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

OCT 30 1984

MEMORANDUM FOR: T. M. Novak, Assistant Director
for Licensing, DL

FROM: L. S. Rubenstein, Assistant Director
for Core and Plant Systems, DSI

SUBJECT: REV. 1 TO BEAVER VALLEY UNIT 2 SAFETY EVALUATION REPORT

Plant Name:	Beaver Valley Unit 2
Docket Number:	50-412
Licensing Stage:	Operating License
Responsible Branch:	Licensing Branch No. 3
Project Manager:	L. Lazo
DSI Review Branch:	Core Performance Branch
Review Status:	Closing 2 Open Issues, and Updating a 3rd in the Thermal-Hydraulics Area

The Core Performance Branch has previously submitted the final SER sections for the Beaver Valley Unit 2 FSAR. The submittal was dated July 16, 1984. Since then the applicant responded satisfactorily to two of our four open issues in the thermal-hydraulics area (Section 4.4).

Loose part detection (Section 4.4.5) remains as an open issue; the applicant must provide a detailed item by item response to the Regulatory Guide 1.133 guidelines (see revised Section 4.4.5). Also, the applicant has stated that the loose parts monitoring system is not seismically qualified for seismic events up to the OBE as specified in Regulatory Guide 1.133. The applicant should modify the system as necessary to assure its operability following an OBE.

The staff is still awaiting information on inadequate core cooling instrumentation (Item II.F.2 of NUREG-0737). The updated sections of the thermal-

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hydraulic SER are enclosed (Enclosure 1).

The two open issues on rod bow, and flow measurement capability are now closed.

Our SALP is enclosed as Enclosure 2.

Original signed by
L. S. Rubenstein

L. S. Rubenstein, Assistant Director
for Core and Plant Systems, DSI

Enclosures:
As stated

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ENCLOSURE 1

BEAVER VALLEY UNIT 2 REVISED SER FOR SECTION 4.4

4.4 Thermal-Hydraulic Design

4.4.1 Performance and Safety Criteria

The performance and safety criteria for the Beaver Valley Unit 2, as stated in Section 4.4.1 of the FSAR, are:

(1) "Fuel damage (defined as penetration of the fission product barrier, i.e., the fuel rod cladding) is not expected during normal operation and operational transients (ANS Condition I) or any transients arising from faults of moderate frequency (ANS Condition II). It is not possible, however, to preclude a very small number of rod failures resulting in the release of fission products. The chemical and volume control system is designed to remove the fission products from the reactor coolant, keeping the reactor coolant activity within plant design bases limits."

(2) "The reactor can be brought to a safe state following a Condition III event with only a small fraction of fuel rods damaged, as defined previously although sufficient fuel damage might occur to preclude resumption of operation without considerable outage time."

(3) "The reactor can be brought to a safe state and the core can be kept subcritical with acceptable heat transfer geometry following transients arising from Condition IV events."

4.4.2 Design Bases

The performance and safety criteria listed above are implemented through the following design bases.

4.4.2.1 Departure from Nucleate Boiling

The margin of departure from nucleate boiling 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 heat flux.

The thermal-hydraulic design basis, as stated in Section 4.4.1(1) of the Beaver Valley Unit 2 FSAR, for the prevention of departure from nucleate boiling is as follows:

"There will be at least a 95-percent probability that departure from nucleate boiling (DNB) will not occur on the limiting fuel rods during normal operation and operational transients, or during transient conditions arising from faults of moderate frequency (ANS Condition I and II events), at a 95 percent confidence level."

4.4.2.2 Fuel Temperature

The fuel temperature design basis given in Section 4.4.1(2) is:

"During modes of operation associated with ANS Condition I and II events, there is at least a 95 percent probability at the 95 percent confidence level that the peak kW/ft fuel rods will not exceed the UO_2 melting temperature." The maximum fuel temperature shall be less than the melting temperature of UO_2.
This design basis is evaluated in this Safety Evaluation Report in Section 4.2, "Fuel System Design."

"A minimum of 95.5 percent of the thermal flow rate passes through the fuel rod region of the core and is effective for fuel rod cooling."

4.4.2.3 Hydrodynamic Stability

The hydrodynamic stability design basis given in Section 4.4.1(4) is as follows.

"Modes of operation associated with ANS Condition I and II events shall not lead to hydrodynamic instability."

4.4.3 Thermal-Hydraulic Design Methodology

4.4.3.1 Departure from Nucleate Boiling

The thermal-hydraulic design analysis was performed using the W-3 Critical Heat Flux (CHF) correlation in conjunction with the THINC-IV computer program. THINC-IV is an open channel computer code which determines the coolant density, mass velocity, enthalpy, vapor void, static pressure, and DNBR distribution along parallel flow channels within a reactor core.

The W-3 correlation was developed from data obtained from experiments conducted with fluid flowing inside single heated tubes. As test procedures progressed to the use of rod bundles instead of tubes, the correlation was modified to include the effects of "R" and "L" mixing vane grids and axially nonuniform power distributions.

The applicant has proposed this minimum departure from nucleate boiling of 1.30 to ensure that there is a 95 percent probability at a 95 percent confidence that critical heat flux will not occur on the limiting fuel rod. The use of the W-3 CHF correlation with a minimum DNBR of 1.30 has been previously approved by the staff.

A correlation factor was developed to adopt the W-3 correlation to 17x17 fuel assemblies with top split mixing vane grids (R grid) such as those

to be used for Beaver Valley Unit 2. This correlation factor, termed the "modified spacer factor," was developed as a multiplier on the W-3 correlation. A description of the 17x17 fuel assembly test program and a summary of the results are described in the NRC approved WCAP-8298-P-A and WCAP-8299-A. As a result of the test program, a 0.88 multiplier is used as part of the 17x17 modified spacer factor. However, a multiplier of 0.856 has been conservatively applied for all DNB analyses. The test results indicated that a reactor core using this geometry may operate with a minimum DNBR of 1.28 and satisfy the design criterion. However, a minimum DNBR of 1.30 is conservatively used for this plant.

A description of the THINC-IV computer code is given in WCAP-7956, "THINC-IV An Improved Program For Thermal-Hydraulic Analysis of Rod Bundle Cores." The design application of the THINC-IV program is given in detail in WCAP-8054, "Application of the THINC-IV Program to PWR Design." Both WCAP-7956 and WCAP-8054 have been reviewed and approved by the staff.

The staff has previously reviewed under a different docket, a November 2, 1977 letter from C. Eicheldinger (Westinghouse) to J. Stolz (NRC) which described THINC-IV analyses using a cosine upper plenum radial pressure gradient with a maximum value of 5 psi at the core center and 0 psi at the periphery. The results of these analyses showed that the effects of a core exit pressure distribution on the minimum DNBR are negligible. The staff conducted a similar sensitivity study using COBRA-IV. Our results also showed that the effects are small (NUREG-0847). Based on these analyses, the staff concludes that the use of a uniform exit pressure gradient in the Beaver Valley Unit 2 thermal-hydraulic design is acceptable.

Based on our findings that the CHF correlation and the thermal-hydraulic computer code used by the applicant have been previously approved by the staff, and the use of a uniform core exit pressure gradient has been adequately justified, the staff concludes that the DNB design methodology used in the design of Beaver Valley Unit 2 is acceptable.

4.4.3.2 Core Flow

The core flow design basis requires that the minimum flow which will pass through the fuel rod region and be effective for fuel rod cooling is 95.5 percent of the primary coolant flow rate. The remainder of the flow, called bypass flow, will be ineffective for cooling since it will take the following bypass paths;

(1) flow through the spray nozzles into the upper head for head cooling purposes;

(2) flow entering into the rod cluster control rod guide thimbles to cool the control rods;

(3) leakage flow from the vessel inlet nozzle directly to the vessel outlet nozzle through the gap between the vessel and the barrel;

(4) flow between the baffle and barrel; and

(5) flow in the gaps between the fuel assemblies on the core periphery and the adjacent baffle wall.

The amount of bypass flow is determined by a series of hydraulic resistance calculations on the core and vessel internals and verified by model flow tests. Since the amount of bypass flow is consistent with approved plants of similar design, the staff concludes that the core bypass flow used in the design analysis, 4.5 percent, is acceptable.

4.4.3.3 Hydrodynamic Instability

For steady-state, two-phase heated flow in parallel channels, the potential for hydrodynamic instability exists. Two specific types of flow instabilities are considered for Westinghouse PWR operation. These are the Ledinegg, or flow excursion type of static instability, and the density wave type of dynamic instability.

The applicant stated that the core was stable because Westinghouse reactors will not experience any Ledinegg instability over Condition I and II operational ranges. Open channel configurations, which are a feature of Westinghouse PWRs, are more stable than closed channel configurations. This was shown by flow stability tests which were conducted at pressures up to 2200 psia. The results showed that for flow and power levels typical of reactor conditions, no flow oscillations could be induced above 1200 psia.

Also, a method developed by Ishii (Saha, et al, 1976) for evaluating density wave stability in parallel closed channel systems was used to assess the stability of typical Westinghouse reactor designs. The results indicate that a large margin to density wave instaiblity exists. Finally, data from numerous rod bundle tests which were performed over wide ranges of operational conditions show no evidence of premature DNB or of inconsistent data which might be indicative of flow instabilities in the rod bundles.

The staff concludes that past operating experience, flow stability experience, and the inherent thermal-hydraulic design of Westinghouse PWRs serve as a basis for issuance of an operating license.

4.4.4 Operating Abnormalities

4.4.4.1 Fuel Rod Bowing

A significant parameter which affects the thermal-hydraulic design of the core is rod-to-rod bowing within fuel assemblies. The Westinghouse methods for predicting the effects of rod bow on DNB, WCAP-8691, Revision 1, "Fuel Rod Bow Evaluation," have been approved by the staff.

For plants designed by Westinghouse, the staff has approved the generic margins given in Table 4.4-1, which may be used to offset the reduction

in DNBR due to rod bowing.

Plant-specific margins which could be available are:

(1) the Technical Specification minimum flow rate is greater than the design flow rate;

(2) the Technical Specification maximum T_{ave} is less than the design T_{ave}; and

(3) the trip setpoints are more limiting than the thermal-hydraulic analysis indicates.

In a letter dated July 12, 1984 the applicant responded to our concerns by stating that Beaver Valley Unit 2 maintained 9.1 percent margin to accommodate full and low flow DNBR penalties. This is consistent with WCAP-8691 which is approved by the staff and thus, is acceptable. The applicant should insert into the Basis of the Technical Specifications any of the generic or plant-specific margins that are used to offset the reduction in DNBR due to rod bowing.

4.4.4.2 Crud Deposition

Crud deposits in the core and an associated change in core pressure drop and flow have been observed in some PWRs, not of Westinghouse design. The staff requested that the applicant provide a description of the procedures to detect flow degradation as a result of crud buildup. The applicant responded that except for steam generator tube plugging, there have been no reports of a significant flow reduction in a relatively short period of time at any Westinghouse plant.

The staff will ensure that the Beaver Valley Unit 2 Technical Specifications contain the requirement that the actual reactor coolant system (RCS) flow rate be verified to be greater than or equal to the minimum

design flow rate plus uncertainties at least once every 12 hours. In addition, the staff will ensure that the applicant performs a channel calibration at least once every 18 months.

4.4.5 Loose Parts Monitoring System

The applicant has provided a description of the Loose Parts Monitoring System (LPMS) which will be used by Beaver Valley Unit 2. The design will consist of ten active instrumentation channels, each comprising a piezoelectric accelerometer (sensor) and signal conditioning equipment. Sensors are fastened mechanically to the reactor coolant system (RCS) at each of the following potential loose parts collection regions:

1. Reactor pressure vessel-upper head region.

2. Reactor pressure vessel-lower head region.

3. Each steam generator-reactor coolant inlet region.

The system will be capable of detecting a metallic loose part that weighs from 0.25 to .30 pounds impacting within 3 feet of a sensor and having a kinetic energy of 0.5 foot-pounds on the inside surface of the RCS pressure boundary.

In response to our concern the applicant stated that the system is not seismically qualified. The applicant should modify the system as necessary to assure its operability following an OBE. Also, the applicant should provide an itemized comparison on conformance of Beaver Valley Unit 2 to Regulatory Guide 1.133.

4.4.6 Thermal-Hydraulic Comparison

The thermal-hydraulic design parameters for the Beaver Valley Unit 2 are listed in Table 4.4.2 and compared to values for the Virgil C. Summer plant.

Beaver Valley Unit 2 was designed to operate at a thermal power comparable with the V. C. Summer Plant. The W-3 CHF correlation and THINC-IV computer program were used in the design of both plants.

The thermal-hydraulic design of the Virgil C. Summer Plant has been previously reviewed and approved by the staff.

The comparability of Beaver Valley Unit 2 with the Virgil C. Summer plant supports the conclusion that the Beaver Valley Unit 2 thermal-hydraulic design is acceptable.

4.4.7 N-1 Loop Operation

N-1 loop operation refers to operation of the reactor with one of the reactor's coolant loops out of service. Thus in the case of Beaver Valley Unit 2 only two coolant loops are available to supply coolant to the reactor core. The applicant has not requested N-1 loop operation.

If the applicant wishes to exercise the option to operate in the N-1 mode, he will be required to provide core thermal-hydraulic analyses taking into account the effect of partial loop operation on core inlet flow distribution, MDNBR, and the effect of N-1 loop operation on postulated transients and accidents. If the applicant chooses to not use the N-1 loop operation then the staff will require that the Technical Specifications include an appropriate provision to ensure that this type of operation is prohibited.

4.4.8 ICC Instrumentation

We have reviewed the applicant's submittal on the instrumentation for the indication of inadequate core cooling and found it insufficient with respect to documentation. Therefore, the staff will require the applicant to provide the itemized documentation of a complete ICCI system, i.e., to respond to the items of NUREG-0737 pp. II.F.2-3,4,5 and 6 and pp. B-1,2,3 and 4. The information should be supplied on a schedule which will permit completion of our review prior to fuel loading.

4.4.9 Conclusion

The thermal-hydraulic design of the Beaver Valley Unit 2 has been reviewed. The acceptance criteria used as the basis for our evaluation are set forth in the Standard Review Plan (SRP), NUREG-0800 in 4.4, Section II, "Thermal and Hydraulic Design Acceptance Criteria." The scope of the review included the design criteria, core design, and the steady-state analysis of the core thermal-hydraulic performance. The review concentrated on the differences between the proposed core design and those designs which have been previously reviewed and found acceptable by the staff. It was found that all such differences were acceptable. The applicant's thermal-hydraulic design analyses were performed using analytical methods and correlations that have been previously reviewed by the staff and found acceptable.

The staff concludes that the initial core has been designed with appropriate margin to ensure that acceptable fuel design limits are not exceeded during steady-state operation and anticipated operational occurrences. The thermal-hydraulic design of the initial core, therefore, meets the requirements of General Design Criterion 10, 10 CFR Part 50, and is acceptable. This conclusion is based on the applicant's analyses of the core thermal-hydraulic performance which were reviewed by the staff and found to be acceptable. However, prior to final design approval and issuance of an operating license, the staff will require the applicant to provide an acceptable itemized comparison on Beaver Valley Unit 2 compliance with Regulatory Guide 1.133 as described in Section 4.4.5 of this SER. This includes the operability of the system following an OBE.

Also, the applicant should provide information on Beaver Valley Unit 2 compliance with the staff's II.F.2 requirements as shown in Section 4.4.8.

ENCLOSURE 2

INFORMATION FOR A SALP EVALUATION - BEAVER VALLEY UNIT 2
SER REVISION

1. Management Involvement.
   The submittal was adequate but required additional information for the completion of the review.

   Rating: Category 2.

2. Approach to Resolution of Technical Issues.
   The licensee showed an understanding of most of the technical issues raised during the review; however, response to our concerns was slow.

   Rating: Category 2

3. Responsiveness to NRC Initiatives.
   It took a long time to get any response to the technical issues raised, and some still need additional information like the loose parts issue and others the applicant has not even responded to yet.

   Rating: Category 3

Table 4.4.1 Generic Margins

Margin	% reduction in rod bow penalties
The use of a design minimum DNBR of 1.30 instead of the 95/95 DNBR limit of 1.28.	1.6
A reduction in fuel rod pitch for the hot channel analysis.	1.7
The use of a Thermal Diffusion Coefficient (TDC) of 0.038 instead of a TDC of 0.051.	1.2
The addition of an extra grid in the design of the Westinghouse 17x17 fuel assembly relative to the 15x15 fuel design.	2.9
The use of a 0.86 multiplier on the modified spacer factor (F's) of the W-3 correlation instead of a 0.88 multiplier.	1.7
Maximum Generic Margin which may be claimed.	9.1

Table 4.4.2 Reactor Design Comparison

	Beaver Valley Unit 2	V. C. Summer
I. Performance Characteristics:		
Reactor core heat output (MWT)	2,652	2,775
System pressure, psia		
Minimum departure from nucleate	2,250	2,250
boiling ratio at nominal design conditions		
Typical flow channel	2.26	1.98
Thimble (cold wall) flow channel	1.83	1.68
Minimum DNBR for design transients	1.30	1.30
Critical heat flux correlation	"R" (W-3 with modified space factor)	"R" (W-3 with modified space factor)
II. Coolant flow:		
Total flow rate (10^6 lb/hr)	100.8	109.6
Effective flow rate for heat transfer (10^6 lb/hr)	96.3	102.6
Average velocity along fuel rods (ft/s)	14.4	15.6
Effective core flow area (ft^2)	41.6	41.6
III. Coolant temperature, °F		
Nominal reactor inlet	542.5	556.0
Average rise in core	70.3	66.6
Pressure drop across core (psi)	21.4±2.1	23.2±2.3
IV. Heat transfer 100 percent power		
Active heat transfer surface area (ft^2)	48,600	48,600
Average heat flux (Btu/hr-ft^2)	181,400	189,800
Maximum heat flux (Btu/hr-ft^2)	420,900	440,400
Average linear heat rate (kW/ft)	5.20	5.44
Maximum thermal output (kW/ft)	12.1*	12.6

* This limit is associated with the value of $F_q = 2.32$