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APR 12 1984

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 #3
Project Manager: M. Ley
DSI Review Branch: Core Performance Branch
Review Status: Seven confirmatory and two open issues in Section 4.2

The Core Performance Branch has prepared the enclosed Draft Safety Evaluation Report input for Section 4.2 of the Beaver Valley Unit 2 FSAR. This review was conducted by PNL under contract FIN B2544.

The confirmatory and open issues are identified as follows:

Confirmatory Issues:

1. Confirmation that the peak pellet design basis burnup of 53,000 MWd/MTU is consistent with the region discharge burnup of 33,000 MWd/MTU (see Section 4.2.1).

2. Specification of the correct values for several parameters (e.g., fuel rod diameter and Zircaloy weight) in the description of and design drawings for Beaver Valley Unit 2 fuel (see Section 4.2.2).

3. Confirmation that the rod bowing analysis has been performed (see Section 4.2.3.1(6)).

4. Confirmation that the fuel rod internal pressure is consistent with WCAP-8963 (see Section 4.2.3.1(8)).

5. Confirmation that the predicted cladding collapse time exceeds the expected residence time of the fuel (see Section 4.2.3.2(2)).

6. Confirmation that combined seismic and LOCA loads, using the SRSS method and a worst-case LOCA, are applied in calculating grid stresses (see Section 4.2.3.3(4)).

Contact: M. Dunenfeld, DSI:CPB
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7. Confirmation of the ability of the reactor coolant letdown radiation monitors to detect fuel rod failures (see Section 4.2.4.2).

Open Issues:

1. Fuel assembly non-grid component forces from combined seismic and LOCA loads have not been shown to meet SRP Section 4.2 guidelines (see Section 4.2.3.3(4)).

2. Commitment to use the on-line detection method to monitor fuel rod failures (see Section 4.2.4.2).

SALP Evaluation

We had no direct interaction with the licensee on this review. The written material, however, is of acceptable quality.

Rating: Category 2.

Original signed by
L. S. Rubenstein

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

Enclosure:
As stated

cc: R. Mattson
D. Eisenhut
G. Knighton
M. Ley

OFFICE	DSI:CPB:ew	DSI:CPB:SL	DSI:CPB:BC	DSI:CAPS:AD			
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DATE	04/9/84	04/11/84	04/12/84	04/12/84			

BEAVER VALLEY UNIT 2

DRAFT SAFETY EVALUATION REPORT

4.2 FUEL DESIGN

The Beaver Valley Unit 2 fuel assembly described in the FSAR (as amended through Amendment No. 2 dated July 1983) is a 17X17 array of 0.374 inch diameter fuel rods. This design will be referred to as the Standard Fuel Assembly (SFA) in the following paragraphs.

Section 4.2 of the FSAR presents the design bases for the SFA. For the Westinghouse (W) analysis, plant design conditions are divided into four categories of operation that are in accordance with their anticipated frequency of occurrence and risk to the public and that are consistent with traditional industry classification (ANSI Standards N18.2-1973 and N-212-1974): Condition I is normal operation, Condition II is incidents of moderate frequency, Condition III is infrequent incidents, and Condition IV is limiting faults. Fuel damage is then related to these conditions of operation, which are coupled to the fuel design bases and design limits. The subsections of the design bases section address topics such as (a) cladding, (b) fuel material, (c) fuel rod performance, (d) spacer grids, (e) fuel assembly, (f) reactivity control assembly, burnable poison rods, and source rods; and (g) surveillance program. Thus, as part of the discussion of the cladding design bases, material and mechanical properties, stress-strain limits, vibration and fatigue, and chemical properties are also presented. A similar approach is taken for the other major subtopics.

The review and safety evaluation will follow Standard Review Plan (SRP) Section 4.2 (NUREG-0800, Revision 2). The objectives of this fuel system safety review are to provide assurance that (a) the fuel system is not damaged as a result of normal operation and anticipated operational occurrences (AOO), (b) fuel system damage is never so severe as to prevent control rod insertion when it is required, (c) the number of fuel rod failures is not underestimated for postulated accidents, and (d) coolability is always maintained. "Not damaged" is defined as meaning that fuel rods do not fail, that fuel system dimensions remain within operational tolerances, and that functional capabilities are not reduced below those assumed in the safety analysis. This objective implements General Design Criterion (GDC) 10 of 10 CFR Part 50, Appendix A ("GDC for Nuclear Power Plants") and the design limits that accomplish this are called Specified Acceptable Fuel Design Limits (SAFDLs). "Fuel rod failure" means that the fuel rod leaks and that the first fission product barrier (the cladding) has, therefore, been breached. Fuel rod failures must be accounted for in the dose analysis required by 10 CFR Part 100 ("Reactor Site Criteria") for postulated accidents. "Coolability," which is sometimes termed "coolable geometry," means, in general, that the fuel assembly retains its rod-bundle geometrical configuration with adequate coolant channels to permit removal of residual heat after a severe accident. The general requirements to maintain control rod insertability and core coolability appear repeatedly in GDC 27 and 35. Specific coolability requirements for the loss-of-coolant accidents are given in 10 CFR Part 50.46 ("Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors").

To meet the above-stated objectives of the fuel system review, the following specific areas are critically examined: (a) design bases, (b) description and design drawings, (c) design evaluation, and (d) testing, inspection, and surveillance plans. In assessing the adequacy of the design, several items involving operating experience, prototype testing, and analytical predictions are weighed in terms of specific acceptance criteria for fuel system damage, fuel rod failure, and fuel coolability. Recently, W developed the Optimized Fuel Assembly (OFA) which is described in WCAP-9500. This report was approved by the NRC (Rubenstein, May 15, 1981 and Tedesco, May 22, 1981). The OFA design also consists of a 17X17 array of fuel rods but with the rods having a diameter of 0.360 in., which is somewhat smaller than the rod diameter in the SFA. Because the format of WCAP-9500 followed Regulatory Guide 1.70, some of the fuel design bases and design limits for the OFA were not presented in WCAP-9500 in a form that facilitated cross-checking by the NRC with the acceptable criteria provided in Section 4.2 of the SRP. Therefore, several questions were issued (Rubenstein, August 8, 1980) to clarify the design bases and limits. Responses to those questions are contained in letters from W (Anderson, January 12, 1981 and April 21, 1981). These responses are applicable to the SFA as well (Petrick, September 9, 1981). Reference to these questions and answers will be made at several places in the review that follows.

4.2.1 Design Bases

Design bases for the safety analysis address fuel system damage mechanisms and suggest limiting values for important parameters such that damage will be limited to acceptable levels. For convenience, acceptance criteria for these design limits are grouped into three categories in the SRP: (a) fuel system damage criteria, which are most applicable to normal operation (W plant Condition I), including AOOs (W plant Condition II), (b) fuel rod failure criteria, which apply to normal operation (W plant Condition I), AOOs (W plant Condition II), and postulated accidents (W plant Conditions III and IV), and (c) fuel coolability criteria, which apply to postulated accidents (W plant Conditions III and IV).

The Beaver Valley Unit 2 FSAR has referenced WCAP-9500 to augment the fuel system design bases and limits. As noted above, some of the design bases and limits requested in the SRP were not presented in the originally submitted WCAP-9500; however, these were supplied in several responses to NRC questions (Anderson, January 12, 1981 and April 21, 1981). WCAP-9500 and the augmenting responses provide the majority of the design bases and limits for the Beaver Valley Unit 2 fuel design.

The FSAR also provides design bases and limits in addition to those provided in WCAP-9500. These design bases and limits were reviewed and found to be acceptable with respect to SRP guidelines. Therefore, the design bases and limits (i.e., Fuel System Damage Criteria,[1] Fuel Rod Failure Criteria,[2] and Fuel Coolability Criteria[3]) presented in the FSAR (and WCAP-9500) are found to be acceptable. The applicant should confirm that the peak pellet design basis burnup of 53,000 MWd/MTU shown in the second paragraph of Section 4.2.1 of the FSAR is consistent with the region discharge burnup of 33,000 MWd/MTU shown under "Basis" in Section 4.3.1.1 of the FSAR.

4.2.2 Description and Design Drawings

The description of fuel system components, including fuel rods, bottom and top nozzles, guide and instrument thimbles, grid assemblies, rod cluster control assemblies, burnable poison assemblies, neutron source assemblies, and thimble plug assemblies, is contained in Section 4.2.2 of the FSAR. In addition, Tables 4.1-1 and 4.3-1 of the FSAR provide numerical values for various core component parameters. While each parameter listed in SRP subsection 4.2.II.B is not provided in the FSAR,

[1]Fuel system damage criteria for cladding design stress, cladding design strain, strain fatigue, fretting wear, oxidation and crud buildup, rod bowing, axial growth, fuel rod and nonfueled rod pressures, assembly liftoff, and control material leaching.

[2]Fuel rod failure criteria for internal hydriding, cladding collapse, overheating of cladding, overheating of fuel pellets, pellet/cladding interaction, and cladding rupture.

[3]Fuel collability criteria for fragmentation of embrittled cladding, violent expulsion of fuel, cladding ballooning and flow blockage, and structural damage from external forces.

enough information is provided in sufficient detail to provide a reasonably accurate representation of the SFA design and this information is thus acceptable. However, there appears to be some OFA data rather than SFA data in several places in the FSAR: (a) in Figure 4.2-1, the fuel rod diameter is shown as 0.360 inch (OFA size) rather than 0.374 inch (SFA size, as indicated in Table 4.1-1) and (b) in comparing entries in Tables 4.1-1 and Table 4.3-1, the Zircaloy weight is listed as 41.415 lb and 38,230 lb, respectively. Several other discrepancies were also noted in the FSAR: (a) in Figure 4.2-1, 204 fuel rods are shown as required, rather than 264 as noted in Section 4.1.1 and Table 4.3-1, (b) certain dimensions (e.g., 154.0 REF, 133.4 REF, 112.8 REF, etc.) appear to be incorrectly shown in Figure 4.2-2, and (c) in comparing entries in Table 4.1-1 and Table 4.3-1, the fuel weight (as UO_2) is listed as 181,190 lb and 181,205 lb, respectively. The applicant should specify the correct values for Beaver Valley Unit 2 fuel.

4.2.3 Design Evaluation

Design bases and limits were presented and discussed in Section 4.2.1, above. In this section, W methods of demonstrating that the SFA fuel design meets the design criteria that have been established are reviewed.

This section will, therefore, correspond point by point to Section 4.2.1, above. The methods of demonstrating that the design criteria have been met include operating experience, prototype testing, and analytical predictions.

4.2.3.1 Fuel System Damage Evaluation

The following paragraphs discuss the evaluation of the ability of the SFA fuel to meet the fuel system damage criteria described in Section 4.2.1, above. Those criteria apply only to normal operation and anticipated transients.

1. Cladding Design Stress

Westinghouse used its Performance-Analysis and Design (PAD) code to analyze cladding stress (WCAP-8720). That code has been reviewed and found acceptable (Stolz, February 9, 1979). Typical calculated design values for cladding effective stress are stated to be considerably below the 0.2 percent offset yield stress design limit.

2. Cladding Design Strain

The NRC-approved W fuel performance code, PAD, was used in the strain analysis, as indicated in the response to Question 231.2[4] and in Section 4.2.3.3 of the FSAR. Typical design values of steady-state and transient creep strain, as calculated by that code, are found to be below the 1 percent strain criterion. Hence, we conclude that the SFA cladding strain design limits have been met.

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[4]All questions and responses referred to in this manner were part of the review of WCAP-9500, and the first application of the SFA, on the Shearon Harris Docket (50-400). References to the FSAR, refer to the Beaver Valley Unit 2 FSAR.

3. Strain Fatigue

As indicated in the response to Question 231.2, W used their approved PAD code for the strain range and strain fatigue life usage analysis. Experimental data obtained from W testing programs (see Section 4.2.3.3 of the FSAR) were used by W to derive the Zircaloy fatigue design curve, according to the response to Question 231.4. For a given strain range, the number of fatigue cycles is less than that required for failure, considering (see Section 4.2.3.3 of the FSAR) a minimum safety factor of 2 on stress amplitude or a minimum safety factor of 20 on the number of cycles, (the fatigue usage factor is less than 1.0). The computations were performed with an approved code. It is concluded that the SFA fatigue design basis has been met.

4. Fretting Wear

With regard to the W fretting analysis of the fuel cladding, the NRC staff concludes the following.

(a) Cladding fretting and fuel vibration have been experimentally investigated, as shown in WCAP-8278 and WCAP-8279, and noted in Section 4.2.3.1(1) of the FSAR. WCAP-8278 and WCAP-8279 have been approved by us (Rubenstein, March 19, 1981 and June 30, 1982).

(b) The out-of-pile flow tests and analyses (WCAP-9401 and WCAP-9402) to determine the magnitude of fretting wear that is anticipated for the OFA design have been previously reviewed and found acceptable (Rubenstein, April 23, 1981). These analyses are also acceptably conservative for SFA applications.

(c) LWR operating experience demonstrates that the number of fretting-induced fuel failures is insignificant.

(d) There should be only a small dependence of cladding stresses on fretting wear because this type of wear is local at grid-contact locations and relatively shallow in depth.

(e) The built-in conservatisms (that is, safety factors of 2 on the stress amplitudes and 20 on the number of cycles) in the strain fatigue analysis as well as the calculated margin to fatigue life limit adequately offset the effect of fretting wear degradation.

Therefore, it is concluded that the SFA fuel rods will perform adequately with respect to fretting wear.

Fretting wear has also been observed on the inner surfaces of guide thimble tubes where the fully withdrawn control rods reside. Significant wear is limited to the relatively soft Zircaloy-4 guide thimble tubes because the Inconel or stainless steel control rod claddings are relatively wear-resistant. The extent of the wear is both time-dependent and plant-dependent and has, in some non-W cases, extended completely through the guide thimble tube wall.

Westinghouse has predicted that an SFA can operate under a rod cluster control assembly (RCCA) for a period of time that exceeds that amount of rodded time expected with current 3-cycle fuel schemes before fretting wear degradation would result in exceeding the present margin to the 6-g load criterion for the fuel handling accident. However, we required several applicants to perform a surveillance program because of the uncertainties in predicting wear rates for the standard 17X17 fuel assembly design. The objective of this program was to demonstrate that there was no occurrence of hole formation in rodded gude thimble tubes, thus providing some confidence that scrammability is ensured. These applicants formed an owners group, which has submitted a generic report (Leasburg, March 1, 1982) that provides postirradiation examination results on guide thimble tube wear in the W 17X17 fuel assembly design. Based on this report, we have concluded (Rubenstein, April 19, 1982) that the W 17X17 fuel assembly design is resistant to guide thimble tube wear.

5. Oxidation and Crud Buildup

In the FSAR, there is no explicit discussion of cladding oxidation, hydriding, and crud buildup. However, it is indicated in Sections 4.4.2.9.1 and 4.4.2.11 of the FSAR that the thermal model used for computation of radial fuel rod temperature distributions combines crud, oxide, fuel-cladding gap, and fuel pellet conductances and that the model has been quantified by (a) comparison of its results with those from in-reactor thermocouple measurements, (b) out-of-reactor measurements of fuel and cladding properties, and (c) measurements of fuel and cladding dimensions during fabrication. In Section 4.4.2.11.5 of the FSAR, it is stated that allowance is made in the fuel center melt evaluation for the temperature rise due to the buildup of oxides and crud on the fuel rod surface over the life of the core. It is stated in Section 4.4.4.5.2 of the FSAR that the effect of crud on flow and enthalpy distribution in the core is accounted for in the steady-state analysis by assuming a crud thickness several times that which would be expected to occur. Also, operating experience of W-designed reactors has indicated that a flow resistance allowance for possible crud deposits is not required as there has been no detectable long-term flow reduction reported for any plant.

The applicable models for cladding oxidation and crud buildup are discussed in the supporting documentation (Salvatori, January 4, 1973) for the fuel performance code PAD-3.1. These models were previously approved by us. A new temperature-dependent cladding oxidation model is also presented in WCAP-9179 (Section 4.2.3 of the FSAR). Because the temperature-independent model in PAD-3.1 is conservative with respect to the approved (Rubenstein, October 21, 1982) model in WCAP-9179, we continue to find the older models applicable. These models affect the cladding-to-coolant heat transfer coefficient and the temperature drop across the cladding wall. Mechanical properties and analyses of the cladding are not significantly impacted by oxide and crud buildup. On the basis of the W discussion (Anderson, January 12, 1981) of the impact of cladding hydriding on fuel performance, and on our previous review of the oxidation and crud buildup models, we conclude that these effects have been adequately accounted for in the SFA.

6. Rod Bowing

A rod bowing correlation (Anderson, April 19, 1978) for the amount of fuel rod bowing as a function of fuel burnup has been approved (Meyer, March 2, 1978). The correlation has also been used by others (Rubenstein, October 21, 1982) to analyze the SFA design. Revision 1 of WCAP-8691, the rod bowing topical report, has been approved (Rubenstein, October 25, 1982). There is no mention in Sections 4.2, 4.3, and 4.4 of the FSAR that the rod bowing analysis was actually performed for Beaver Valley Unit 2 fuel. The applicant must confirm that the rod bowing analysis has been performed.

7. Axial Growth

As noted in the DSER for Shearon Harris (Rubenstein, October 21, 1982), which also uses the SFA design, we are aware of supporting information (Bloom, April 1972, and Appleby, April 1972) that was not cited by W, but which also implies that irradiation growth of strainless steel should not be significant at the temperatures and fluences that are associated with PWR operation. Furthermore, because we are unaware of any operating experience that indicates axial-growth-related problems in W NSSS plants, we conclude (Rubenstein, October 21, 1982) that W has made sufficient accommodation for control, source, and burnable poison rod growth in their NSSS designs.

The W analysis of shoulder gap spacing (e.g., see Section 4.2.3.5.1 of the FSAR) for the SFA has found that interference will not occur until achieving burnups beyond traditional values. We, therefore, find (Rubenstein, October 21, 1982) that the required shoulder gap spacing has been reasonably accommodated. However, for extended burnup applications, the adequacy of the spacing should be reverified. Furthermore, because stress-free irradiation growth of zirconium-bearing alloys is sensitive to texture (preferred cystallographic orientation) and retained cold work, which, in turn, are strongly dependent on the specific fabrication techniques that are employed during component production, reverification of the design shoulder gap should be performed if W current fabrication specifications are significantly altered.

Finally, we find (Rubenstein, October 21, 1982) the W analysis of fuel assembly growth to be acceptable. However, as stated in the above discussion on shoulder gap spacing, reverification of the fuel assembly growth should be performed if significant changes are made in the W current fabrication techniques.

8. Fuel Rod and Non-fueled Rod Pressures

As noted in Section 4.2.3.1(2) of the FSAR, the approved (Stolz, February 9, 1979) W PAD-3.3 fuel performance code, WCAP-8720 (and WCAP-8785), was used in determining the internal gas pressure of the fuel rods as a function of irradiation time. The applicant needs to confirm that the fuel rod internal pressure is consistent with approved (Stolz, May 19, 1978) topical report, WCAP-8963.

The analysis of non-fueled rod internal pressure for the SFA is generally based on Section III, Subsections NB and NG, of the ASME Code (see Section 4.2.1.6 of the FSAR). As noted in Sections 4.2.1.6, 4.2.2.3.2, and 4.2.2.3.2 of the FSAR, the control rod, neutron source rod, and burnable poison rod cladding is cold-worked Type 304 stainless steel, which is not covered by the ASME code. Westinghouse, therefore, defines the stress intensity limit, Sm, for this material as equal to 2/3 of the 0.2 percent offset yield stress. The yield for this material is approximately 62,000 psi. A strain limit of 1 percent also applies to the cladding. Predicted maximum values of rod internal pressure have been provided in the response to Question 231.2 and they are well below those imposed by the cladding stress and strain limits.

We conclude that there is adequate assurance that nonfueled core component rods can operate safely during Conditions I and II because appropriate stress and strain limits are met even though the maximum internal rod pressure may exceed system pressure.

9. Assembly Liftoff

In response to our question on this topic, W has confirmed (Rubenstein, October 21, 1982) that momentary liftoff will occur only during a turbine overspeed transient (this is also stated in Section 4.4.2.6.2 of the FSAR). Westinghouse has further found that (a) proper reseating will occur after momentary liftoff, (b) damage to adjacent assemblies will not occur even if one assembly is fully lifted and the adjacent ones remain seated, and (c) no adverse consequences of momentary liftoff are expected. We agree with the W conclusions and, therefore, conclude that fuel assembly liftoff has been adequately addressed for the SFA design.

10. Control Material Leaching

While the design basis for the SFA control rods is to maintain cladding integrity, and while the probability of control rod cladding failures appears to be quite low, we have considered the corrosion behavior of the control material and burnable poison and conclude that a breach in the cladding should not result in serious consequences because the hafnium absorber material and the poison material (borosilicate glass) are relatively inert.

4.2.3.2 Fuel Rod Failure Evaluation

The following paragraphs discuss the evaluation of (a) the ability of the SFA fuel to operate without failure during normal operation and anticipated transients, and (b) the accounting for fuel rod failures in the applicant's accident analysis. The fuel rod failure criteria described in Section 4.2.1, above, were used for this evaluation.

1. Internal Hydriding

Westinghouse has used moisture and hydrogen control limits in the manufacture of earlier fuel types and has found that typical end-of-life cladding hydrogen levels are less than 100 ppm--a level below which hydride blister formation is not anticipated in fuel cladding. As described in Section 4.2.3.1(3) of the FSAR, the moisture levels in the uranium dioxide fuel are limited by W to less than or equal to 20 ppm. This specification is compatible with the ASTM specification for sintered uranium dioxide pellets, which allows two micrograms of hydrogen per gram of uranium (2 ppm). These are the same limits provided in the SRP and are, therefore, acceptable.

We, therefore, conclude that reasonable evidence has been provided that hydriding as a fuel failure mechanism will not be significant in the SFA.

2. Cladding Collapse

In calculating the time at which cladding collapse will occur, W uses the generic methods described in WCAP-8377, which is approved (Stello, January 14, 1975) for licensing applications. Inputs to the analysis include cladding ovality, helium prepressurization, free volume of the fuel rod, and limiting power histories.

The applicant has not yet demonstrated that the calculated cladding collapse time for Beaver Valley Unit 2 fuel using WCAP-8377 methods is more than the expected residence of the fuel. We will report the resolution of this issue in a supplement to this SER.

3. <u>Overheating of Cladding</u>

As stated in SRP Section 4.2, adequate cooling is assumed to exist when the thermal margin criterion to limit the departure from nuclear boiling (DNB) or boiling transition in the core is satisfied. The method employed to meet the DNB design basis is reviewed in Section 4.4.

4. <u>Overheating of Fuel Pellets</u>

The design evaluation of the fuel centerline melt limit is performed with the <u>W</u> fuel performance code, PAD-3.3 (WCAP-8720). This code, which has been approved (Stolz, February 9, 1979), is also used to calculate initial conditions for transients and accidents described in Chapter 15 of the SRP (see Paragraph 4.2.3.3(1) below for further comments on PAD-3.3).

In applying the PAD-3.3 code to the centerline melting analysis, the melting temperature of the UO_2 is assumed to be 5081°F unirradiated and is decreased by 58°F per 100,000 MWd/t. This relation has been almost universally adopted by the industry and has been accepted by us in the past. The expressions for thermal conductivity and gap conductance, described in Section 4.4.2.11 of the FSAR, are unchanged from that originally described in the PAD code.

In order to avoid using the PAD code to calculate a continuous set of burnup-dependent conditions necessary to cause centerline melting, W has performed the calculation for a single case, as described in WCAP-9500. This was done by assuming a UO_2 melting temperature of 4701°F, which corresponds to the melting temperature at 65,000 MWd/t, and melting occurred at a linear power rating of approximately 21 kW/ft. The limiting local power for the worst Condition II transient, boron dilution with automatic rod control, is less than or equal to 18 kW/ft for W plants with 17X17 fuel. Thus, the centerline melt criterion is satisfied in an acceptable manner.

5. Pellet/Cladding Interaction

The only two PCI criteria in current use in licensing (1 percent cladding strain and no fuel melting), while not broadly applicable, are easily satisfied. As noted in the discussion of the cladding stress and strain evaluation, W uses an approved code, PAD, to calculate creep strain, and the values calculated by that code are found to be below the 1 percent strain criterion. And, as indicated in the discussion on overheating failures, the non-centerline-melt criterion is satisfied based on an analysis (described in Section 4.2.3.2(4) of the FSAR) with an approved code. Therefore, the two existing licensing criteria for PCI have been satisfied.

In addition to the SRP-type treatment of PCI, however, FSAR Section 4.2.3.3(a) addresses PCI from the standpoint of its effect on fatigue life. PCI produces cyclic stresses and strains that can affect fatigue life of the cladding. Furthermore, gradual compressive creep of the cladding onto the fuel pellet occurs due to the differential pressure exerted on the fuel rod by the coolant. Westinghouse contends that, by using prepressurized fuel rods, the rate of cladding creep is reduced, thus delaying the time at which fuel-to-cladding contact first occurs. We agree that fuel rod prepressurization should improve PCI resistance, albeit in a presently unquantified amount.

In conclusion, W has used approved methods to demonstrate that the present PCI acceptance criteria have been met.

6. Cladding Rupture

The large break LOCA analysis for Beaver Valley Unit 2 was performed with a revised cladding rupture temperature correlation that has recently been approved (Miller, December 1981) as an integral part of the 1981 ECCS evaluation model. This new model eliminates the need for supplemental calculations, which have been required from applicants that used earlier ECCS models. The use of the new cladding rupture temperature correlation is found to be acceptable. The overall impact of cladding rupture on the response of the Beaver Valley Unit 2 fuel design to the loss-of-coolant accident is evaluated in Section 15.6.5 and is not reviewed further in this section.

4.2.3.3 <u>Fuel Coolability Evaluation</u>

The following paragraphs discuss the evaluation of the ability of the SFA fuel to meet the fuel coolability criteria described in Section 4.2.1, above. Those criteria apply to postulated accidents.

1. <u>Fragmentation of Embrittled Cladding</u>

The primary degrading effect of a significant degree of cladding oxidation is embrittlement of the cladding. Such embrittled cladding will have a reduced ductility and resistance to fragmentation. The most severe occurrence of such embrittlement is during a LOCA. The overall effects of cladding embrittlement on the SFA design for the loss-of-coolant accident are analyzed in Section 15.6.5 and are not reviewed further in this section.

One of the most significant analytical methods that is used to provide input to the analysis in Section 15.6.5 is the steady-state fuel performance code, which is reviewed in Section 4.2. This code provides fuel pellet temperatures (stored energy) and fuel rod gas inventories for the ECCS evaluation model as prescribed by Appendix K to 10 CFR 50. The code accounts for fuel thermal conductivity, fuel densification, gap conductance, fuel swelling, cladding creep, and other phenomena that affect the initial stored energy. For this purpose, <u>W</u> uses a relatively new fuel performance code called PAD-3.3 (WCAP-8720). This code was approved by our safety evaluation (Stolz, February 9, 1979). We, therefore, find the analysis described for the Beaver Valley Unit 2 fuel design acceptable as docketed for all cycles of operation.

For non-LOCA events, the locked rotor accident (one-pump seizure with two and three loops operating) is the most severe undercooling event that is analyzed. This event is analyzed in Section 15.3.3 of the FSAR, where it is found that the peak cladding temperature is well below the 2700^{o}F design limit. The analysis of this event is reviewed in Section 15.3.3 of this report, but it is clear that the Beaver Valley Unit 2 fuel design meets the non-LOCA peak cladding temperature design limit.

2. Violent Expulsion of Fuel Material

The analysis that demonstrates that the design limits are met for this event with the SFA is presented in Section 15.4.8 of the FSAR and is reviewed in that section of this report.

3. Cladding Ballooning and Flow Blockage

The large break LOCA analysis for Beaver Valley Unit 2 was performed with the revised cladding ballooning and assembly flow blockage models which have recently been approved (Miller, December 1, 1981) as integral parts of the 1981 ECCS evaluation model. These revised models eliminate the need for supplemental calculations that have been required from applicants that used the earlier ECCS models. The use of the revised models is found to be acceptable.

The overall impact of cladding ballooning and assembly flow blockage on the response of the Beaver Valley Unit 2 fuel design to the loss-of-coolant accident is evaluated in Section 15.6.5 and is not reviewed further in this section.

4. Structural Damage from External Forces

It is stated in Section 4.2.3.4 of the FSAR that W has performed these analyses utilizing models described in WCAP-8236 (and WCAP-8288) and WCAP-9401 (and WCAP-9402). WCAP-9401 essentially augments the information presented in WCAP-8236 because both WCAP reports apply to similar assemblies. WCAP-9401 has been reviewed and approved (Rubenstein, April 23, 1981); therefore, these models are acceptable for these analyses.

It is unclear from the discussion of the grid analysis in Section 4.2.3.4 of the FSAR whether this analysis includes the combined LOCA and seismic loads using the square-root-of-sum-of-squares (SRSS) method (as per SRP Section 4.2, Appendix A) or if these loads are considered separately. Consequently, the use of combined LOCA and seismic loads using the SRSS method needs to be confirmed to satisfy SRP Section 4.2 guidelines.

Fuel assembly non-grid component stresses from combined LOCA and seismic loads have not been shown in the FSAR to remain below P(crit) as defined in SRP Section 4.2, Appendix A. These non-grid component forces must be provided by the applicant in order to enable us to complete our review.

4.2.4 Testing, Inspection, and Surveillance Plans

4.2.4.1 Testing and Inspection of New Fuel

As required by SRP Section 4.2, testing and inspection plans for new fuel should include verification of significant fuel design parameters.

While details of the manufacturer's testing and inspection programs should be documented in quality control reports, the programs for onsite inspection of new fuel and control assemblies after they have been delivered to the plant should also be described in the FSAR.

The W quality control program which will be applied to the Beaver Valley Unit 2 fuel is discussed in Section 4.2.4 of the FSAR and addresses fuel system components and parts, pellets, rod inspection, assemblies, other inspections, and process control. Fuel system components and parts inspection depends on the component parts and includes dimensional and visual examinations, audits of test reports, material certification, and nondestructive examinations. Pellet inspections, for example, are performed for dimensional characteristics such as diameter, density, length, and squareness of ends. Fuel rod, control rod, burnable poison, and source rod inspections reportedly consist of nondestructive examination techniques such as leak testing, weld inspection, and dimensional measurements. Process control procedures are described in detail. In addition, the applicant states in Section 4.2.4.4 of the FSAR that if any tests and inspections are to be performed by others on behalf of W, W will review and approve the quality control procedures, inspection plans, and so forth, to ensure that they are equivalent to the description provided in Sections 4.2.4.2 and 4.2.4.3 and are performed properly to meet all W requirements.

Based on the information provided in Section 4.2.4 of the FSAR and the commitment by W to ensure the acceptability of any tests and inspections performed by others on behalf of W, we conclude that the fuel testing and inspection program for new fuel is acceptable.

4.2.4.2 On-Line Fuel System Monitoring

The applicant shall provide information regarding the plant's on-line fuel rod failure detection methods to satisfy the guidelines described in Paragraph II.D.2 of SRP Section 4.2. The reactor coolant radiation monitors, which include high-and low-range off-line liquid monitors in the reactor coolant letdown line that can detect conditions that indicate fuel rod failure, are briefly mentioned in Sections 4.2.3.3 and 4.2.4.7 of the FSAR and are discussed in Sections 11.5.2.2 and 11.5.2.5.10 of the FSAR. The ability of the reactor coolant letdown radiation monitors to detect fuel rod failures needs to be confirmed along with the applicant's commitment to use these techniques to monitor failures as per SRP Section 4.2.

4.2.4.3 Postirradiation Surveillance

The W test (Eggleston 1978) and surveillance (Jones and Iorii, May 1982; Skaritka and Iorii, August 1983) programs to examine detailed aspects of the 17X17 fuel assembly are noted in Section 4.2.4.5 of the FSAR. In Section 4.2.4.6 of the FSAR, the applicant states that (a) postirradiation fuel inspections are routinely conducted during refueling, (b) these inspections include a qualitative visual examination of some discharged fuel assemblies from each refueling, (c) gross problems of structural integrity, fuel rod failure, rod bowing and crud deposition are identified, and (d) additional surveillance is provided if the visual examination identifies unusual behavior or if the plant instrumentation indicates gross fuel failures.

We conclude that the applicant has satisfied the guidelines described in Paragraph II.D.3 of SRP Section 4.2 regarding the need for postirradiation surveillance.

4.2.5 Evaluation Findings

The following have not yet been provided by the applicant.

1. Confirmation that the peak pellet design basis burnup of 53,000 MWd/MTU is consistent with the region discharge burnup of 33,000 MW/d/MTU (see Section 4.2.1).

2. Specification of the correct values for several parameters (e.g., fuel rod diameter and Zircaloy weight) in the description of and design drawings for Beaver Valley Unit 2 fuel (see Section 4.2.2).

3. Confirmation that the rod bowing analysis has been performed with latest approved correlation (see Section 4.2.3.1(6)).

4. Confirmation that the fuel rod internal pressure is consistent with WCAP-8963 (see Section 4.2.3.1(8)).

5. Confirmation that the predicted cladding collapse time exceeds the expected residence of the fuel (see Section 4.2.3.2(2)).

6. Confirmation that combined seismic and LOCA loads, using the SRSS method and a worst-case LOCA, are applied in calculating grid stresses (see Section 4.2.3.3(4)).

7. Fuel assembly non-grid component forces from combined seismic and LOCA loads have not been shown to meet SRP Section 4.2 guidelines (see Section 4.2.3.3(4)).

8. Confirmation of the ability of the reactor coolant letdown radiation monitors to detect fuel rod failures (see Section 4.2.4.2).

9. Commitment to use the on-line detection method to monitor fuel rod failures (see Section 4.2.4.2).

When the above are provided, we will be able to conclude that the Beaver Valley Unit 2 fuel has been designed so that (a) the fuel design limits will not be exceeded as a result of normal operation and AOOs, (b) fuel damage during postulated accidents would not be severe enough to prevent control rod insertion when it is required, and (c) coolability will always be maintained, even after severe postulated accidents, and thereby meets the related requirements of 10 CFR Part 50.46, 10 CFR Part 50, Appendix A, GDC 10, 27, and 35, 10 CFR Part 50, Appendix K, and 10 CFR Part 100. This conclusion is based on the following:

1. The applicant has provided sufficient evidence that these design objectives will be met based on operating experience, prototype testing, and analytical predictions. Those analytical predictions dealing with structural response, control rod ejection, and fuel densification have been performed in accordance with (1) the guidelines of Regulatory Guide 1.77, and methods that the staff has reviewed and found to be acceptable alternatives to Regulatory Guides 1.60 and 1.126, and (b) the guidelines for "Evaluation of Fuel Assembly Structural Response to Externally Applied Forces" in Appendix A to SRP Section 4.2.

2. The applicant has provided for testing and inspection of the fuel to ensure that it is within design tolerances at the time of core loadings. The applicant has made a commitment to perform on-line fuel failure monitoring and postirradiation surveillance to detect anomalies or confirm that the fuel has performed as expected.

Following satisfactory resolution of the open items we will be able to conclude that the applicant has described methods of adequately predicting fuel rod failures during postulated accidents so that radioactivity releases are not underestimated and thereby meets the related requirements of 10 CFR Part 100.

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