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

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

SUBJECT: BEAVER VALLEY UNIT 2 DRAFT 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:	Section 4.2 later, one confirmatory issue in Section 15.4.3, and four open issues in Section 4.4

The Core Performance Branch has prepared the enclosed DSER input for Sections 4.3, 4.4, 15.4.1, 15.4.2, 15.4.3, 15.4.7, and 15.4.8 of Beaver Valley Unit 2.

Due to a shortage of staff resources, Section 4.2 of the Beaver Valley Unit 2 FSAR is being reviewed by a contractor (PNL) and some schedule delays have been unavoidable. For that section of the FSAR, DSER input will be provided on a delayed schedule (March 4, 1984), but final SER input will be provided on the original schedule (July 27, 1984).

The confirmatory issue and open issues are identified as follows:

Confirmatory Issue:

Confirmation that the analysis of the dropped control rod event meets DNB limits (see Section 15.4.3, second paragraph).

Open Issues:

1. Provide a commitment to supply a report describing the loose parts detection program and implementation of the system (see Section 4.4.5).

2. Supply the information for Item II.F.2 of NUREG-0737 (see Section 4.4.8).

3. Provide a description of flow measurement capability and procedure (see Section 4.4.4.2).

Contacts: H. Richings, DSI:CPB ~~X-29410~~ A. Gill, DSI:CPB ~~X-27091~~ M. Dunenfeld, DSI:CPB ~~X-28097~~

OFFICE						
SURNAME						
DATE						

4. Address concerns regarding the effect of rod bow on DNB (see Section 4.4.4.1).

Original signed by:
L. S. Rubenstein, Assistant Director
for Core and Plant Systems, DSI

Enclosure:
As stated

cc: R. Mattson
D. Eisenhut
G. Knighton
L. Lazo

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DATE	01/25/84	01/ /84	01/26/84	01/ /84	02/9/84	02/10/84	

4.3 Nuclear Design

The Beaver Valley Unit 2 power plant has a reactor core consisting of 157 fuel assemblies of the Westinghouse 17X17 design. The core has a design heat output of 2652 thermal Megawatts and is similar in most respects to the Virgil C. Summer reactor and other recent Westinghouse 3 loop reactors. We have reviewed the nuclear design of the Beaver Valley Unit 2 reactor. Our review was conducted in accordance with the guidelines provided by the Standard Review Plan, Section 4.3, and was based on information contained in the Final Safety Analysis Report, amendments thereto, and the referenced Topical Reports.

4.3.1 Design Bases

Design bases are presented which comply with the applicable General Design Criteria. Acceptable fuel design limits are specified (GDC 10), a negative prompt feedback coefficient is specified (GDC 11) and tendency toward divergent operation (power oscillation) is not permitted (GDC 12). Design bases are presented which require a control and monitoring system (GDC 13) which automatically initiates a rapid reactivity insertion to prevent exceeding fuel design limits in normal operation or anticipated transients (GDC 20). The control system is required to be designed so that a single malfunction or single operator error will cause no violation of fuel design limits (GDC 25). A reactor coolant boration system is provided which is capable of bringing the reactor to cold shutdown conditions (GDC 26) and the control system is required to control reactivity changes during accident conditions when combined with the engineered safety features (GDC 27). Reactivity accident conditions are required to be limited so that no damage to the reactor coolant system boundary occurs (GDC 28).

We find the design bases presented in the FSAR to be acceptable.

4.3.2 Design Description

The FSAR contains the description of the first cycle fuel loading which consists of three different enrichments and has a first cycle length of approximately one and a half to two years. The enrichment distribution, burnable poison distribution, soluble poison concentration and higher

isotope (actinide) content as a function of core exposure are presented. Values presented for the delayed neutron fraction and prompt neutron life-time at beginning and end of cycle are consistent with those normally used and are acceptable.

Power Distribution

The design bases affecting power distribution are:

- The generic design peaking factor for the reactor is 2.32. However, at present, the peaking factor in the core can not be greater than 2.19 during normal operation of full power in order to meet the initial conditions assumed in the loss of coolant accident analysis.

- Under normal conditions (including maximum overpower) the peak fuel power will not produce fuel centerline melting.

- The core will not operate during normal operation or anticipated operational occurrences, with a power distribution that will cause the departure from nucleate boiling ratio to fall below 1.3 (W-3 correlation with modified spacer factor).

The first part of the following discussion assumes the Westinghouse generic design total peaking factor, F_Q, of 2.32 provides the (only) required limiting peak power density. This is, at present, the only value of F_Q discussed in Section 4.3 of the FSAR, and the methodology and calculations and surveillance used to demonstrate limiting conditions of operation are appropriate only to that value of 2.32 (for first cycle). The F_Q limit of 2.19 will be discussed following that for 2.32.

The 2.32 F_Q peaking factor is determined and maintained via calculations of extremes of allowed transient power distributions and periodically measured radial power distributions and radial peaking factors F_{xy} and $F_{\Delta H}$. These also provide maximum initial conditions for events described in

Section 15 which assure that peak full power does not cause center line fuel melting or result in departure from nucleate boiling during anticipated operational occurrences.

The applicant has described the manner in which the core will be operated and power distribution monitored so as to assure that these limits are met. The core will be operated in the Constant Axial Offset Control (CAOC) mode which has been shown to result in peaking factors less than 2.32 for both constant power and load following operation. The applicant has elected to use an improved load follow package, developed by Westinghouse, in Beaver Valley Unit 2.

CAOC is described in WCAP-8385 (Proprietary) and WCAP-8403 (non-Proprietary), "Power Distribution Control and Load Following Procedures." This report contains methodology for operation with and without part length control rods. The former mode allows better return to power capability than the latter. Use of part length rods has been withdrawn from Westinghouse reactors. The improved load follow strategy provides a return to power capability during operation without part length rods comparable to the level previously obtainable from operation with part length rods.

The improved load follow strategy involves a redesignated control rod bank and modified overlap that allows greater reactivity insertion than the former design bank within the constraints of a widened, asymmetric CAOC band. The control bank has been changed from eight to four rods. The four rods removed from the control bank have been reassigned as a shutdown bank, thus maintaining shutdown margins. (There are also an extra eight rods assigned to shutdown banks, compared to other Westinghouse three loop reactors.) The CAOC band has been changed from ±5 to +3, 12, ΔI (delta flux difference). The greater inserted reactivity is available for return to power capability upon control rod withdrawal. Another element in the load follow strategy is the use of moderator temperature reductions to augment return to power capability. The temperature reduction adds reactivity during rapid return to power through the inherently negative moderator temperature coefficient.

The analysis used to calculate the maximum peaking factor which can occur using the improved strategy expands the set in the CAOC topical report to 18 calculational cases. However, with the reassigned control bank, maneuvers resulting in greater control rod insertion for a longer duration become operationally practical but tend to become slightly more limiting in terms of total peaking factors. Therefore, simulated load follow maneuvers which return ΔI to the target value (and thereby reduce control rod insertion) have been replaced by load follow strategies which maintain the deeper rod insertion. As a result of our evaluation, we agree with Westinghouse's conclusion that substitution of these more conservative cases will maintain the limiting nature of the 18 case load following analysis.

The analysis performed by Westinghouse indicated that the peaking factor limit could not be met at BOL of Cycle 1 due to the wide ΔI band. This resulted in limiting the width of the band for the first 20% of the cycle typically, and until 3,000 MWD/MTU burnup for Beaver Valley Unit 2 to the value of $\pm 5\% \Delta I$. This $\pm 5\% \Delta I$ is the value previously justified by the CAOC analysis. These features will be incorporated in the Beaver Valley Unit 2 Technical Specifications.

We conclude, for the reasons stated above, that the improved load follow package will continue to prevent the 2.32 peaking factor limit from being exceeded in normal operation of the power plant, and therefore is acceptable.

Two types of instrumentation systems are normally provided to monitor core power distribution. Excore detectors with two axial sections are used to monitor core power, axial offset and azimuthal tilt for the 2.32 F_Q limit, and movable incore detectors permit detailed power distributions to be measured. These systems are used in operating reactors supplied by Westinghouse and we find their use acceptable for Beaver Valley Unit 2 when a 2.32 limit is the minimum requirement (or possibly lower when cycle specific 18 case analyses so indicate).

The 2.32 peaking factor is an acceptable limit for all events considered in Section 15 with the exception of LOCA. For that event the analysis for Beaver Valley Unit 2 as described in Section 15.6, uses a 2.19 value to maintain a clad temperature of less than 2200°F. Since this is less than the generic 2.32 value, some additional or altered means of power distribution analysis or surveillance would have to be used to assure compliance with the 2.19 limit. The generic 2.32 value is essentially never reached in reactor-cycle specific normal operation and lower limiting values can usually be determined and used in conjunction with cycle specific 18 case analyses or with improved surveillance.

In response to questions in this area Beaver Valley has indicated that it intends to provide an improved surveillance system, the Axial Power Distribution Monitoring System (APDMS). The APDMS system chosen by Beaver Valley Unit 2 uses incore moveable detectors. (This is different than the excore detector system chosen by Shearon Harris.) This incore system has been described in a Westinghouse Topical Report, WCAP-8589. This report and the APDMS system has been reviewed and approved by the NRC staff and has been successfylly employed for a number of years by several utilities, including Beaver Valley Unit 1. Uncertainty analysis and Technical Specifications involved in using this system exist and have been approved. Thus the use of this incore APDMS system is fully acceptable. The required F_q value can be maintained.

Reactivity Coefficients

The reactivity coefficients are expressions of the effect on core reactivity of changes in such core conditions as power, fuel and moderator temperature, moderator density, and boron concentration. These coefficients vary with fuel burnup and power level. The applicant has presented values of the coefficients in the FSAR and has evaluated the uncertainties of these values. We have reviewed the calculated values of reactivity coefficients and have concluded that they adequately represent the full range of expected values. We have reviewed the reactivity coefficients used in the transient and accident analyses and conclude that they conservatively bound the expected values, including uncertainties. Further, moderator and power Doppler coefficients along with boron worth are measured as part of the startup physics testing to assure that actual values are within those used in these analyses.

Control

To allow for changes in reactivity due to reactor heatup, load following, and fuel burnup with consequent fission product buildup, a significant amount of excess reactivity is built into the core. The excess reactivity is controlled by a combination of full length control rods and soluble boron. Soluble boron is used to control changes due to:

- Moderator density and temperature changes from ambient to operating temperatures.
- Equilibrium xenon and samarium buildup.
- Fuel depletion and fission product buildup - that portion not controlled by lumped burnable poison.
- Transient xenon resulting from load following.

Control rods are used to control reactivity change due to:

- Moderator reactivity changes from hot zero to full power.
- Fuel temperature changes (Doppler reactivity changes).

Burnable poison rods placed in some fuel assemblies are used for radial flux shaping and to control part of the reactivity change due to fuel depletion and fission product buildup.

The applicant has provided data to show that adequate control exists to satisfy the above requirements with enough additional control rod worth to provide a hot shutdown effective multiplication factor less than the design basis value of 0.982 during initial and equilibrium fuel cycles with the most reactive control rod stuck out of the core. In addition, the chemical and volume control system will be capable of shutting down the reactor by adding soluble boron and maintaining it shut down in the cold, xenon free condition at any time in core life. These two systems satisfy the requirements of General Design Criterion 26.

Comparisons have been made between calculated and measured control rod bank worth in operating reactors and in critical experiments. These comparisons lead to the conclusion that bank worths may be calculated to within approximately ten percent. In addition bank worth measurements are performed as part of the startup test program to assure that conservative values have been used in safety analyses.

Based on these comparisons, we conclude that the applicant has made suitably conservative assessments of reactivity control requirements and that adequate control rod worths have been provided to assure shutdown capability.

Control Rod Patterns and Reactivity Worths

The control rods are divided into two categories - shutdown rods and regulating rods. The shutdown rods are always completely out of the core when the reactor is at operating conditions. Core power changes are made with regulating rods which are nearly out of the core when it is operating at full power. Regulating rod insertion will be controlled by power-dependent insertion limits required in the Technical Specifications to assure that:

- There is sufficient negative reactivity available to permit rapid shutdown of the reactor with adequate margin.

- The worth of a control rod that might be ejected is not greater than that which has been shown to have acceptable consequences in the safety analyses.

We have reviewed the calculated rod worths and the uncertainties in these worths, and conclude that rapid shutdown capability exists at all times in core life assuming the most reactive control rod assembly is stuck out of the core.

Stability

The stability of the Beaver Valley Unit 2 core to xenon induced spatial oscillations is discussed in the FSAR. The overall negative reactivity (power) coefficient provides assurance that the reactor will be stable

against total power oscillation. The applicant also concluded that sustained radial or azimuthal xenon oscillations are not possible. This conclusion is based on measurements on an operating reactor of the same dimensions which showed stability against these oscillations. We concur with this conclusion.

This core is predicted to be unstable with respect to axial xenon oscillations after about 12000 Megawatt days per ton of exposure. The applicant has acceptably shown that axial oscillations may be controlled by the regulating rods to prevent reaching any fuel damage limits.

Criticality of Fuel Assemblies

Criticality of fuel assemblies outside the reactor is precluded by adequate design of fuel transfer and storage facilities. The applicant presents information on calculational techniques and assumptions used to assure that criticality is avoided. We have reviewed this information and the criteria which will be employed and find them to be acceptable.

Vessel Irradiation

Values are presented for the neutron flux in various energy ranges at mid-height of the pressure vessel inner boundary. Core flux shapes calculated by standard design methods are input to a transport theory calculation (Sn) which results in a neutron flux of 2.9×10^{10} neutrons per square centimeter per second having energy greater than 10^6 electron-volts at the inner vessel boundary. This results in a fluence of 2.9×10^{19} neutrons per square centimeter for a forty year vessel life with an 80 percent use factor. The methods used for these calculations are state of the art, and we conclude that acceptable analytical procedures have been used to calculate the vessel fluence. The Materials Engineering Branch will review the requirements for surveillance programs and the pressure-temperature limits for operation.

4.3.3 Analytical Methods

The applicant has described the computer programs and calculational techniques used to obtain the nuclear characteristics of the reactor design. The calculations consist of three distinct types, which are performed in sequence:

determination of effective fuel temperatures, generation of macroscopic few-group parameters, and space-dependent few-group diffusion calculations. The programs used (e.g., LASER, TWINKLE, LEOPARD, TURTLE and PANDA) have been applied as part of the applications for most earlier Westinghouse designed nuclear plant facilities and the predicted results have been compared with measured characteristics obtained during many startup tests for first cycle and reload cores. These results have validated the ability of these methods to predict experimental results. We, therefore, conclude that these methods are acceptable for use in calculating the nuclear characteristics of Beaver Valley Unit 2.

4.3.4 Summary of Evaluation Findings

The Beaver Valley Unit 2 nuclear design was reviewed according to Section 4.3 of the Standard Review Plan (NUREG-0800). All areas of review and review procedures from that section have been followed either for this reactor or for previous similar reactors (e.g., Summer) or for Topical Report reviews.

The applicant has described the computer programs and calculational techniques used to predict the nuclear characteristics of the reactor design and has provided examples to demonstrate the ability of the analyses to predict reactivity and physics characteristics of the Beaver Valley Unit 2 plant.

To allow for changes of reactivity due to reactor heatup, changes in operating conditions, fuel burnup, and fission product buildup, a significant amount of excess reactivity is designed into the core. The applicant has provided substantial information relating to core reactivity balances for the first cycle and has shown that means have been incorporated into the design to control excess reactivity at all times. The applicant has shown that sufficient control rod worth is available to make the reactor subcritical with an effective multiplication factor no greater than 0.982 in the hot condition at any time during the cycle with the most reactive control rod stuck in the fully withdrawn position. On the basis of our review, we conclude that the applicant's assessment of reactivity control requirements over the first core cycle is suitably conservative, and that adequate negative worth has been provided by the control system to assure shutdown capability. Reactivity control require-

ments will be reviewed for additional cycles as this information becomes available. We also conclude that nuclear design bases, features, and limits have been established in conformance with the requirements of General Design Criteria 10, 11, 12, 13, 20, 25, 26, 27, and 28.

This conclusion is based on the following:

1. The applicant has met the requirements of GDC 11 with respect to prompt inherent nuclear feedback characteristics in the power operating range by:

   a. Calculating a negative Doppler coefficient of reactivity, and

   b. Using calculational methods that have been found acceptable.

   The staff has reviewed the Doppler reactivity coefficients in this case and found them to be suitably conservative.

2. The applicant has met the requirements of GDC 12 with respect to power oscillations which could result in conditions exceeding specified acceptable fuel design limits by:

   a. Showing that such power oscillations are not possible and/or can be easily detected and thereby remedied, and

   b. Using calculational methods that have been found acceptable.

3. The applicant has met the requirements of GDC 13 with respect to provisions of instrumentation and controls to monitor variables and systems that can affect the fission process by:

   a. Providing instrumentation and systems to monitor the core power distribution, control rod positions and patterns, and other process variables such as temperature and pressure, and

b. Providing suitable alarms and/or control room indications for these monitored variables.

4. The applicant has met the requirements of GDC 26 with respect to provision of two independent reactivity control systems of different designs by:

   a. Having a system that can reliably control anticipated operational occurrences,

   b. Having a system that can hold the core subcritical under cold conditions, and

   c. Having a system that can control planned, normal power changes.

5. The applicant has met the requirements of GDC 27 with respect to reactivity control systems that have a combined capability in conjunction with poison addition by the emergency core cooling system of reliably controlling reactivity changes under postulated accident conditions by:

   a. Providing a movable control rod system and a liquid poison system, and

   b. Performing calculations to demonstrate that the core has sufficient shutdown margin with the highest-worth stuck rod.

6. The applicant has met the requirements of GDC 28 with respect to postulated reactivity accidents by (reviewed under Section 15.4.8):

   a. Meeting the regulatory position in Regulatory Guide 1.77,

   b. Meeting the criteria on the capability to cool the core, and

c. Using calculational methods that have been found acceptable for reactivity insertion accidents.

7. The applicant has met the requirements of GDC 10, 20, and 25 with respect to specified acceptable fuel design limits by providing analyses demonstrating:

   a. That normal operation, including the effects of anticipated operational occurrences, have met fuel design criteria,

   b. That the automatic initiation of the reactivity control system assures that fuel design criteria are not exceeded as a result of anticipated operational occurrences and assures the automatic operation of systems and components important to safety under accident conditions, and

   c. That no single malfunction of the reactivity control system causes violation of the fuel design limits.

REFERENCES

WCAP-8385, T. Morita, et al., "Power Distribution Control and Load Follow Procedure", September 1974.

WCAP-8589, K. A. Jones, et al., "Axial Power Distribution Monitoring System", August, 1975.

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 to 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 on Section 4.2, "Fuel System Design."

4.4.2.3 Core Flow

Section 4.4.1(3) of the FSAR has the following core flow design basis.

"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 van 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. The test program predicted heat flux includes a 0.88 multiplier which is part of the 17x17 modified spacer factor. However, a multiplier of 0.865 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 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 nonuniform 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 instability 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.

Prior to issuance of the Technical Specifications, the staff will ensure that the thermal margin reductions consistent with the methods discussed above have been included.

For plants designed by Westinghouse, the staff has approved the generic margins given in Table 4.4-1 (taken from "Interim Safety Evaluation Report on the Effects of Fuel Rod Bowing on Thermal Margin Calculations for Light Water Reactors," December 1976), 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.

The applicant should insert into the basis of the Technical Specification any of the generic or plant-specific margins that may be used to offset the reduction in DNBR due to rod bowing.

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

4.4.4.2 Crud Deposition

Operating experience on two pressurized water reactors indicate a significant reduction in the core flow rate can occur over a relatively short period of time as a result of crud deposition on the fuel rods. In establishing the Technical Specifications for Beaver Vally Unit 2, we will require provisions to assure that the minimum design flow rates are achieved. We also require that the applicant provides a description of the flow measurement capability for Beaver Valley Unit 2 as well as a description of the procedure to measure flow.

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 the 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 order to complete our review, we will require the following additional information from the applicant:

(1) a description of the monitoring equipment including sensor type and location, data acquisition, recording and calibration equipment;

(2) a description of how alert levels will be determined, including sources of internal and external noise, diagnostic procedures used to confirm the presence of a loose part, and precautions to ensure acquisition of quality data;

(3) a description of the operation program, including signature analysis during startup, normal containment environment operation, the seismic design, and system sensitivity;

(4) a detailed discussion of the operator training program for operation of the LPMS, planned operating procedures, and record keeping procedures;

(5) a report from the applicant which contains an evaluation of the system for conformance to Regulatory Guide 1.133; and

(6) a commitment from the applicant to supply, prior to power operation, a report describing operation of the system hardware and implementation of the loose part detection program.

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 the 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 analysis taking into account the effect of partial loop operation on core inlet flow distribution and MDNBR. If the applicant chooses to not use the N-1 loop operation then the staff will require that the Technical Specification include appropriate provision to ensure that this type of operation is prohibited.

4.4.8 ICC Instrumentation

A clarification of requirements for ICC instrumentation which is to be installed and operating prior to fuel load was provided in Section II.F.2 of NUREG-0737 "Clarification of TMI Action Plan Requirements." On November 4, 1982, the Commission determined that an instrumentation system for detection of inadequate core cooling (ICC) consisting of upgraded subcooling margin monitors, core exit thermocouples, and a reactor coolant inventory tracking system is required for the operation of pressurized water reactor facilities. The staff has also completed the review of several proposed generic reactor level or inventory tracking systems for the detection of ICC in PWRs and has found that the Combustion Engineering Heated Junction Thermocouple (HJTC) system and the Westinghouse Reactor Vessel Level Instrumentation System (RVLIS) are acceptable for tracking reactor coolant system inventory. The details of the staff review of this generic system are reported in NUREG/CR-2627

for Westinghouse systems. We have reviewed the applicant's submittal of the instrumentation for indication of inadequate core cooling (Section 4.4.6.4) and found it insufficient; therefore, the staff will require the applicant to provide the itemized documentation of a complete ICCI system on a schedule which will permit completion of our review prior to fuel load. We will report our findings in a future SER supplement.

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 $F_q = 2.32$

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 for design approval. 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 perform the following:

(1) provide a commitment to supply a report describing the loose parts detection program and implementation of the system as described in Section 4.4.5 of this SER;

(2) supply the II.F.2 information previously enumerated in Section 4.4.8 of this SER;

(3) provide a description of the flow measurement capability and the procedure used to measure flow as described in Section 4.4.4.2 of this SER;

(4) address the concerns regarding the effect of rod bow on DNBR as described in Section 4.4.4.1 of this SER.

These issues will be addressed in a supplement to this SER.

15.4.1 Uncontrolled Rod Cluster Control Assembly (Rod) Bank Withdrawal From Zero Power Conditions

Discussion

The consequences of an uncontrolled rod cluster control assembly bank withdrawal at zero power have been analyzed. Such a transient can be caused by a failure of the reactor control or rod control systems. The analysis assumed a conservatively small (in absolute magnitude) negative Doppler coefficient and a conservative moderator coefficient. Further, hot zero power initial conditions with the reactor just critical are chosen because they are known to maximize the calculated consequences. The reactivity insertion rate is assumed to be equivalent to the simultaneous withdrawal of the two highest worth banks at maximum speed (45 inches per minute).

Reactor trip is assumed to occur on the low setting of the power range neutron flux channel at 35 percent of full power (a ten percent uncertainty has been added to the setpoint value). The maximum heat flux is much less than the full power value and average fuel temperature increases to a value lower than the nominal full power value. The minimum DNBR at all times remains above the limiting value of 1.30.

Evaluation Findings

We have reviewed this event according to the Standard Review Plan (NUREG-0800) Section 15.4.1.

The possibilities for single failures of the reactor control system which could result in uncontrolled withdrawal of control rods under low power start-up conditions have been reviewed. The scope of the review has included investigations of initial conditions and control rod reactivity worths, the course

of the resulting transients or steady-state conditions, and the instrument response to the transient or power maldistribution. The methods used to determine the peak fuel rod response, and the input into the analysis, such as power distributions and reactivity feedback effects due to moderator and fuel temperature changes, have been examined.

We conclude that the requirements of General Design Criteria 10, 20, and 25 have been met. This conclusion is based on the following:

The applicant has met the requirement of GDC 10 that the specified acceptable fuel design limits are not exceeded, GDC 20 that the reactivity control systems are automatically initiated so that specified acceptable fuel design limits are not exceeded, and GDC 25 that single malfunctions in the reactivity control system will not cause the specified acceptable fuel design limits to be exceeded. These requirements have been met by comparing the resulting extreme operating conditions and response for the fuel (i.e., fuel duty) with the acceptance criteria for fuel damage (e.g., critical heat flux, fuel temperatures, and clad strain limits should not be exceeded), to assure that fuel rod failure will be precluded for this event. The basis for acceptance in the staff review is that the applicant's analyses of the maximum transients for single error control rod withdrawal from a subcritical or low-power condition have been confirmed, that the analytical methods and input data are reasonably conservative and that specified acceptable fuel design limits will not be exceeded.

15.4.2 Uncontrolled Rod Cluster Control Assembly (Rod) Bank Withdrawal at Power

Discussion

The consequences of uncontrolled withdrawal of a rod bank in the power operating range have been analyzed. The effect of such an event is an increase in coolant temperature (due to the core-turbine power mismatch) which must be terminated prior to exceeding fuel design limits.

The analysis is performed as a function of reactivity insertion rates, reactivity feedback coefficients, and core power level. Protection is provided by the high neutron flux trip, the overtemperature ΔT and overpower ΔT trips, and pressurizer pressure and pressurizer water level trips. In no case does the departure from nucleate boiling ratio fall below 1.30. Adequate fuel cooling is therefore maintained. The maximum heat flux reached including uncertainties does not exceed 118 percent of full power, thus precluding fuel centerline melting.

Evaluation Findings

We have reviewed this event according to Section 15.4.2 of the Standard Review Plan (NUREG-0800).

The possibilities for single failures of the reactor control system which could result in uncontrolled withdrawal of control rods beyond normal limits under power operation conditions have been reviewed. The scope of the review has included investigations of possible initial conditions and the range of reactivity insertions, the course of the resulting transients and the instrumentation response to the transient. The methods used to determine the peak fuel rod response, and the input into the analysis, such as power distributions, rod reactivities, and reactivity feedback effects of moderator and fuel temperature changes, have been examined.

We conclude that the requirements of General Design Criteria 10, 20, and 25 have been met. This conclusion is based on the following:

The applicant has met the requirements of GDC 10 that the specified acceptable fuel design limits are not exceeded, GDC 20 that the reactivity control systems are automatically initiated so that specified acceptable fuel design limits are not exceeded, and GDC 25 that single malfunctions in the reactivity control system will not cause the specified acceptable fuel design limits to be exceeded. These requirements have been met by comparing the resulting extreme operating conditions and response for the fuel (i.e., fuel duty) with the acceptance

criteria for fuel damage (e.g., critical heat flux, fuel temperatures and clad strain limits should not be exceeded), to assure that fuel rod failure will be precluded for this event. The basis for acceptance in the staff review is that the applicant's analysis of maximum transients for single error control rod malfunctions have been confirmed, the analytical methods and input data are reasonably conservative and that specified acceptable fuel design limits will not be exceeded.

15.4.3 Rod Cluster Control Assembly Malfunctions

Discussion

Rod cluster control assembly misalignment incidents including a dropped full length assembly, a dropped full length bank, a misaligned full length assembly and the withdrawal of a single assembly while operating at power have been analyzed by the applicant. Misaligned rods are detectable by: (1) asymmetric power distributions sensed by excore nuclear instrumentation or core exit thermocouples, (2) rod deviation alarm, and (3) rod position indicators. A deviation of a rod from its bank by about 15 inches or twice the resolution of the rod position indicator will not cause power distribution to exceed design limits. Additional surveillance will be required to assure rod alignment if one or more rod position channels are out of service.

In the event of a dropped assembly or group of assemblies the reactor will typically scram on a neutron flux negative rate trip, and analysis indicates that thermal limits will not be exceeded for the event. If the rod locations are such that the reactor does not scram, however, the automatic controller may return the reactor to full power and the control could result in a power overshoot. An analysis methodology for this event has been developed by Westinghouse, and reported in WCAP-10297-P, "Dropped Rod Methodology for Negative Flux Rate Trip Plants," January 1982. This methodology has been reviewed and approved by the NRC staff. The review is in a memorandum for F. Miraglia from L. Rubenstein, "Review of the Westinghouse Report 'Dropped Rod Methodology for Negative Flux Rate Trip Plants'", December 1983. Generally,

detailed analyses for most reactors, for most cycles, show that if this event occurs thermal limits will not be exceeded. However, the analysis is reactor and cycle specific, and the analyses for Beaver Valley Unit 2 for Cycle 1 have not been completed as yet. The staff has also accepted an interim position for operating reactors which consists of a restriction on operations above ninety percent power such that either the reactor is in manual control or rods are required to be out greater than 215 steps. This restriction will be applied to Beaver Valley Unit 2 in the event that calculations for Cycle 1 operation are not completed in time for initial operations. With this restriction thermal limits will not be exceeded. Approval of the analysis specific to Beaver Valley Unit 2 for Cycle 1 will result in removing the restriction. Similar analysis will also be needed for each subsequent reload cycle.

For cases where a group is inserted to its insertion limit with a single rod in the group stuck in the fully withdrawn position analysis indicates that departure from nucleate boiling will not occur. We have reviewed the calculated estimates of the expected reactivity and power distribution changes that accompany postulated misalignments of representative assemblies. We have concluded that the values used in this analysis conservatively bound the expected values including calculational uncertainties.

The inadvertent withdrawal of a single assembly requires multiple failures in the control system, multiple operator errors or deliberate operator actions combined with a single failure of the control system. As a result the single assembly withdrawal is classified as an infrequent occurrence. The resulting transient is similar to that due to a bank withdrawal but the increased peaking factor may cause departure from nucleate boiling to occur in the region surrounding the withdrawn assembly. Less than five percent of the rods in the core experience departure from nucleate boiling for such a transient.

Evaluation Findings

We have reviewed this event according to Section 15.4.3 of the Standard Review Plan (NUREG-0800).

The possibilities for single failures of the reactor control system which could result in a movement or malposition of control rods beyond normal limits have been reviewed. The scope of the review has included investigations of possible rod malposition configurations, the course of the resulting transients or steady-state conditions, and the instrumentation response to the transient or power maldistribution. The methods used to determine the peak fuel rod response, and the input to that analysis, such as power distribution changes, rod reactivities, and reactivity feedback effects due to moderator and fuel temperature changes, have been examined.

We conclude that the requirements of General Design Criteria 10, 20, and 25 have been met. This conclusion is based on the following:

The applicant has met the requirements of GDC 10 that the specified acceptable fuel design limits are not exceeded, GDC 20 that the reactivity control systems are automatically initiated so that specified acceptable fuel design limits are not exceeded, and GDC 25 that single malfunctions in the reactivity control system will not cause the specified acceptable fuel design limits to be exceeded. These requirements have been met by comparing the resulting extreme operating conditions and response for the fuel (i.e., fuel duty) with the acceptance criteria for fuel damage (e.g., critical heat flux, fuel temperatures and clad strain limits should not be exceeded), to assure that fuel rod failure will be precluded for this event. The basis for acceptance in the staff review is that maximum configurations and transients for single error control rod malfunctions have been analyzed, that the analysis methods and input data are resonably conservative and that specified acceptable fuel design limits will not be exceeded.

15.4.7 Inadvertent Loading of a Fuel Assembly into Improper Position

Discussion

Strict administrative controls in the form of previously approved established procedures and startup testing are followed during fuel loadings to prevent operation with a fuel assembly in an improper location or a misloaded burnable poison assembly. Nevertheless, an analysis of the consequences of a loading error has been performed.

Comparisons of power distributions calculated for the nominal fuel loading pattern and those calculated for five loadings with misplaced fuel assemblies or burnable poison assemblies are presented by the applicant. The selected non-normal loadings represent the spectrum of potential inadvertent fuel misplacement. Calculations included, in particular, the power in assemblies which contain provisions for monitoring with incore detectors.

As part of the required startup testing, the incore detector system is used to detect misloaded fuel prior to operating at power. The analysis described above shows that all but one of the above misloading events would be detected by this test. In the excepted case, an interchange of Region 1 and 2 assemblies near the center of the core, the increase in the power peaking is approximately equal to the uncertainty in the measurement of this quantity (5 percent). This uncertainty is allowed for in analyses so that this misloading event does not result in unacceptable consequences.

Evaluation Findings

We have reviewed this event according to Section 15.4.7 of the Standard Review Plan (NUREG-0800).

We have evaluated the consequences of a spectrum of postulated fuel loading errors. We conclude that the analyses provided by the applicant have shown for each case considered that either the error is detectable by the available instrumentation (and hence remediable) or the error is undetectable but the

offsite consequences of any fuel rod failures are a small fraction of 10 CFR Part 100 guidelines. The applicant affirms that the available incore instrumentation will be used before the start of a fuel cycle to search for fuel loading errors.

We conclude that the requirements of General Design Criterion 13 and 10 CFR Part 100 have been met. This conclusion is based on the following:

The applicant has met the requirements of GDC13 with respect to providing adequate provisions to minimize the potential of a misloaded fuel assembly going undetected and meets Part 100 with respect to mitigating the consequences of reactor operations with a misloaded fuel assembly. These requirements have been met by providing acceptable procedures and design features that will minimize the likelihood of loading fuel in a location other than its designated place.

15.4.8 Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection)

Discussion

The mechanical failure of a control rod mechanism pressure housing would result in the ejection of a rod cluster control assembly. For assemblies initially inserted, the consequences would be a rapid reactivity insertion together with an adverse core power distribution, possibly leading to localized fuel rod damage. Although mechanical provisions have been made to make this accident extremely unlikely, the applicant has analyzed the consequences of such an event.

Methods used in the analysis are reported in WCAP-7588, Revision 2, "An Evaluation of the Rod Ejection Accident in Westinghouse Reactors Using Spatial Kinetics Methods," which has been reviewed and accepted by the staff. This report demonstrated that the model used in the accident analysis is conservative relative to a three dimensional kinetics calculation.

The applicant's criteria for gross damage of fuel are a maximum clad temperature of 2700 degress Fahrenheit and an energy deposition of 200 calories per gram in the hottest pellet. These criteria are more conservative* than those proposed in Regulatory Guide 1.77. Therefore, they are acceptable.

Four cases were analyzed: beginning-of-cycle at 102 percent and zero power and end-of-cycle at 102 percent and zero power. The highest clad temperatures, 2606 degrees Fahrenheit, and the highest fuel enthalpy, 169 calories per gram, were reached in the end-of-cycle zero power and beginning-of-cycle full power cases respectively. The analysis also shows that less than 10 percent of the fuel experiences departure from nucleate boiling and less than 10 percent of the hot pellet melts. Analyses have been performed to show that the pressure surge produced by the rod ejection is mild and will not approach the Reactor Coolant System emergency limits. Further analyses have shown that a cascade effect, i.e., the ejection of a further rod due to the ejection of the first one, is not credible.

Evaluation Findings

We have reviewed this event according to Section 15.4.8 of the Standard Review Plan (NUREG-0800).

We conclude that the analysis of the rod ejection accident is acceptable and meets the requirements of General Design Criterion 28. This conclusion is based on the following:

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* Regulatory Guide 1.77 has an acceptance criterion of 280 calories per gram energy deposition and no criterion for clad temperature other than that implicit in requirements for fuel and pressure vessel damage.

The applicant met the requirements of GDC 28 with respect to preventing postulated reactivity accidents that could result in damage to the reactor coolant pressure boundary greater than limited local yielding, or cause sufficient damage that would signficantly impair the capability to cool the core. The requirements have been met by demonstrating that the regulatory positions of Regulatory Guide 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for PWR's" are complied with. The staff has evaluated the applicant's analysis of the assumed control rod ejection accident and finds the assumptions, calculation techniques, and consequences acceptable. Since the calculations resulted in peak fuel enthalpies less than 280 cal/gm, prompt fuel rupture with consequent rapid heat transfer to the coolant from finely dispersed molten UO_2 was assumed not to occur. The pressure surge was, therefore, calculated on the basis of conventional heat transfer from the fuel and resulted in a pressure increase below "Service Limit C" (as defined in Section III, "Nuclear Power Plant Components," of the ASME Boiler and Pressure Vessel Code) for the maximum control rod worths assumed. The staff believes that the calculations contain sufficient conservatism, both in the initial assumptions and in the analytical models, to ensure that primary system integrity will be maintained.