231-1 (4.4.2.1)

Section 4.4.2.1 stated that "... the grid design has changed from the standard Inconel R-grid design to a Zircaloy design". Besides the different material used for the six grids in the high flux region (Zircaloy), is there any other difference between the new grid and the R-grid or the L-grid? Provide more information (In addition to that in Section 4.2.2.2.4) on the difference between the new and the old grids including valves of the grid pressure loss coefficients. Also provide the OFA inlet and exit loss coefficients.

Response:

The basic grid design for both the Inconel (STD.) and the Zircaloy (OPT) grids is a square pitch egg crate design with six-point rod support. (Two dimples and one spring in each of two directions) The basic construction (straps) and design functions are identical for both designs.

Some differences between the two types of grids are:

1. The OPT grid height is greater than the STD grid height.

2. The grid strap thickness for the OPT grid is greater than the STD grid.

3. The strap joining methods are different in that the STD grids are brazed, the OPT grids are welded.

Numerical values of the above differences are supplied in Table 1-1 of Reference 5.

The loss coefficients for the 17 x 17 optimized fuel assembly, based on the ungridded bundle flow area are:

Bottom nozzle
Non-mixing vane grid K
Mixing vane grid K
Top nozzle

[ ] +a,b,c

231.2 (Pertaining to WCAP-9500 Section 4.4.1.2)
"Provide a comparison of the average fuel centerline temperature for the OFA vs. Byron and Braidwood units."

Response: A comparison between 17x17 standard fuel temperatures and 17x17 optimized fuel temperatures shows that average and centerline temperatures for optimized fuel are nearly the same (within 5°F) as corresponding temperatures for standard fuel in the range of interest for safety analyses (i.e. <15 kw/ft).

231.3 (4.4.2.9) Provide a discussion of the changes in the values of the sensitivity factors S_i introduced by using the WRB-1 correlation instead of the W-3 correlation for DNB heat flux.

Response: See attached table1.

231.4 (4.4.2.9) Provide a list of all the parameters and range of values treated in a statistical manner using the Improved Thermal Design Procedure for the DNBR limit of the OFA.

Response: See attached table1.

231.5 (4.4) Provide discussions of the method used in applying the statistical method to the minimum DNBR's calculated for the nominal and design transient conditions as shown in Table 4.4-1.

Response: The minimun DNBR calculated at nominal conditions are based on the values shown in the "Nominal Values" column in the response to Question 231.4 using the THINC IV code and the WRB-1 DNB correlation.

The minimum DNBR for Design Transients is the limit DNBR used in the safety analyses of the plant is further discussed in Section 4.4.1.1.

231.6 (4.4.1.3) Explain the following differences.

	non-UHI Plants	UHI Plants
From Section 4.4.1.3 Flowrate for effective fuel rod cooling, $10^6 lb_m/hr$	143.5 x .942 = 135.2	143.3 x .925 = 132.6
From Table 4.4-1 heat transfer, $10^6 lb_m/hr$	137.3	134.7

Response: The difference is due to the treatment of nominal bypass and the maximum thermal design bypass as discussed in Section 4.4 and Table 4.1 of Reference 3. For both UHI and non-UHI applications, this difference is the same.

231.7 (4.4.2.2) Describe the basis and magnitude of DNBR rod bow penalty to be applied to the OFA.

Response: The amount of fuel rod bowing to be accounted for in the OFA is described in Section 4.2.3.1 (page 4.2 - 27) resulting in the same DNBR rod bow penalty as the stand-

DNBR penalty for 17x17 standard fuel is contained in <u>Fuel Rod Bow Evaluation</u>, WCAP-8691 (Rev. 1) (Westinghouse Proprietary) and WCAP-8692 (Rev. 1) (Non-Proprietary), July 1979.

231.8 (4.4.2.9.5) Was the transient analysis code uncertainty ($\pm$1% of LOFTRAN) included in the DNRB analyses with the statistical method?

Response: Yes. See also response to Question 231.4.

231.9 (4.4.4 5.4) Provide a discussion of the applicability of the statistical method described in WCAP-8567 to part-loop operation.

Response: The main intent of WCAP-9500 is to serve as a reference licensing document for plants with all loops operating. The one loop out of service operation information is provided in Table 4.4-2 as a matter of historical precedent. The Improved Thermal Design Procedure would be expected to be applicable to DNB analyses for this mode of operation.

231.10 (4.4.2.9.6) For both UHI and non-UHI plants, provide bases for the difference in the thermal design flow and best-estimate loop flow (2.5% for non-UHI plants and 4% for UHI plants).

Response: The difference is not due to the distinction between UHI and non-UHI plants, but due to the fact that non-UHI plant (white pages) employs the more accurate flow measurement system as further discussed in Section 4.4.2.9.6 (white pages).

231.11 (4.4.2.9.6) (4.4.1.3)	Fractions of the thermal design flow that is allotted as bypass flow are inconsistent in Sections 4.4.1.3 and 4.4.2.9.6. Provide the bases for the assumptions and make it consistent.
Response:	This was a typographical error. In Section 4.4.1.3, page 4.4-4 (white), change 7.5 percent to 5.8 percent. In section 4.4.1.3, page 4.4-4 (blue), change 5.8 percent to 7.5 percent.
231.12 (4.4.2.9.1)	Quantify the "...excellent heat transfe ..." and "...the film temperature drop..." in the cladding temperature calculation uncertainty described in Section 4.4.2.9.1.
Response:	The surface heat transfer coefficients are discussed in Section 4.4.2.7.1. In the single-phase region, applying the Dittus-Boelter correlation and representative dimensions and fluid properties, the convective heat transfer coefficient is approximately 5000 BTU/hr-ft^{2}°F. With the onset of nucleate boiling, the Thom's correlation is applicable and the convective heat transfer coefficent is approximately 40,000 BTU/hr-ft^{2}°F.
231.13 (4.4.2.9.5)	Quantify the "...conservatively high values of the nuclear peaking factors..." used in the THINC-IV analysis for DNBR's.
Response:	By conservatively high values of the nuclear peaking factors, it is not meant that a best estimate value over time is used. The radial peaking factor ($F^N_{\Delta H}$) value used still considers and bounds all times in life, the rod insertion positions, as defined in Section 4.4.4.3.1. The axial peaking factor values used are defined in Section 4.4.4.3.2.

231.14 For Ledinegg instability, provide discussions on $(\delta \Delta P/\delta G)$ external <o and $(\delta \Delta P/\delta G)$ inerntal >o. Is it a designed feature or ..ults of tests applicable to Conditions I and II events operational ranges?

Response: $\partial \Delta P/$ $_{external}$ <0 is a generic characteristic of W reactor coolant pumps and $\partial \Delta P/\partial G|_{internal}$ >0 is a generic characteristic of W cores. These values are a result of the W pump and core designs.

Item 231.15 (4.4.4.6) For dynamic stability, justify the statement "an open channel configuration is more stable than the closed channel analysis under the same boundary conditions." Explain the basis for extrapolating from the previous tests (Reference 76, 77 from Section 4.4) to the current OFA design. Provide Reference 72 from Section 4.4.

Response: The method used .o analyze Westinghouse Cores for density wave instabilities was developed for closed channel systems. The statement that "an open channel configuration is more stable than the closed channel analysis under the same boundary conditions" (i.e. channel inlet and outlet conditions) is based on experimental evidence(76).

The tests referred to in Reference 76 show in a relative fashion that closed channel systems are less stable than open channel systems. The tests referred to in Reference 77 are used only as supplementary evidence to support the conclusion that no flow oscillation can occur at PWR Condition I and II thermal hydraulic conditions. The method developed by Ishii was used to assess the OFA design geometry at Condition I and II thermal hydraulic conditions.

Item 231.16 For the distributions shown in Figures 4.4-4 to 4.4-6, indicate the axial step size, amount of axial cross flow (assembly-to-assembly), inlet flow distribution (velocity of mass flux), pressure drop (along the elements and in the grid spacer), mixing coefficient used, and axial void fraction.

Response: The axial step size used was 5.3 inches. The amount of assembly-to-assembly crossflow is minimal as evidenced by the values of $G/\overline{G}$. The inlet flow distribution was uniform as defined in Section 4.4.2.5. The pressure drops can be calculated using the relationships identified in Section 4.4.2.7.2 and the loss coefficients provided in the response to Question 231.1. The mixing coefficient (TDC) used was zero. The core average void fraction was 0.2 percent. The maximum assembly void fraction was less than one percent.

Item 2312.17 How accurate are the results of the 1/7th scale model pressure drops when compared to actual operating plants (p. 4.4-17, para. 2)? How significant is the pressure drop across the upper internals in the vessel (with respect to total core losses) and how accurate are theoretical predictions in this region?

Response: The ability to correctly predict pressure drops is verified before initial criticality in that a reactor coolant flow test is performed to ensure that the proper flow rates have been used in the core thermal and hydraulic analysis as described in Sections 4.4.2.7.2 and 4.4.5.1.

Item 213.18 Since no hot channel allowances are included in the design for quadrant power tilts, what is the worst case (e.g., a dropped for misaligned RCCA) hot channel factor developed and the resulting increase in coolant temperature or decrease in DNBR (p. 4.4-23, para. 2)?

Response: The statically misaligned RCCA and dropped RCCA analysis are described in Section 15.4 (reactivity and Power Distribution Anomalies).

Item 231.19 Verify that the Plant A and Plant B maximum bypass flows are interchanged (p. 4.4-4, para. 4).

Response: See response to question 231.11.

Item 231.20 How is w' determined (p. 4.4-9, eq. 4.4-3)? How does TDC vary with spacing distance (16, 20, 26, and 32 in.)? What is TDC for natural turbulence? How sensitive are transverse coolant temperature differences to TDC? How does boiling during transients affect TDC and, thereby, transverse coolant temperature differences? How is the large difference (0.038 to 0.059) in TDC's physically explained (p. 4.4-10, para. 3)?

Response: The parameter w' is not independently determined. It is calculated internal to the THINC analysis as a function of the fluid thermal gradients between adjacent subchannels as described in Reference 12. TDC, as a function of spacing distance, is given in Figure 4-7 of Reference 13. Reference 13 also states that for the range of Reynolds numbers studied, TDC is not affected by Reynolds number and indicates the range of transverse coolant temperature differences measured in the mixing tests. Two-phase flow effects on TDC are addressed in Section 4.4.2.2.3, indicating that the single-phase value is less than or equal to the two-phase values.

The TDC used in mixing calculations for design purposes is 0.038 while the TDC value of 0.059 is the mean value of the data obtained from tests (see Reference 17). In other words, this is a conservatism in our analysis.

(Note: The range of bulk outlet quality given on the bottom of page 4.4-9 should read "-52.1 to -13.5 percent.")

Item 231.21 If core hydraulic loads are twice fuel assembly weight during pump overspeed transients that produce 20 percent flow over design (p. 4.4-15, para. 1), how great are the hydraulic loads at design flow?

Response: The pump overspeed 20 percent excess flow results in approximately 44 percent greater pressure drop induced hydraulic lift forces (i.e. $(1.2)^2 = 1.44$). The fuel assembly hold down springs are designed to provide a force that exceeds the net fuel assembly lift force at normal operating conditions (hot or cold). At pump overspeed conditions the fuel assembly net lift forces are allowed to exceed the available hold down spring force; however, after the pump overspeed the springs must still meet the normal operating condition criteria.

Item 231.22 The DNBR values for thimble and typical cells for plant safety analyses are taken at 1.82 and 1.85 (Plant A) and 1.47 and 1.49 (Plant B), respectively (p. 4.4-3, para. 1, white and blue). Explain this significant difference and provide the process by which these DNBR values were obtained.

Response: The biggest single distinction is that Plant A utilizes the Integrated Protection System (IPS) while Plant B does not. The IPS is described in Section 4.4.4.3 (white), 4.4.4.3.1 (white), and 4.4.4.3.2 (white). The IPS is a significant state of the art advancement in core protection and was previously reviewed in the RESAR 414 PDA.

The DNBR values for plant safety analyses are chosen considering the results of the accident analysis and the plant operating flexibility requirements. This results in the plant allowance margin between the DNBR's used in safety analysis and the design DNBR for the purposes identified in Section 4.4.1.1.

Item 231.23 If Reference 15 the proper one to quote here (p. 4.4-8, para. 3)?

Response: It should read Reference [5] not [15].

TABLE 1

Parameter	Nominal Value	Range	Uncertainty Equivalent Standard Deviation	Sensitivity (% DNBR/% Parameter) Typical Cell	Thimble Cell
Power	100% Power				
Inlet Temperature	561.6°F (562.5°F)*				
Pressure	2280 psia				
Vessel Flow	387600 GPM (388400 GPM)*				
Effective Flow Fraction (Bypass)	0.94 (0.957)*				
$F_{\Delta H}^{N}$	1.49				
$F_{\Delta H,1}^{E}$	1.0				
THINC IV	-				
Transient Code	-				

+(a,c)

*The IPS Plant is in Parenthesis.