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

MAY 25 1977

Docket No. P-566

MEMORANDUM FOR: Domenic B. Vassallo, Assistant Director for LWRs, DPM
FROM: D. F. Ross, Jr., Assistant Director for Reactor Safety, DSS
SUBJECT: BSAR-205 SER INPUT

Plant Name: BSAR-205
Licensing Stage: CP
Docket No.: P-566
Milestone No.: 24-22
Responsible Branch and Project Manager: LWR-1, T. Cox
Systems Safety Branch Involved: Analysis Branch
Description of Review: SER Input
Requested Completion Date: 1/31/77
Review Status: Complete at Milestone, Outstanding Issues Noted

Enclosed please find the SER input for BSAR-205, Section 4.4.

D.F. Ross, Jr.
D. F. Ross, Jr., Assistant Director
for Reactor Safety
Division of Systems Safety

Enclosure:
SER Input

- cc: S. Hanauer
- R. Heineman
- D. Ross
- T. Cox
- Z. Rosztoczy
- L. Phillips
- W. Hodges
- G. Kelly

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4.4 Thermal and Hydraulic Design

The principal criterion for the thermal-hydraulic design of a reactor is avoidance of thermally induced fuel damage during normal steady-state operation and during anticipated operational occurrences. Babcock & Wilcox uses the following design limits to satisfy this criterion.

1. The fuel rod cladding, fuel pellets, and fuel rod internals must be designed so that the fuel-to-clad gap characteristics ensure that the maximum fuel temperature does not exceed the fuel melting limit at the 112 percent design overpower at any time during core life. The fuel melting temperature is 5090°F at BOL and reduces linearly to 4800°F at EOL (43,000 Mwd/mtu).
2. The minimum allowable departure from nucleate boiling ratio (DNBR) during steady-state operation and anticipated transients is 1.30 with the BAW-2 correlation.
3. Flow stability is required during all steady-state and operational transient conditions.

The thermal and hydraulic design parameters for the reactors are listed in Table 4.4-1. A comparison of these parameters with those of Bellefonte is given in the table. The hydraulic analysis was based on vessel model flow tests for a 205 fuel assembly design. Review of the 205 fuel assembly vessel model flow tests (BAW-10025P)^{1*}, applicable to the design of BSAR-205, will be completed by the final design review stage to confirm the acceptability of the thermal-hydraulic calculations.

*References at end of section.

TABLE 4.4-1
THERMAL AND HYDRAULIC PARAMETERS

	BSAR-205	Bellefonte 1 & 2
Reactor Core Heat Output (Mwt)	3800	3600
System Pressure, nominal (PSIA)	2250	2250
Minimum DNBR at design power	1.83	1.82
Minimum DNBR at design overpower (112%)	1.44	1.4
Total Reactor Coolant Flow (10^6 lb/Hr)	158.63	150.5
Core coolant average velocity (Ft/s)	16.9	16.2
Coolant temperature (F)		
design nominal inlet	569.0	572.3
average rise in core	57.0	56.5
Total heat transfer surface in core (Ft ²)	63,991	63,991
Average heat flux (BTU/Hr/Ft ²)	197,151	186,800
Maximum heat flux (BTU/Hr/Ft ²)	523,083	507,000
Maximum thermal output (KW/Ft)	15.2	14.74
Maximum thermal output at		
overpower (KW/FT)	17.0	16.51
Maximum fuel central temperature (F)		
100% power	4280	3670
112% power	4540	4470

Prevention of DNB for steady-state operation and anticipated transients will assure that the hot spot of the fuel cladding will be maintained at a temperature only slightly greater than that of the coolant, which will not lead to a loss of cladding integrity. B&W has calculated that at the beginning of core life at 100% power, with a linear heat generation rate of 15.2 kilowatts per foot, the fuel centerline temperature will be 4280 degrees Fahrenheit. The peak power density that would occur for a reactor trip at 112 percent maximum over power trip is 17.0 kilowatts per foot. At a linear heat generation rate of 17.0 kilowatts per foot, B&W calculated a centerline temperature of approximately 4540 degrees Fahrenheit at BOL, thus indicating no fuel melting.

The margin to DNB at any point in the core is expressed in terms of the departure from nucleate boiling ratio (DNBR). The DNBR is defined as the ratio of the heat flux required to produce departure from nucleate boiling at the calculated local coolant conditions to the actual local heat flux. The DNB correlation used for the design of this core is the BAW-2 correlation. The BAW-2 correlation was derived from data on 6 ft. heaters which simulated the rod diameter and spacing of 15x15 fuel assemblies. B&W has committed to tests on full length heaters of the 17x17 geometry with uniform and non-uniform axial heat flux. Results from these tests and the statistical data analysis must confirm the thermal-hydraulic design prior to approval of the final design report for BSAR-205 plants.

Another parameter that must be accounted for in the thermal-hydraulic design of the core is the rod-to-rod bowing within fuel assemblies. During the Oconee 1 refueling, six fuel assemblies were examined visually and

dimensionally. The water channel and line scan measurements indicated a maximum rod bow of approximately 30 mils. B&W states that the observed rod bow is accommodated within the current design and that B&W is pursuing a program to demonstrate this. B&W plans to develop bow correlations and predictive techniques to analyze the data and the predicted bow from a thermal-hydraulic standpoint in the near future and has committed to provide results of the localized effect on DNB. Operating restrictions can be applied at the final design review stage if rod-to-rod bowing proves to reduce the margin to DNB for the 17x17 fuel assembly design.

Currently Rancho Seco is the only operating B&W plant with a rod bow penalty. The penalty is expected to be reduced or removed when sufficient rod bow data become available.

Parallel channel flow stability analysis is performed with the HYTRAN^{2*} computer code, in conjunction with the CHATA³ code. Topical reports CHATA and HYTRAN are presently under review by the Staff. Any limitations to the thermal-hydraulic design resulting from the review of HYTRAN (BAW-10109) and CHATA (BAW-10110) will be compensated for by appropriate operating restrictions; however, no operating restrictions are anticipated.

In Section 4.4.5.9.1 BSAR-205 commits to provide a Loose Parts Monitoring System (LPMS) to detect the presence of loose parts in the reactor core and primary coolant system. Additional information on the loose parts monitoring system is to be supplied in the license applications for individual BSAR-205 plants. The staff considers the BSAR-205 commitment as an interface requirement and will review the acceptability of the LPMS on a plant-by-plant basis. We will require that the LPMS be appropriately

identified as an interface on the BSAR-205 docket.

The 0.99 flow distribution factor proposed by B&W was justified based on statistical analysis of the data from the Vessel Model Flow Test^{1*} for the 205 assembly plants. Subject to our acceptance of the 205 assembly model flow tests now under review by the staff, we believe the justification presented in BSAR-205 provides reasonable assurance that the minimum flow factor is 0.99. The staff finds the 0.99 flow factor acceptable with the stated proviso. The operating license applications for BSAR-205 plants should include as-built drawings of the core flow distributor plate for comparison to the plate dimensions to be included in the Vessel Model Flow Test topical report.^{1*}

Part of the protective mechanism to prevent DNB in the core is provided by the Reactor Protection System's calculating module. The design and implementation of this protective software used to guard against DNB must be reviewed and approved by the Staff prior to approval of the final report for BSAR-205 plants. A topical report^{4*} on the Reactor Protection System (RPS) has been submitted to the staff for review.

Conclusions

On the basis of our review we conclude that the thermal-hydraulic design of the core is acceptable for Preliminary Design Approval of the reference system. However, we will require that several items be resolved prior to the Final

Design Approval; these are:

1. Development of data with full length heaters for the BAW-2 correlation. This data base should include both uniform and non-uniform axial heat flux tests on full length heaters with the 17x17 rod diameter and

spacing. If necessary, the BAW-2 correlation should be modified to agree with the data.

2. Statistical analysis of the DNB data to verify the minimum DNBR limit for compliance with the 95/95 design criterion.
3. Review of the flow model tests for the 205 fuel assembly configuration (BAW-10025P)¹.
4. Determination of extent and effect of rod-to-rod bowing.
5. Review and approval of the HYTRAN flow stability computer code (BAW-10109)² and the CHATA code (BAW-10110)³.
6. Review and approval of RPS calculating module (BAW-10085P, Rev. 2).

References

1. R. M. Gribble, "Reactor Vessel Model Flow Tests for 205-Fuel Assembly Core," BAW-10025P, Babcock and Wilcox Company, (August 1976).
2. H. S. Kao, W. R. Cardwell, C.D. Morgan, "Hydraulic Transient Code for Investigating Channel Flow Stability (HYTRAN)," BAW-10109, Babcock and Wilcox Company, (January 1976).
3. J. M. Alcoren, R. H. Wilson, "Core Hydraulics and Thermal Analysis (CHATA)," BAW-10110, Babcock and Wilcox Company, (January 1976).
4. D. B. Fairbrother, D. R. Vincent, L. M. Lesnaik, E. G. Orgera, "Reactor Protection System - Revisions 2," BAW-10085P, Rev. 2., (Proprietary version), Babcock and Wilcox Company, January 1977.

ACRS Generic Items - Analysis Branch

Group I - Resolved Generic Issues

Item 9 - Section 3.12.20 of BSAR-205 commits to a comprehensive test program in compliance with regulatory guide 1.20 - Vibration Measurements on Reactor Intervals (revision 2, May 1976).

Group II - Resolution Pending

Item 5 - Loose Parts Monitoring

See Section 4.4

Group IIA - Resolution Pending - Items since December 18, 1972

Item 7 - Steam Generator Tube Leakage

System design to preclude or mitigate the consequences of steam generator tube failure is considered in the applicant's scope and will be reviewed when the applicant references BSAR-205.