat fluxes and dividing by the temperature differential between the average metal temperature and the containment temperature for the respective regions. The values represent an average U for each of the regions. The spatial variation in the heat transfer coefficients is insignificant (i.e standard deviation is small).

The overall heat transfer coefficient for the lower region (2.83 Btu/h-ft2-f) was found to be high by an order of magnitude. B&W has determined the impact of the error to be conservative. The combination of a higher heat transfer coefficient in the lower upper head region, which would yield lower metal temperatures, coupled with the higher temperatures predicted for the upper head region, a result of the 115 Btu/h-ft2-f heat transfer coefficient applied to the inner surface of the steam region, yields conservatively high RV closure head and upper shell region stress predictions.

The insulation material is mirror insulation with the following heat transfer properties:

- o Thermal conductivity = .070 BTU/h-ft-F
- o Density = 14.4 lbm/ft3
- o Specific heat = .134 Btu/lbm-F
- o Surface Emissivity = .160

<u>Question 6</u> On pages 19 and 20 of Reference 2 an effective film coefficient is computed to represent radiation heat transfer. Provide justification for selecting T2= 450 F (or 910 R) and dt= 50 R or 40 R for computing the film coefficient.

<u>Response 6</u> - The choice of T2 = 450 F and a DT = 50 F were assumptions based upon the engineers knowledge of the steady state temperature distribution from previous analyses. The

corresponding heat flux for this region agrees well with the heat flux calculated in reference 3.

Question 7 A convective heat transfer coefficient for the closure head, identified as Fluid Block 1, is computed on pages 39 through 44 of Reference 2. Several questions are applicable to this calculation.

Question 7A Provide justification for the mass flow rate in this region to be 8% of the natural circulation loop flow.

Response 7A - Detailed upper head thermal-hydraulic calculations have been prepared by B&W which predict flowrates into the upper head region, above the plenum cover, to equal 8% to 12% of the total system flow. The flow paths leading from the core exit to the hotlegs are presented in Figure 1. A description of each flow path is provided below:

PATH	PATH DESCRIPTION
1	From fuel assembly to open plenum area
2	From fuel assembly to column weldments (control rod guide tubes)
3	Through lower exit ports in column weldments
4	Through 3 inch diam. holes in plenum cyl. to outlet nozzle
5	Through 22 in. and 34 in. diam. holes in plenum cyl.
	Over top of plenum cyl. into outer annulus

7 From column weldments (69 of them) into upper head and to outlet plenum

8 Through outlet nozzle

The percentage of total flow through path 7 has been calculated by B&W to be 10.5% of the total flow through the hot legs. The flow, however, penetrates approximately 6 to 12 inches above the plenum cylinder cover. Consequently, the heat transfer correlation for fluid block 1 (liquid portion) should not be calculated via forced convection and in fact was calculated based upon natural or free convection (pg 42 ref. 2).

Question 7B Justify computing the velocity using the flow area that was used for Fluid Block 4.

Response 7B - The final film coefficient was correctly based upon natural convection as opposed to forced convection (pg. 42 of Ref. 2). Consequently the velocity calculation with respect to forced convection is not applicable.

Question 7C The characteristic length L is defined as L = De/2 in the heat transfer correlation. Why is it used as L = De in the Reynolds Number calculation?

Response 7C - Please see response 7B.

Question 7D Are the values of length and flow area used consistent with each other?

Response 7D - Please see response 7B.

Question 7E Provide justification for using a 15 F difference between the fluid and the metal.

Response 7E - Please see response to question 1.

Question 8 Provide justification for assuming the flow rate to be 8% of the natural circulation loop flow rate in the evaluation of the film coefficient for fluid block 4. Provide justification for using a correlation for forced convection as opposed to the natural convection correlation used for Fluid Block 1 with the same flow rate.

Response 8 - The 8% flow, which penetrates the upper head above the plenum cover, returns to the outlet annulus as shown on pg 51 of Reference 2. A forced convection correlation is therefore applicable for fluid block 4.

III. Thermal Stress Analysis

Question 1 Since the inlet and outlet openings in the reactor vessel are localized, provide justification for the choice of a finite element model for the stress analysis which is axisymmetric about the reactor vessel vertical axis. Also, provide details regarding the degree of accuracy in the modeling of the temperature and film coefficients within the vessel as axisymmetric.

Response 1 - The vessel nozzle openings were designed in accordance with the area replacement rule (ASME Boiler and Pressure Vessel Code, Section III, Subsection NB, Paragraph NB 3332.2). Thus, on any plane passed through the opening, the area of metal removed by the opening is replaced by reinforcement in a localized area around the opening. Therefore the stiffness of the vessel is essentially axisymmetric.

During the natural circulation cooldown transient analyzed, reactor coolant flow in the system is very low. Therefore, below the assumed void line, flow in the reactor vessel in the axial and tangential directions are both low and there is no inaccuracy in the axisymmetric modeling of film coefficients. Above the assumed void line, flows in all directions are essentially zero.

Question 2 Provide justification for not including in the analysis the reaction forces on the reactor nozzles.

Response 2 - The reaction forces due to system heat-up are small. Recent calculations indicate that the stresses in the 3.3 inch thick hot leg (largest) pipe at the reactor vessel are on the order of 500 psi. The resultant stress in the 8.4 inch thick reactor vessel shell are even smaller and therefore not significant to the analysis.

Question 3 In Section 9.0 of Reference 2 it is mentioned that the thermal stress analyses were performed for three structural models. For all the stress analyses the shear stress in the vessel closure bolts have not been reported. What is the magnitude of the shear stresses in the vessel closure bolts for case 1, which uses a friction coefficient of zero at the flange interfaces? Included in the above question are the thermal stress runs made to comply with review comments 2 and 6 (page 94 of Reference 2).

Response 3 - Three structural models were run for Transient Condition 1 and two structural models each for Transient Conditions 2 and 3. The in-plane results for the studs are shown on pages 79, 80 and 81 of Reference 2. The transverse shear stresses in the studs are small and do not affect the analysis result. This is shown below where the shear stress, extracted from the computer runs, are combined with the membrane stresses to obtain the resultant stress intensities.

The analyses performed in response to review comments 2 and 6 give similar results.

TRANSIENT CONDITION	CASE NO.	LOAD STEP TIME-HRS	AVG. MEMBRANE STRESS-PSI	MAX. SHEAR STRESS-PSI	STRESS INTENSITY PSI
1	1	0	35,609	56	35,609
1	1	6	35,130	165	35,132
1	1	24	37,998	59	37,998
1	2	0	35,050	300	35,055
1	2	6	35,412	284	35,417
1	2	24	38,797	304	38,802
1	3	0	35,348	419	35,358
1	3	6	35,411	284	35,416
1	3	24	38,796	303	38,801
2	1	0	35,554	521	35,569
2	1	6	34,374	148	34,375
2	1	24	36,925	86	36,925
2	2	0	35,000	300	35,005
2	2	6	34,700	287	34,705
2	2	24	37,794	311	37,799
3	1	0	34,850	547	34,867
3	1	6	22,169	435	22,186
3	1	24	21,517	457	21,536
3	2	0	34,230	301	34,235
3	2	6	21,554	193	21,557
3	2	24	21,018	254	21,024

Question 4 The radial displacement of stud node 921 (Figure 4 of Reference 2, page 26) and flange node 259 at the interface were coupled to allow for stud shear force and bending moment to be transmitted to the vessel flange at two locations (page 96 of Reference 2). Subsequently, the thermal stress analysis for transient condition 2 and structural cases 1 and 2 was repeated. Why is a linear constraint equation, such as the one at the bottom of page 34 of reference 2, not provided between nodes 259 and its adjacent shell node? In addition, why are nodes 259 and 921 not coupled in the vertical direction?

<u>Response 4</u> - The stud model was coupled to the vessel at one location each at the top and bottom. The coupling allows transfer of vertical and radial (shear) forces and bending moments. The top connection represents the bearing of the stud nut against the top of the closure head flange and is representative of the actual configuration. The single connection at the bottom of the stud is conservative since all loads are transferred to a point instead of through the entire length of the engaged threads as in the actual configuration. The location of the bottom of the stud model is at an intermediate point along the length of engaged threads.

<u>Question 5</u> For the analysis done to comply with review comments 2 and 6 the reactor vessel was assumed to be free to grow radially at the bottom of the nozzle belt region. Does this boundary condition have any effect on limiting the strains in the vessel and on the magnitude of the shear forces in the vessel closure bolts? In addition, it is not totally clear from the answer to review comment 3, page 96 of Reference 2, if the reaction of the plenum cover on the upper head was considered in the initial thermal/stress analysis, or only in the analysis performed after the review comments.

<u>Response 5</u> - The stresses in the reactor vessel in the area of interest are unaffected by the radial boundary conditions at the end of the model since the model cylindrical length is greater

than the characteristic length of the vessel. The cylindrical model length is 100 inches while the characteristic length of a cylinder 180 inches in diameter and 8.4 inches in thickness is 64 inches.

The original analysis did not include the plenum cover reaction on the upper head. This omission was corrected in the analysis performed after the review comments.

Question 6 In Table 1 (page 22) of Reference 2 the coefficients of thermal expansion, , are average coefficients of thermal expansion in going from 70F to the indicated temperature. Provide justification for using the average values in lieu of the instantaneous values.

Response 6 - The ANSYS computer code uses average coefficients of thermal expansion in tabular form. For intermediate temperatures, the code interpolates between points in the table.

Question 7 On page 27 of Reference 2, the effective length of the bolt was assumed to be 40.25 inches. Provide justification for choosing this bolt length to calculate the bolt prestress.

Response 7 - The bolt (stud) length was modeled as the total free length plus one-half of the length of engaged threads. This yields an axial stiffness which is in close agreement with data measured during stud tensioning. The stud preload is given, not calculated. (A copy of a typical stud tensioner data sheet is attached. The Average Strain readings are in micro inches per inch). The computer analysis is used to iterate on the stud preload until the given value is achieved; therefore the preload is not determined by the stud length.

REFERENCES

1. "Stress Analysis of the Reactor Vessel Closure Region for a Natural Circulation Cooldown Transient," Babcock & Wilcox Owners Group Analysis Committee, Report No. 77-1152846-00, July 1984.

2. "Stress Analysis of the Reactor Vessel Closure Region for a Natural Circulation Cooldown Transient," Babcock & Wilcox, Calculation No. 32-1151155-00, June 21, 1984.

3. B. L. Bowlman, "Reactor Vessel Head Stress Analysis Inputs," Calculation No. 32-1150499-00, B&W, March 18, 1984.

4. Crystal River Unit 3 FSAR.

5. General Functional Specification for R.C.S. Components for CPCO, No. 18-1092000012-05.

6. R. W. Winks, R. C. Twilley, Jr., "Natural Circulation Occurrences at Operating B&W Plants," Babcock & Wilcox Co., May 7, 1979.

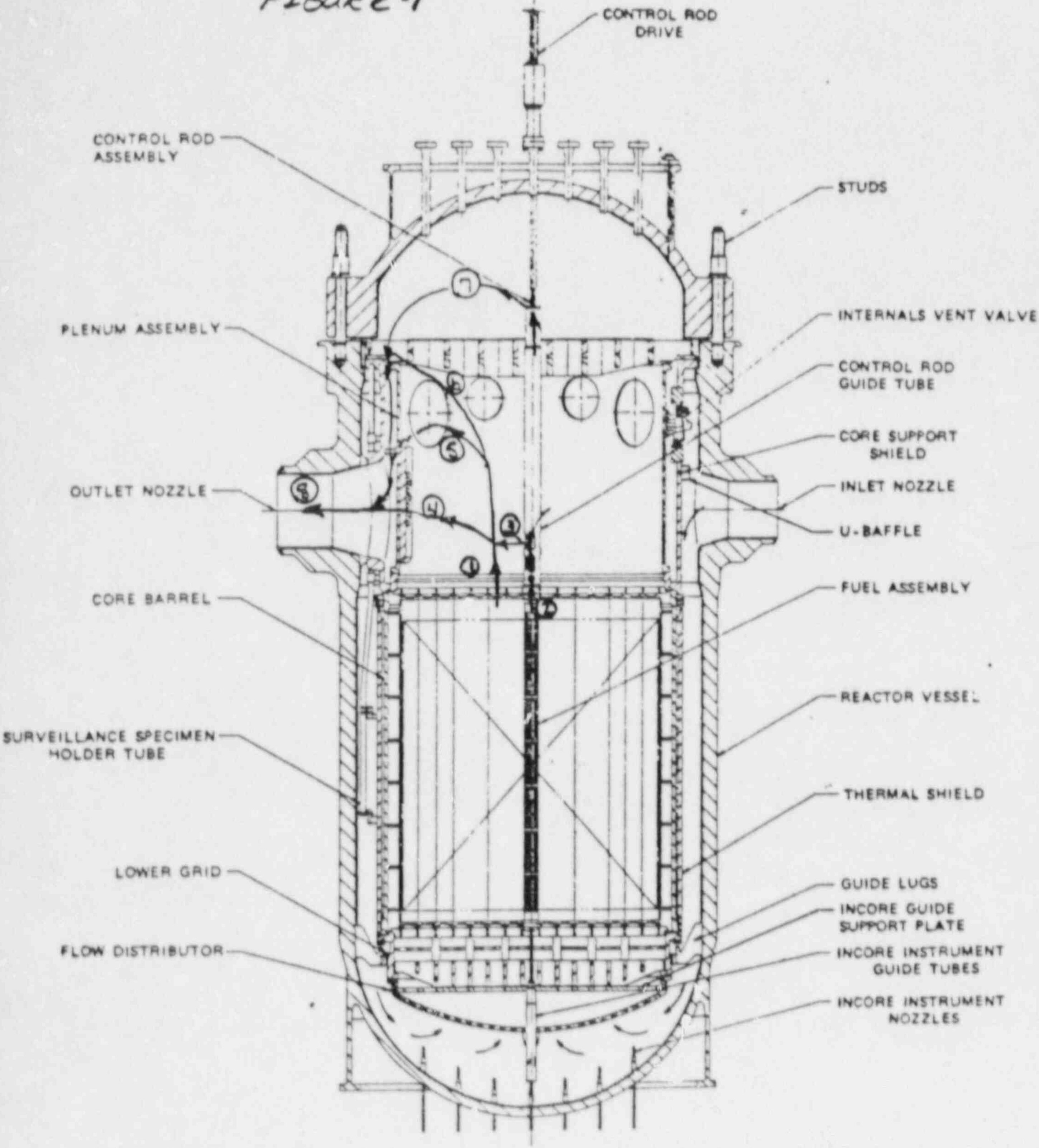
7. N. T. Simms, "RETRAN-02 Comparison of Natural Circulation Flow Rates at B&W 177-FA Plant," Proceedings: Third International RETRAN Conference, EPRI NP-3803-SR, 1985.

8. B&W Calculation No. 32-1150499-00, "Reactor Vessel Head Stress Analysis Inputs," for all 177-FA Owners Group Contracts. [Same as Reference 3]

9. J. P. Holman, "Heat Transfer," McGraw-Hill Book Company, 1963.

10. Babcock & Wilcox Calculation No. 32-1140915-00, "CPC. R.V. CRDM Motions," N55 12 & 13 Contracts.

FIGURE-1



CONTROL ROD DRIVE
CONTROL ROD ASSEMBLY
STUDS
PLENUM ASSEMBLY
INTERNALS VENT VALVE
CONTROL ROD GUIDE TUBE
CORE SUPPORT SHIELD
OUTLET NOZZLE
INLET NOZZLE
U-BAFFLE
CORE BARREL
FUEL ASSEMBLY
REACTOR VESSEL
SURVEILLANCE SPECIMEN HOLDER TUBE
THERMAL SHIELD
LOWER GRID
GUIDE LUGS
INCORE GUIDE SUPPORT PLATE
FLOW DISTRIBUTOR
INCORE INSTRUMENT GUIDE TUBES
INCORE INSTRUMENT NOZZLES

DIAMOND POWER SPECIALTY CORP.

HYDRAULIC PRESSURE P.S.I.G.	LOAD HELD BY:	AVERAGE STRAIN	STUD ELONGATION IN	AVERAGE STRESS PSI	AVERAGE LOAD LBS	RELAXATION FACTOR	EFFECTIVE LENGTH IN	ACTIVE LENGTH IN
2,500	TENSIONER	299	.0150	8925	266860	1.520		50.0
	NUT	197	.0085	5870	175530		43.1	
5,000	TENSIONER	602	.0300	17954	536839	1.283		49.7
	NUT	469	.0200	13991	418333		42.5	
7,500	TENSIONER	892	.0445	26589	795012	1.262		49.8
	NUT	706	.0290	21053	629505		41.0	
10,000	TENSIONER	1164	.0580	34687	1037147	1.236		49.8
	NUT	941	.0385	28056	838895		40.8	
12,500	TENSIONER	1462	.0725	43582	1303116	1.237		49.5
	NUT	1182	.0480	35231	1053408		40.6	
15,000	TENSIONER	1759	.0865	52433	1567749	1.235		49.1
	NUT	1424	.0575	42442	1269035		40.3	
17,500	TENSIONER	2039	.1015	60777	1817235	1.240		49.7
	NUT	1644	.0670	49013	1465505		40.7	
20,000	TENSIONER	2344	.1160	69851	2088550	1.227		49.4
	NUT	1910	.0765	56925	1702070		40.0	

TITLE: STUD LOADING SUMMARY SHEET	DIAMOND ORDER NO. 502701-N
CUSTOMER: B&W FOR CONSUMER	DATE JANUARY 2, 1974
CLASS 1,000 TON TENSIONER	BY G. E. ROTHENBERGER, T.H.H.
SERIAL NO. 2-74 6-1/2" Dia. Stud	DWG. NO. SK-17988-D

REF: SK-17989-P