Docket No.: 50-412

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Mr. Earl J. Woolever, Vice President
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Dear Mr. Woolever:

Subject: Review of Beaver Valley Power Station, Unit 2

By letter dated March 1, 1984, we provided you with a copy of the draft SER for Beaver Valley, Unit 2. The SER presented the results of the staff's review of various technical areas and identified the open items for those areas. Subsequently, additional evaluations have been prepared for other technical areas.

The purpose of this letter is to provide you with an additional evaluation in the areas of core performance, materials engineering, quality assurance, reactor systems and emergency preparedness. Enclosure 1 is a list of the open items resulting from this review and Enclosure 2 presents the confirmatory items. Enclosure 3 is the staff evaluation which should be incorporated into the BVPS-2 draft SER

We request that you review the enclosures and provide responses to the issues identified in the evaluation. In accordance with the BVPS-2 review schedule, responses should be submitted to the NRC within one month of receipt of this letter. Should you have any questions concerning this request, please contact Licensing Project Managers Marilyn Ley (301) 492-7792 or Manny Licitra (301) 492-7200.

Sincerely,

/S/

George W. Knighton, Chief
Licensing Branch No. 3
Division of Licensing

Enclosure: As stated

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

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Enclosure 1

Additional Open Items (CPB, MTEB, QAB, RSB, EPB)
for BVPS-2 FSAR Review

(153) Analysis of combined LOCA and seismic loads (4.2.3.3)

(154) Testing and inspection of new fuel (4.2.4.1)

(155) On-line detection method to monitor fuel rod failures (4.2.4.2)

(156) Underclad cracking of forgings in reactor vessel (5.2.3, 5.3.1.2)

(157) Review of structures, systems and components under Quality Assurance Program (17.5)

(158) PORV setpoint values (5.2.2.2)

(159) RHR operation requirements outside control room (5.4.7.1)

(160) RHR overpressure protection system (5.4.7.3)

(161) Qualification of RHR pumps inside containment (5.4.7.4)

(162) Natural circulation test (5.4.7.5)

(163) Programs and procedures for containment sump operation (6.3.1)

(164) Review of off-site emergency plans (FEMA) (13.3.1)

(165) On-site emergency planning (13.3.2)

Enclosure 2

Additional Confirmatory Items (CPB, MTEB)
for BVPS-2 FSAR Review

(17) Peak pellet design basis (4.2.1)

(18) Specification of fuel parameters (4.2.2)

(19) Rod bowing analysis (4.2.3.1)

(20) Fuel rod internal pressure (4.2.3.1)

(21) Cladding collapse time (4.2.3.2)

(22) Use of austenitic stainless steels (4.5.1, 5.2.3)

Enclosure 3

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,

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[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. Overheating of Cladding

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. Overheating of Fuel Pellets

The design evaluation of the fuel centerline melt limit is performed with the W 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^{o}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 pre-pressurized 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 Fuel Coolability Evaluation

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. Fragmentation of Embrittled Cladding

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, W 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 and 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.

4.2.6 References

Anderson, T. M., Westinghouse, Letter No. NS-TMA-1760 to J. F. Stolz, USNRC, "Fuel Rod Bowing," April 19, 1978.

Anderson, T. M., Westinghouse, Letter No. NS-TMI-2410 to J. F. Stolz, USNRC, October 22, 1979.

Anderson, T. M., Westinghouse, Letter No. NS-TMA-2366 to J. R. Miller, USNRC, "Responses to Request Number 3 for Additional Information on WCAP-9500, NRC Letter From J. R. Miller to T. M. Anderson, August 15, 1980," January 12, 1981.

Anderson, T. M., Westinghouse, Letter No. NS-TMA-2436 to J. R. Miller, USNRC, "Responses to Questions on WCAP-9500, Section 4.2, Resulting from NRC/Westinghouse Meeting on March 19, 1981," April 21, 1981.

ANSI N-212-1974. American National Standard Nuclear Safety Criteria for the Design of Stationary Boiling Water Reactor Plants, American Nuclear Society, LaGrange Park, Illinois.

ANSI N18.2-1973. American Nuclear Standard Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants, American Nuclear Society, LaGrange Park, Illinois.

Appleby, W. K., et al., "Fluence and Temperature Dependence of Void Formation in Highly Irradiated Stainless Steels," page 156, Radiation-Induced Voids in Metals, USAEC Proceedings of the 1977 International Conference held at Albany, New York, June 9-11, 1971 (April 1972).

ASME Boiler and Pressure Vessel Code, Section III, "Rules for Construction of Nuclear Power Plant Components," American Society of Mechanical Engineers, New York, NY.

Bloom, E. E., "Nucleation and Growth of Voids in Stainless Steels During Fast-Neutron Irradiation," page 1, Radiation-Induced Voids in Metals, USAEC Proceedings in the 1977 International Conference held at Albany, New York, June 9-11, 1971 (April 1972).

Eggleston, F. T., Safety-Related Research and Development for Westinghouse Pressurized Water Reactors, Program Summaries--Winter 1977 - Summer 1978. WCAP-8768, Revision 2, 1978.

Jones, R. G. and Iorii, J. A., Operational Experience with Westinghouse Cores (Up to December 31, 1981). WCAP-8183, Revision 11, May 1982.

Leasburg, R. H., VEPCo, Letter to H. R. Denton, USNRC, "Fuel Assembly Guide Thimble Tube Wear Examination Report," March 1, 1982.

Meyer, R. O., USNRC, Memorandum to D. F. Ross, "Revised Coefficients for Interim Rod Bowing Analysis," March 2, 1978.

Miller, J., USNRC, Letter to E. P. Rahe, Westinghouse, December 1, 1981.

NUREG-0800, Revision 2, "Section 4.2, Fuel System Design," Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants - LWR Edition, U.S. Nuclear Regulatory Commission, July 1981 (formerly NUREG-75/087).

Petrick, N., SNUPPS, Letter to H. R. Denton, USNRC, August 31 and September 9, 1981.

Rubenstein, L. S., USNRC, Memorandum to R. L. Tedesco, USNRC, "Review of Westinghouse Optimized Fuel Assembly Topical Report (TAR-5254)," August 8, 1980.

Rubenstein, L. S., USNRC, Memorandum to R. L. Tedesco, USNRC, "Review of Topical Report WCAP-8278," March 19, 1981.

Rubenstein, L. S., USNRC, Memorandum to R. L. Tedesco, USNRC, "Review of Topical Report WCAP-9401," April 23, 1981.

Rubenstein, L. S., USNRC, Memorandum to R. L. Tedesco, USNRC, "Safety Evaluation Report on WCAP-9500," May 15, 1981.

Rubenstein, L. S., USNRC, Memorandum to T. M. Novak, USNRC, "Resolution of the Westinghouse Guide Thimble Tube Wear Issue," April 19, 1982.

Rubenstein, L. S., USNRC, Memorandum to R. L. Tedesco, USNRC, "Safety Evaluation of WCAP-8720 Addendum 1," June 30, 1982.

Rubenstein, L. S., USNRC, Memorandum to R. L. Tedesco, USNRC, "SSER Input for Callaway Concerning Hafnium Control Rod Surveillance," June 30, 1982.

Rubenstein, L. S., USNRC, Memorandum to T. M. Novak, USNRC, "Shearon Harris Draft Safety Evaluation Report," October 21, 1982.

Rubenstein, L. S., USNRC, Memorandum to T. M. Novak, USNRC, "SERs for Westinghouse, Combustion Engineering, Babcock & Wilcox, and Exxon Fuel Rod Bowing Topical Reports," October 25, 1982.

Salvatori, R., Westinghouse, Letter to D. Knuth, USNRC, 'Core Coolant and Rod Surface Temperature," January 4, 1973.

Skaritka, J., and J. A. Iorii, Operational Experience with Westinghouse Cores (Up to December 31, 1982). WCAP-8183, Revision 12, August 1983.

Stello, V., USNRC, Memorandum to R. DeYoung, "Evaluation of Westinghouse Report WCAP-8377, Revised Clad Flattening Model," January 14, 1975.

Stolz, J. F., USNRC, Letter to T. M. Anderson, Westinghouse, "Safety Evaluation of WCAP-8963," May 19, 1978.

Stolz, J. F., USNRC, Letter to T. M. Anderson, Westinghouse, "Safety Evaluation of WCAP-8720," February 9, 1979.

Tedesco, R. L., USNRC, Letter to T. M. Anderson, Westinghouse, "Acceptance for Referencing of Licensing Topical Report WCAP-9500," May 22, 1981.

WCAP-8236 (Proprietary) and WCAP-8288 (Nonproprietary). Gesinski, L. and Chiang, D., Safety Analysis of the 17X17 Fuel Assembly for Combined Seismic and Loss-of-Coolant Accident, 1973.

WCAP-8278 (Proprietary) and WCAP-8279 (Nonproprietary). Demario, E. E., Hydraulic Flow Test of the 17X17 Fuel Assembly, 1974.

WCAP-8377 (Proprietary) and WCAP-8381 (Nonproprietary). George, R. A.; Lee, Y. C.; and English, G. H., Revised Clad Flattening Model, 1974.

WCAP-8691, Revision 1 (Proprietary) and WCAP-8692, Revision 1 (Nonproprietary). Skaritka, J., et al., Fuel Rod Bow Evaluation, July 1979.

WCAP-8720 (Proprietary) and WCAP-8785 (Nonproprietary). Miller, J. V., ed., Improved Analytical Models Used in Westinghouse Fuel Rod Design Computations, 1976.

WCAP-8963 (Proprietary). Risher, D. H., et al., Safety Analysis for the Revised Fuel Rod Internal Pressure Design Basis, November 1976.

WCAP-9179, Revision 1 (Proprietary) and WCAP-9224 (Nonproprietary) and Appendix A, Hafnium. Beaumont, M. D., et al., Properties of Fuel and Core Component Materials, 1978.

WCAP-9401-P-A (Proprietary) and WCAP-9402-A (Proprietary). Beaumont, M. D., et al., Verification Testinghouse and Analysis of the 17X17 Optimized Fuel Assembly, August 1981.

WCAP-9500. Reference Core Report 17X17 Optimized Fuel Assembly, July 1979.

4.5 Reactor Materials

4.5.1 Control Rod Drive Structural Materials

The staff concludes that the control rod drive mechanism structural materials are generally acceptable and meet the requirements of General Design Criteria 1, 14, and 26 as well as 10 CFR Part 50, Section 50.55a. A confirmatory response from the applicant is required concerning the yield strength of austenitic stainless steels in these components.

This conclusion is based on the applicant having demonstrated that the properties of materials selected for the control rod drive mechanism components exposed to the reactor coolant satisfy Appendix I of Section III of the ASME Code, and Parts A, B, and C of Section II of the Code. The applicant should confirm conformance with the staff position that the yield strength of cold-worked austenitic stainless steels do not exceed 90,000 psi. Conformance to the recommendations of Regulatory Guide 1.85 is discussed in 5.2.1.2.

In addition, the controls imposed upon the austenitic stainless steel of the mechanisms satisfy, to the extent practical, the recommendations of Regulatory Guide 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal," and Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel." The alternative method of control of ferrite content by testing of the purchased material and the modification of testing procedures to evaluate weldments for stress corrosion cracking have been reviewed by the staff and are acceptable. The applicant has confirmed that the tempering temperatures and aging temperatures of heat treatable materials in the control rod drive mechanism are specified to eliminate the susceptability to stress corrosion cracking in reactor coolant. The fabrication and heat treatment practices performed provide assurance that stress corrosion cracking will not occur during the design life of the components. The compatibility of all materials used in the control rod system in contact with the reactor coolant satisfies the criteria of Articles NB-2160 and NB-3120 of Section III of the Code. Cleaning and cleanliness controls are in accordance to the extent practical with ANSI Standard N 45.2.1-1973, "Cleaning of Fluid Systems and Associated Components During Construction Phase of Nuclear Power Plants," and Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants." The applicant's use of oxygen saturated reactor grade water when flushing open systems and use of controlled disposable materials and standard cleaning methods to control contamination levels of harmful elements and their compounds is acceptable to the staff.

4.5.2 Reactor Internals Materials

The staff concludes that the materials used for the construction of the reactor internal and core support structure are acceptable and meet the requirements of General Design Criterion 1 and Section 50.55a of 10 CFR Part 50. The conclusion is based upon the following considerations:

The applicant has met the requirements of GDC 1 and Section 50.55a of 10 CFR Part 50 with respect to assuring that the design, fabrication, and testing of the materials used in the reactor internal and core support structure are of high quality standards and adequate for structural integrity. The controls imposed upon components constructed of austenitic stainless steel satisfy, to the extent practical, the recommendations of Regulatory Guide 1.31," Control of Ferrite Content in Stainless Steel Weld Metal," and Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel." Where the recommendations of these Regulatory Guides were not followed, the alternative approaches taken by the applicant have been reviewed by the staff and are acceptable (see 4.5.1).

The materials used for construction of components of the reactor internal and core support structure have been identified by specification and found to be in conformance with the requirements of NG-2000 of Section III and Parts A, B, and C of Section II of the ASME Code. Conformance to the recommendations of Regulatory Guide 1.85, "Code Case Acceptability ASME Section III Materials" is discussed in 5.2.1.2. As proven by extensive tests and satisfactory performance, the specified materials are compatible with the expected environment and corrosion is expected to be negligible. The controls imposed on the reactor coolant chemistry provide reasonable assurance that the reactor internal and core support structure will be adequately protected during operation from conditions which could lead to stress corrosion of the materials and loss of component structural integrity.

The material selection, fabrication practices, examination and testing procedures, and control practices performed in accordance to these recommendations provide reasonable assurance that the materials used for the reactor internal and core support structure are in a metallurgical condition to preclude inservice deterioration. Conformance with requirements of the ASME Code and the recommendations of the regulatory guides constitute an acceptable basis for meeting in part requirements of General Design Criterion 1 and Section 50.55a of 10 CFR Part 50.

5.2.3 Reactor Coolant Pressure Boundary Materials

The staff concludes that the plant design is generally acceptable and meets the requirements of General Design Criteria 1, 4, 14, 30, and 31 of Appendix A of 10 CFR Part 50; the requirements of Appendices B and G of 10 CFR Part 50; and the requirements of 50.55a of 10 CFR Part 50.55a of 10 CFR Part 50. Confirmation by the applicant of the staff position concerning the yield strength of austenitic stainless steels in the reactor coolant pressure boundary and that underclad cracking has not occurred in ASME SA-508 Class 2 forgings due to high heat input weld cladding is required. This conclusion is based on the staff's review of the FSAR.

The materials used for construction of components of the reactor coolant pressure boundary (RCPB) have been identified by specification and found to be in conformance with the requirements of Section III of the ASME Code. Compliance with the above Code provisions for materials specifications satisfies the quality standards requirements of GDC 1, GDC 30, and 55.55a.

The materials of construction of the RCPB exposed to the reactor coolant have been identified and all of the materials are compatible with the primary coolant water, which is chemically controlled in accordance with appropiate technical specifications. This compatibility has been proven by extensive testing and satisfactory performance. This includes satisfying, to the extent practical, the recommendations of Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel." Where the recommendations of the regulatory guide were not followed, the alternative approaches taken have been reviewed by the staff and are acceptable (see 4.5.1).

General corrosion of all materials in contact with reactor coolant is negligible, and accordingly, general corrosion is not of concern. Compatibility of the materials with the coolant and compliance with the Code provisions satisfy the requirements of GDC 4 relative to compatibility of components with environmental conditions.

The materials of construction for the RCPB are compatible with the thermal insulation used in these areas. The thermal insulation used on the RCPB is either the reflective stainless steel type or is made of nonmetallic compounded materials that meet most of the recommendations of Regulatory Guide 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steels." The use of standard commercial packaging with receipt inspection for damage, as an alternative approach to the special packaging recommendations in the guide, is acceptable to the staff. Conformance with the above recommendations satisfies the requirements of GDC 14 and GDC 31 relative to prevention of failure of the RCPB.

The ferritic steel tubular products and the tubular products fabricated from austenitic stainless steel have been found to be acceptable by non-destructive examinations in accordance with provisions of the ASME Code, Section III. Compliance with these Code requirements satisfies the quality standards requirements of GDC 1, GDC 30, and 50.55a.

The fracture toughness tests required by the ASME Code, augmented by Appendix G, 10 CFR Part 50, provide reasonable assurance that adequate safety margins against nonductile behavior or rapidly propagating fracture can be established for all pressure retaining components of the reactor coolant pressure boundary. The use of Appendix G of the ASME Code, Section III, and the results of fracture toughness tests performed in accordance with the Code and NRC regulations in establishing safe operating procedures, provide adequate safety margins during operating, testing, maintenance, and postulated accident conditions. Compliance with these Code provisions and NRC regulations satisfies the requirements of GDC 31 and 50.55a regarding prevention of fracture of the RCPB.

The applicant has taken alternative approaches to the recommendations of Regulatory Guide 1.50, "Control of Preheat Temperature for Welding Low Alloy Steels." The alternative approaches taken by the applicant are that welding procedures are qualified within the preheat temperature range (minimum limit plus 50°F) rather than at the minimum preheat temperature, and preheat temperatures are maintained for an extended period of time rather than preheat temperatures maintained until the start of post-weld heat treatment. The staff concludes that these alternative approaches are adequate to prevent hydrogen cracking (the concern of this regulatory guide) and will not cause other hazards. Accordingly, the staff accepts these alternative approaches.

The controls used provide reasonable assurance that cracking of components made from low-alloy steels will not occur during fabrication. If cracking does occur, the required Code inspections should detect such flaws. These controls satisfy the quality standards requirements of GDC 1, GDC 30, and 50.55a.

Regulatory Guide 1.34, "Control of Electroslag Weld Properties," is not applicable because the electroslag welding process was not used on RCPB components.

The controls imposed on welding ferritic and austenitic steels under conditions of limited accessibility satisfy, to the extent practical, the recommendations of Regulatory Guide 1.71, "Welder Qualification for Areas of Limited Accessibility." The applicant's contractors maintain close supervisory control of the welders and reoccurrence of welding situations in production are adequate to assure that the most skilled welders are used in areas of limited accessibility. The staff concludes, that as such welds are inspected, qualification of the welders making acceptable welds occurs automatically under the Code. These controls satisfy the quality standards requirements of GDC 1, GDC 50, and 50.55a. The controls imposed on weld cladding of low-alloy steel components by austenitic stainless steel are not in accordnce with the recommendations of Regulatory Guide 1.43, "Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components." The applicant must provide assurance that the practices used have not resulted in underclad cracking of ASME SA-508 Class 2 forgings.

The applicant has not addressed the staff position limiting RCPB components constructed of austenitic stainless steel to a maximum yield strength of 90,000 psi.

The controls to avoid stress corrosion cracking in reactor coolant pressure boundary components constructed of austenitic stainless steels satisfy, to the extent practical, the recommendations of Regulatory Guides 1.44, "Control of the Use of Sensitized Stainless Steel," and 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems Associated Components of Water Coolant Nuclear Plants." The alternate approaches taken by the applicant were reviewed by the staff and are acceptable (see 4.5.1).

The controls followed during material selection, fabrication, examination, protection, sensitization, and contamination, provide reasonable assurance that the RCPB component of austenitic stainless steels are in a metallurgical condition that minimizes susceptibility to stress corrosion cracking during service. These controls meet the requirements of GDC 4 relative to compatibility of components with environmental conditions and requirements of GDC 14 relative to prevention of leakage and failure of the RCPB.

The controls imposed during welding of austenitic stainless steels in the RCPB satisfy, to the extent practical, the recommendations of Regulatory Guides 1.31, 1.34 and 1.71. The alternate approaches taken by the applicant were reviewed by the staff and are acceptable (see 4.5.1).

These controls provide reasonable assurance that welded components of austenitic stainless steel did not develop microfissures during welding and have high structural integrity. These controls meet the quality standards requirements of GDC 1, GDC 30, and 50.55a and satisfy the requirements of GDC 14 relative to prevention of leakage and failure of the RCPB.

5.3 Reactor Vessel

5.3.1 Reactor Vessel Materials ~~(Materials and Fabrication)~~

5.3.1.2 Compliance of Reactor Vessel Materials, Fabrication, and Nondestructive Examination Methods to SRP 5.3.1

The staff concludes that the reactor vessel materials are generally acceptable and meet the requirements of General Design Criteria 1, 4, 14, 30, 31, and 32 of Appendix A of 10 CFR Part 50; the material testing and monitoring requirements of Appendices B, G, and H of 10 CFR Part 50; and the requirements of 50.55a of 10 CFR Part 50. Additional input by the applicant is required concerning examination results indicating that underclad cracking of ASME SA-508 Class 2 forgings in the reactor vessel has not occurred. This conclusion is based on the following:

The materials used for construction of the reactor vessel and its appurtenances have been identified by specification and found to be in conformance with Section III of the ASME Code. Special requirements of the applicant with regards to control of residual elements have been identified and are considered acceptable. Compliance with the above Code provisions for material specifications satisfies the quality standards requirements of GDC 1, GDC 30, and 50.55a.

Ordinary processes were used for the manufacture, fabrication, welding, and nondestructive examinations of the reactor vessel and its appurtenances. Nondestructive examinations in addition to Code requirements were also performed. Since certification has been made by the applicant that the requirements of Section III of the ASME Code have been complied with, the processes and examintions used are condsidered acceptable. Compliance with these Code provisions meets the quality standards requirements of GDC 1, GDC 30, and 50.55a.

When welding components of ferritic steels, Code controls are supplemented by conformance with the recommendations of regulatory guides as follows:

a. The controls imposed on welding preheat temperatures are in comformance to the extent practical with the recommendations of Regulatory Guide 1.50, "Control of Preheat Temperature for Welding of Low-Alloy Steel." The alternative approaches taken by the applicant were reviewed by the staff and are acceptable (see 5.2.3). These controls provide reasonable assurance that cracking of components made from low-alloy steels did not occur during fabrication and minimize the potential for subsequent cracking. These controls also satisfy the quality standards requirements of GDC 1, GDC 30, and 50.55a.

b. Regulatory Guide 1.34, "Control of Electroslag Weld Properties," is not applicable because this process is not used in reactor vessel fabrication.

c. The controls imposed during weld cladding of ferritic steel components are not in conformance with the recommendations of Regulatory Guide 1.43, "Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components." The applicant must provide assurance that underclad cracking did not occur during the weld cladding of ASME SA-508 Class 2 forgings.

When welding components of austenitic stainless steels, Code controls are supplemented by conformance with the recommendations of regulatory guides as follows:

a. The controls imposed on delta ferrite in austenitic stainless steel welds satisfy to the extent practical, the recommendations of Regulatory Guide 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal." The alternate approaches taken by the applicant have been reviewed by the staff and are acceptable (see 4.5.1). The controls used provide reasonable assuance that the welds do not contain micro-cracks. These controls also satisfy the quality standards requirement of GDC 1, GDC 30, and 50.55a and the requirement of GDC 14 regarding fabrication to prevent rapid propagating failure of the RCPB.

b. Regualtory Guide 1.34, "Control of Electroslag Weld Properties" is not applicable because this process is not used in reactor vessel fabrication.

The controls (during, all stages of welding) to avoid contamination and sensitization that could cause stress corrosion cracking in austenitic stainless steels conform with the recommendations of regulatory guides as follows:

a. The controls to avoid contamination and excessive sensitization of austenitic stainless steel satisfy, to the extent practical, the recommendations of Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel." The alternative approaches taken by the applicant have been reviewed by the staff and are acceptable (see 4.5.1). The controls used provide assurance that welded components were not be contaminated or excessively sensitized prior to and during the welding process. These controls satisfy the quality standards requirement of GDC 1, GDC 30, and 50.55a and the GDC 4 requirement relative to material compatibility.

b. The controls regarding onsite cleaning and cleanliness controls of austenitic stainless steel are in conformance with the recommendations of Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants" or the applicant's alternative approaches are acceptable to the staff as discussed in 4.5.1. These controls provide assurance that austenitic stainless steel components were properly cleaned onsite and satisfy Appendix B of 10 CFR Part 50 regarding controls for onsite cleaning of materials and components.

Integrity of the reactor vessel studs and fasteners is assured by conformance to the extent practical with the recommendations of Regulatory Guide 1.65, "Materials and Inspections for Reactor Vessel Closure Studs." The applicants alternative approaches of (a), using a modified SA-540, Grade B-24 for closure stud material which is allowed by Code Case 1605 and (b), not specifying a maximum ultimate tensile strength and relying on the bolting material's low alloy steel chemistry, heat treatment and toughness requirements to control ultimate tensile strength are acceptable to the staff. Compliance with these recommendations and the applicants alternative approaches satisfy the the quality standards requirements of GDC 1, GDC 30, and §50.55a; the prevention of fracture of the RCPB requirement of GDC 31; and the requirements of Appendix G, 10 CFR Part 50, as detailed in the provisions of the ASME Code, Sections II and III.

5.4.2.1 Steam Generator Materials

The staff concludes that the steam generator materials specified are acceptable and meet the requirements of GDC 1, 14, 15, and 31, and Appendix B to 10 CFR Part 50. This conclusion is based on the following:

1. The applicant has met the requirements of GDC 1 with respect to codes and standards by assuring that the materials selected for use in Class 1 and Class 2 components were fabricated and inspected in conformance with codes, standards, and specifications acceptable to the staff. Welding qualification, fabrication, and inspection during manufacture and assembly of the steam generators were done in conformance with the requirements of Section III and IX of the ASME Code.

2. The requirements of GDC 14 and 15 have been met to assure that

the reactor coolant boundary and associated auxiliary systems have been designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapid failure and of gross rupture, during normal operation and anticipated operational occurrences.

The primary side of the steam generator is designed and fabricated to comply with ASME Class 1 criteria as required by the staff. The secondary side pressure boundary parts of the steam generator is designed, manufactured, and tested to ASME Class 2 Code.

The crevice between the tubesheet and the inserted tube is minimal because the tube was expanded to the full depth of insertion of the tube in the tubesheet. The tube expansion and subsequent positive contact pressure between the tube and the tubesheet preclude a buildup of impurities from forming in the crevice region and reduce the probability of crevice boiling.

3. The requirements of GDC 31 have been met with respect to the fracture toughness of the ferritic materials since the pressure boundary materials of ASME Class 1 components of the steam generators will comply with the fracture toughness requirements and tests of Subarticle NB-2300 of Section III of the Code.

The materials of the ASME Class 2 components of the steam generators will comply with the fracture toughness requirements of Subarticle NC-2300 of Section III of the Code.

4. The requirements of Appendix B of 10 CFR Part 50 have been met since the onsite cleaning and cleanliness controls during fabrication conform to the recommendations of Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants" or the applicant's alternative approaches are acceptable to the staff as discussed in 4.5.1. The controls placed on the secondary coolant chemistry are in agreement with staff technical positions.

Reasonable assurance of the satisfactory performance of the steam generator tubing and other generator materials is provided by the design provisions and the manufacturing requirements of the ASME Code and rigorous secondary water monitoring and control. The controls described above combined with conformance with applicable codes, standards, staff positions, and regulatory guides constitute an acceptable basis for meeting in part the requirements of General Design Criteria 1, 14, 15, and 31, and Appendix B, 10 CFR Part 50.

6.1.1 Engineered Safety Features Materials

The staff concludes that the engineered safety features materials specified are acceptable and meet the requirements of GDC 1, 4, 14, 31, 35, and 41 of Appendix A of 10 CFR Part 50; Appendix B of 10 CFR Part 50, and 10 CFR Part 50, 50.55a. This conclusion is based on the following:

1. General Design Criteria 1, 14, and 31, and 10 CFR Part 50, 50.55a have been met with respect to assuring an extremely low probability of leakage, of rapidly propagating failure, and of gross rupture. This is shown since the materials selected for the engineered safety features satisfy Appendix I of Section III of the ASME Code, and Parts A, B, and C of Section II of the Code, and the staff position that the yield strength of cold-worked stainless steels shall be less than 90,000 psi.

   In this time frame, the Code allowed waiving of impact testing of Class 2 and 3. However, based upon the results of impact testing by other applicants of the same specification steels, and correlations of the metallurgical characterization of these steels with the fracture toughness data presented in NUREG-0577, we conclude that the fracture toughness properties of the ferritic materials in the engineered safety features have adequate margins against the possibility of nonductile behavior and rapidly propagating fracture.

The controls on the use and fabrication of the austenitic stainless steel of the systems satisfy the requirements of Regulatory Guide 1.31, "Control of Ferrite Content of Stainless Steel Weld Metal," and Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel." The alternate approaches taken by the applicant have been reviewed and are acceptable to the staff (see 4.5.1). Fabrication and heat treatment practices performed accordingly provide assurance that the probability of stress corrosion cracking will be reduced during the postulated accident time interval.

Conformance with the Codes and Regulatory Guides and with the staff positions mentioned above, constitute an acceptable basis for meeting requirements the of General Design Criteria 1, 4, 14, 35, 41; Appendix B to to 10 CFR part 50, and 10 CFR Part 50, §50.55a, in which the systems are to be designed, fabricated, and erected so that the systems can perform their functions as required.

2. General Design Criteria 1, 14, 31, and Appendix B to 10 CFR Part 50 have been meet with respect to assuring that the RCPB and associated auxiliary systems have an extremely low probability of leakage, of rapidly propagating failure, and of gross rupture. The controls placed on concentrations of leachable impurities in nonmetallic thermal insulation used on components of the engineered safety features are in accordance with the recommendations of Regulatory Guide 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steels" or the applicant's alternative approaches are acceptable to the staff as discussed in 5.2.3. Compliance with the recommendations of Regulatory Guide 1.36 forms a basis for meeting the requirements of GDC 1, 14, and 31.

   Protective coating systems are discussed in 6.1.2.

3. The requirements of GDC 4, 35, 41 and Appendix B to 10 CFR Part 50 have been met with respect to compatibility of ESF components with environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss of cooling accidents.

   The controls on the pH and chemistry of the reactor containment sprays and the emergency core cooling water following a loss-of-coolant or design basis accident are adequate to reduce the probability of stress corrosion cracking of austenitic stainless steel components and welds of the engineered safety features systems in containment throughout the duration of the postulated accident to completion of cleanup.

Also, the controls of the pH of the sprays and cooling water, in conjunction with controls on selection of containment materials, are in accordance with Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident," and provide assurance that the sprays and cooling water will not give rise to excessive hydrogen gas evolution resulting from corrosion of containment metal or cause serious deterioration of the materials in containment.

The controls placed upon component and system cleaning are in accordance with the recommendations of Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants" or the applicant's alternative approaches have been reviewed and approved by the staff as discussed in 4.5.1. These controls provide a basis for the finding that the components and systems have been protected against damage or deterioration by contaminants as stated in the cleaning requirements of Appendix B, 10 CFR Part 50.

10.3.6 Main Steam and Feedwater Materials

The staff concludes that the main steam and feedwater system materials are acceptable and meet the relevant requirements of 10 CFR Part 50, 50.55a, General Design Criteria 1, and Appendix B to 10 CFR Part 50. This conclusion is based on the following:

The applicant selected materials for Class 2 and 3 components of the steam and feedwater systems that satisfy Appendix I of Section III of the ASME Boiler and Pressure Vessel Code, and meet the requirements of Parts A, B, and C of Secton II of the Code. Conformance to the recommendations of Regulatory Guide 1.85 is discussed in 5.2.1.2.

In this time frame, the Code allowed waiving of impact testing of main steam and feedwater materials. However, based upon the results of impact testing by other applicants of the same specification steels, and correlations of the metallurgical characterization of these steels with the fracture toughness data presented in NUREG-0577, we conclude that the fracture toughness properties of the ferritic materials in the main steam and feedwater systems have adequate safety margins against the possibility of nonductile behavior and rapidly propagating fracture.

The applicant has satisfied to the extent practical, the recommendations of Regulatory Guide 1.71, "Welder Qualification for Areas of Limited Accessibility" by meeting the regulatory positions in Regulatory Guide 1.71 or by meeting alternative approaches which the staff has reviewed and found to be acceptable (see 4.5.1). The onsite cleaning and cleanliness controls during fabrication satisfy the positions given in Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants," and the requirements of ANSI Standard N 45.2.1-1973, "Cleaning of Fluid Systems and Associated Components During Construction Phase of Nuclear Power Plants" or the applicant alternative approaches have been reviewed and are acceptable to the staff as discussed on 4.5.1.

17.0 Quality Assurance

17.1 General

The description of the quality assurance (QA) program for the operations phase of the Beaver Valley Power Station - Unit 2 (BV-2) is contained in Chapter 17 of the FSAR. Our evaluation of this QA program is based on a review of this information and discussions with representatives from Duquesne Light Company (DLC) and the NRC Region I Office. We assessed DLC's QA program for the operations phase to determine if it complies with the requirements of 10 CFR 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," and the Standard Review Plan, Section 17.2, Rev. 2, dated July 1981, "Quality Assurance During the Operations Phase." (NUREG-0800)

17.2 Organization

The DLC's organization responsible for the operation of BV-2 is shown in Figure 17.1 and consists of five divisions which are the Nuclear Division, the Nuclear Construction Division, the Engineering and Construction Division, the Operations Division and the General Services Division. Each of these divisions is headed up by a Vice President who with the exception of the General Services Division reports directly to the President of Duquesne Light Company who in turn reports to the Chairman of the Board. The Vice President of General Services reports directly to the Chairman of the Board. The General Services Division has overall responsibility for the procurement of material and equipment as requested by either the Nuclear Division or the Engineering and Construction Division. The Operations Division is responsible for providing testing and maintenance support to BV-2 for specified maintenance activities. The Engineering and Construction Division is responsible for design changes and engineering during the operation of BV-2.

The Nuclear Division is directly responsible for the safe and efficient operation of BV-2; for the development and implementation of the Quality Control (QC), Radiological Control and Environmental Surveillance Programs; for training, administrative services and plant security; and for engineering activities. Within this division is the Operational Quality Control (OQC) organization which is divided into three sections. The OQC Maintenance Section is responsible for implementing the quality requirements of plant maintenance, surveillance activities, and maintenance related nondestructive examination. The

OQC Inservice Inspection Section is responsible for implementing the quality requirements of inservice inspection and control of nondestructive examination procedures. The OQC Refueling and Modifications Section is responsible for implementing the quality requirements of refueling, new fuel receipt and modification activities.

The Nuclear Construction Division is responsible for the construction and modifications activities associated with BV-2. This Division, through the QA Department, is also responsible for establishing, managing, and measuring the overall effectiveness of the QA programs for construction and operations.

The Manager of the QA Department has reporting to him a QA Director for Operations and a QA Director for Design, Construction, and Procurement. The QA Director, Operations, is responsible for assuring that the fabrication and installation of fuel, operation, testing, and maintenance, is performed in accordance with the DLC QA program. The QA Director for Design, Construction and Procurement is responsible for assuring that design, engineering, procurement, construction and modification activities during both the construction and operation phases are performed in accordance with DLC QA program.

The QA Manager has the authority to report quality matters to any level necessary within DLC in order to establish effective corrective action. The QA and QC personnel have sufficient authority and organizational freedom from the pressures of cost and schedules to identify quality problems; initiate, recommend or provide solutions to quality problems through designated channels; verify implementation of solutions to quality problems and control further processing, delivery or installation of nonconforming items until proper dispositioning has occurred.

17.3 Quality Assurance Program

The QA program for the operation of BV-2 is described in Chapter 17 of the FSAR and is implemented by means of written policies, procedures, and instructions. DLC has committed its QA program for the operations phase to be in compliance with the provisions of the NRC Regulatory Guides listed in Table 1.

DLC QA program requires that implementing procedures and instructions contain detailed controls for (1) translating codes, standards, regulatory requirements, technical specifications, engineering, and process requirements into drawings and specifications, procedures and instructions; (2) developing, reviewing, and approving procurement documents, including changes; (3) prescribing quality-related activities by documents, instructions, procedures, drawings, and specifications; (4) issuing and distributing approved documents; (5) purchasing items and services; (6) identifying materials, parts, and components; (7) performing special processes; (8) inspecting and/or testing materials, equipment, processes or services; (9) calibrating and maintaining measuring and test equipment; (10) handling, storing, and shipping of items; (11) identifying the inspection, test, and operating status of items; (12) identifying and dispositioning nonconforming items; (13) correcting conditions adverse to quality; (14) preparing and maintaining QA records; and (15) auditing of activities which affect quality.

The indoctrination and training program assures that personnel performing activities affecting quality are knowledgeable and that they have competence and skill in the performance of their quality-related activities. It also provides for retraining of personnel performing activities affecting quality.

Quality is verified through checking, review, surveillance, inspection, testing, and audit of quality-related activities. The QA program requires that quality verification be performed by individuals who are not directly responsible for performing the quality-related activities. Inspections are performed by qualified personnel in accordance with procedures instructions, approved by the QA/QC organization.

Audits are performed in accordance with pre-established written checklists by qualified personnel not having direct responsibilities in the areas being audited. Periodic audits will be performed to evaluate all aspects of the QA program including the effectiveness of the QA program implementation. The QA program requires the review of audit results by the person having responsibility in the area audited and to determine and take corrective action where necessary. Follow-up audits are performed to determine that nonconformances and deficiencies are effectively corrected and that the corrective action precludes repetitve occurrences.

17.4 Conclusion

Based on our detailed review and evaluation of the QA program description contained in Chapter 17 of the FSAR for BV-2 we conclude that:

(1) The organizations and persons performing QA functions appear to have the required independence and authority to effectively carry out the QA program without undue influence from those directly responsible for cost and schedules.

(2) The QA program with the exception of the outstanding issue described in 17.5 of this SER, describes requirements, procedures, and controls that, when properly implemented, comply with the requirements of Appendix B to 10 CFR Part 50 and with the acceptance criteria contained in Section 17.2 of the Standard Review Plan (Revisions 2).

Accordingly, the staff concludes that DLC's description of the QA program, with the exception of the outstanding issue noted below, is in compliance with applicable NRC regulations.

17.5 Outstanding Quality Assurance Issue

The NRR staff is currently evaluating those structures, systems, and components which are under the control of the QA Program. A supplement to the SER will be provided after completion of this review.

TABLE 1

REGULATORY GUIDANCE APPLICABLE TO

QUALITY ASSURANCE PROGRAM

1. Regulatory Guide 1.30, "Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electrical Equipment," (8/11/72).

2. Regulatory Guide 1.33-Rev. 2, "Quality Assurance Program Requirements (Operation)," (2/78).

3. Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluids Systems and Associated Components of Water-Cooled Nuclear Power Plants," (3/16/73).

4. Regulatory Guide 1.38-Rev. 2, "Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants," (5/77).

5. Regulatory Guide 1.39-Rev. 2, "Housekeeping Requirements for Water-Cooled Nuclear Power Plants," (9/77).

6. Regulatory Guide 1.58-Rev. 1, "Qualification of Nuclear Power Plant Inspection, Examination, and Testing Personnel," (9/80).

7. Regulatory Guide 1.64-Rev. 2, "Quality Assurance Requirements for the Design of Nuclear Power Plants," (6/76).

8. Regulatory Guide 1.74, "Quality Assurance Terms and Definitions," (2/74).

9. Regulatory Guide 1.88-Rev. 2, "Collection, Storage, and Maintenance of Nuclear Power Plant Quality Assurance Records," (10/76).

10. Regulatory Guide 1.94-Rev. 1, "Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants," (4/76).

11. Regulatory Guide 1.116-Rev. 0-R, "Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems," (5/77).

12. Regulatory Guide 1.123-Rev. 1, "Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants," (7/77).

13. Regulatory Guide 1.144-Rev. 1, "Auditing Quality Assurance Programs for Nuclear Power Plants," (9/80).

14. Regulatory Guide 1.146, "Qualification of Quality Assurance Program Audit Personnel for Nuclear Power Plants," (8/80).

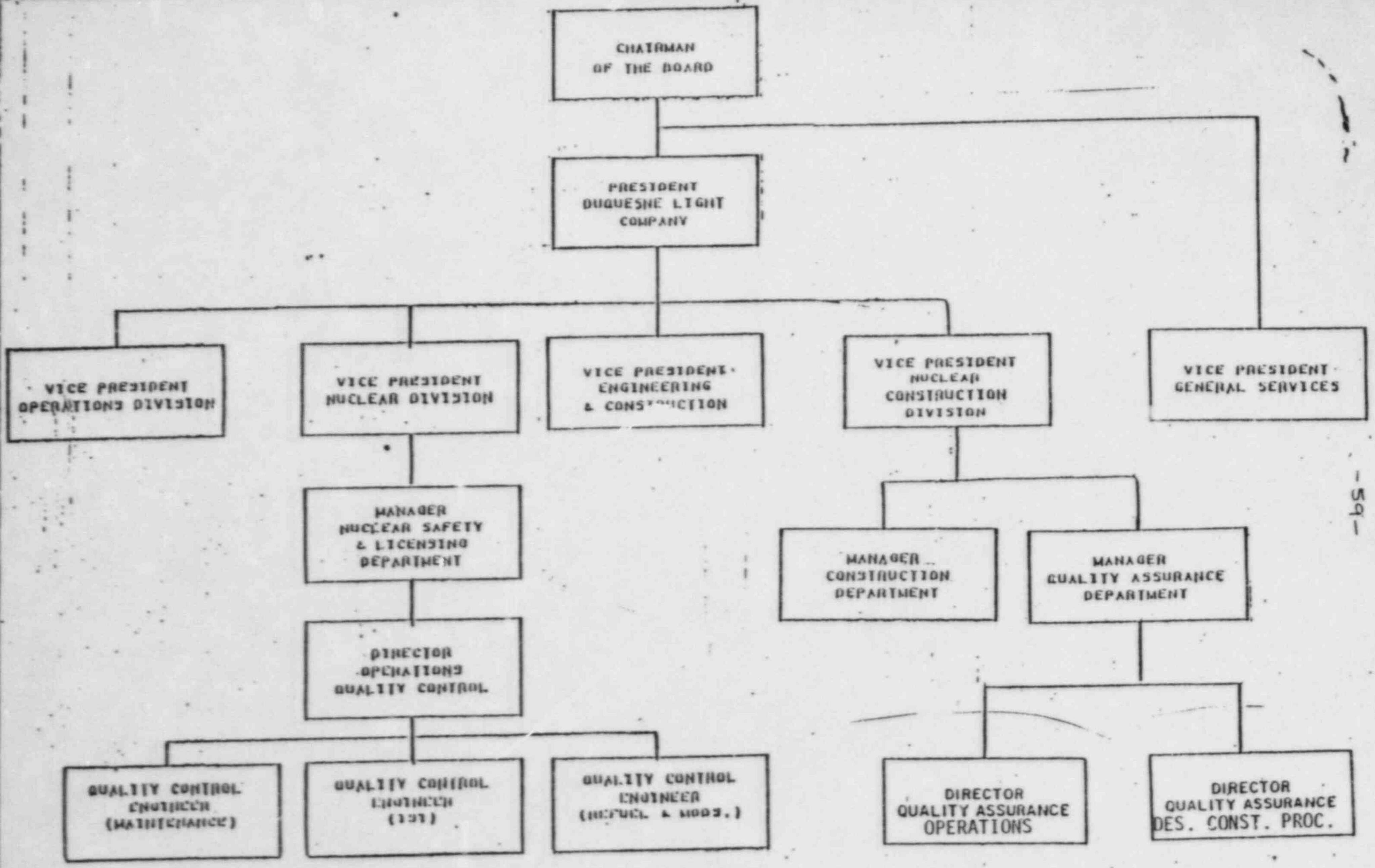

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FIGURE 17.1
ORGANIZATION STRUCTURE
BEAVER VALLEY POWER STATION-UNIT 2

5 REACTOR COOLANT SYSTEM

5.1 Summary Description

5.2 Integrity of Reactor Coolant Pressure Boundary

5.2.1 Compliance with ASME Code and Code Cases

5.2.2 Overpressure Protection

Overpressure protection for Beaver Valley Unit 2 has been reviewed in accordance with SRP 5.2.2 (NUREG-0800). Conformance with the acceptance criteria, except as noted, formed the basis for the staff's conclusion that the design of the facility for overpressure protection is acceptable.

The reactor coolant pressure boundary (RCPB) is protected from overpressurization by three safety relief valves and three power-operated relief valves in combination with the reactor protection system and operating procedures. This combination of features provides overpressurization protection in accordance with the criteria of GDC 15; the ASME Code, Section III; and 10 CFR 50, Appendix G. These criteria ensure RCPB overpressure protection for both power operation and low temperature operation (startup and shutdown). Following is a discussion of overpressure protection for each mode of operation.

5.2.2.1 Overpressure Protection During Power Operation

Overpressure protection during power operation is provided by the pressurizer spray system, three power-operated relief valves (PORVs), and three spring-loaded safety relief valves (SRVs), all of which are connected to the pressurizer.

The pressurizer spray system is designed to maintain the reactor coolant system (RCS) pressure below the PORV relief setpoint of 2335 psig during normal design transients.

The PORVs are sized to prevent actuation of a high pressurizer pressure reactor trip at 2410 psig for all design transients up to and including the design step load decrease with steam dump. The PORVs also limit undesirable openings of the SRVs.

The SRVs provide the final overpressure protection during power operation.

The PORVs and SRVs are both safety grade, and they are designed in accordance with ASME Code, Section III. Periodic testing and inspection are performed in accordance with Section XI of the Code. In FSAR Chapter 14 the applicant states that the safety relief valves will be checked and adjusted as a prerequisite to the initial test program in accordance with RG 1.68 Revision 2. In response to NUREG-0737 Item II.D.1, the applicant states that valves and piping configurations, similar to those at Beaver Valley Unit 2, have been tested in the Electric Power Research Institute (EPRI) safety and relief valve test program. The evaluation of the applicant's compliance with II.D.1 is included in SER Section 3.9.3. In response to NUREG-0737 Item II.D.3, the applicant states that the PORV status and safety valve status will be indicated in the control room. The evaluation of the applicant's compliance with II.D.3 is included in SER Section 7.5.

Each SRV has a relieving capacity of 345,000 pounds of saturated steam per hour at 2485 psig. Each PORV has a relieving capacity of 210,000 pounds of saturated steam per hour at 2335 psig. The combined capacity of two of these three safety valves is adequate to prevent the pressurizer pressure from exceeding the ASME Boiler and Pressure Vessel Code, Section III limit of 110% design pressure following the worst reactor coolant system pressure transient. This is identified to be a complete loss of steam load from full power without a direct reactor trip and with concurrent loss of main feedwater. This event was analyzed with no credit taken for the automatic steam dump system, automatic rod control, or auxiliary feedwater and with no credit taken for signals generated by a turbine trip, which would normally trip the reactor. The reactor is assumed to be tripped by high pressurizer pressure, overtemperature ΔT, or high pressurizer water level signals. The analysis for this case takes no credit for the pressurizer spray system, the pressurizer PORVs, or steam dump, but it does take credit for the operation of the steam generator safety valves.

There are five of these Code safety valves in each of the three secondary loops. They have a combined relieving capacity of 13.4×10^6 pounds of steam per hour with one of the valves stuck closed. This is fifteen percent more than the rated capacity of 11.6×10^6 lb/hr and is sufficient relieving capacity for the secondary system.

The above analyses were performed with a full-plant simulation. This included the reactor coolant system with an explicit reactor vessel, hot leg, pressurizer with an explicit surge line, primary side of the steam generator, reactor coolant pump, cold leg, and the secondary side of the steam generator. These were modeled using the LOFTRAN digital computer program, which has been reviewed by the staff and found acceptable.

5.2.2.2 Overpressure Protection During Low-Temperature Operation

The criteria for overpressure protection during low-temperature operation of the plant are in BTP RSB 5-2.

Low-temperature overpressure protection is primarily provided by two of the three pressurizer PORVs. These two have their opening setpoints automatically adjusted as a function of reactor coolant temperature. The reactor coolant temperature measurements will be auctioneered to obtain the lowest value. This temperature will be translated into a PORV setpoint curve that will adequately account for the lag in the temperature change of the reactor vessel and for possible single failures in the auctioneering system, so the system pressure will always be below the maximum allowable pressure. This PORV setpoint curve shall be periodically updated, as shall be specified in the bases for the technical specifications, to ensure that the stress intensity factors for the reactor vessel at any time in life are lower than the reference stress intensity factors as specified in 10 CFR 50, Appendix G.

The system logic will first annunciate at a predetermined low RCS temperature to alert the operator to arm the system. Another alarm on the main control board will annunciate whenever the measured pressure approaches, by a predetermined amount, the reference pressure. On further increase of the measured pressure, an actuation signal will be transmitted to the PORVs to mitigate the pressure transient.

The applicant has performed low-temperature overpressure transient analyses to determine the maximum pressure for the postulated worst case mass input and heat input events. The mass input transient analysis was performed assuming the inadvertent actuation of a high head safety injection pump, which pressurizes the RCS. The heat input analysis was performed for an incorrect reactor coolant pump start assuming that the RCS was water solid at the initiation of the event and that a 50°F mismatch existed between the RCS (250°F) and the secondary side of the steam generators (300°F). These temperatures were assumed because at lower temperatures the mass input case is limiting. The results of these analyses show that the allowable limits will not be exceeded. The applicant will provide PORV setpoint values later, and the staff will report its evaluation of these in a supplement to this SER.

An acceptance criterion for Item II.G.1 of NUREG-0737 is that the PORVs and associated block valves have safety grade emergency power supplies. Section 8.3 of the SER provides a discussion of Beaver Valley Unit 2's compliance with this criterion.

As a backup to the low-temperature overpressure protection system, both inlet lines to the residual heat removal (RHR) system have a pressure relief valve, which is designed to relieve the combined flow of two charging pumps (i.e., high head safety injection pumps) at the set pressure of the relief valves. These RHR relief valves provide overpressure protection after the RHR system is put into operation and the RHR suction isolation valves are open at an RCS pressure of less than 425 psig.

Assuming a single failure of one of the two PORVs, and taking no credit for the RHR system relief valves, the low temperature overpressure protection system can relieve the capacity of only one HHSI/charging pump and maintain pressure below the Appendix G limits. Thus operating procedures will require the removal of power from all HHSI/charging pumps that are not required to be operable. To prevent an accidental overpressurization by an accumulator discharge, operating procedures will stipulate that the accumulator isolation valves shall be closed when the RCS pressure is below the safety injection (SI) unblock set point, and that after they are closed their operating power shall be removed. To prevent overpressurization due to an excessive temperature differential between the RCS and an isolated steam generator, there will also be restrictions on the conditions under which a reactor coolant pump may be started. We will require technical specifications on these three items.

5.2.2.3 Conclusions

Subject to the generation of a conservative PORV setpoint curve and appropriate Technical Specifications, the staff concludes that the overpressure protection system meets the relevant criteria of GDC 15 and is, therefore, acceptable. Conformance to Appendix G to 10 CFR 50 criteria will be confirmed when the PORV setpoint curve is found acceptable. This conclusion is based on the following:

The overpressure protection system prevents overpressurization of the RCPB under the most severe transients and limits reactor pressure during normal operational transients. Overpressurization protection is provided by three safety valves. These valves discharge to the pressurizer relief tank through a common header from the pressurizer. The safety and power-operated relief valves in the primary system, in conjunction with the steam generator safety and atmospheric steam dump valves in the secondary system, and the reactor protection system, will protect the primary system against overpressure.

The peak primary system pressure following the worst transient is limited to the ASME Code allowable value (110% of the design pressure) with no credit taken for nonsafety-grade relief systems. The Beaver Valley Unit 2 plant was assumed to be operating at design conditions (102% of rated power) and the reactor is shut down by a high pressurizer pressure trip signal. The calculated pressure is less than 110% of design pressure.

Overpressure protection during low-temperature operation of the plant is provided by two PORVs and RHR suction relief valves in conjunction with administrative controls.

The applicant has met GDC 15 and 31. Appendix G criteria are expected to be met when the PORV setpoint curve is generated. In addition, the applicant has responded to Task Action Plan Items II.D.1 and II.D.3 of NUREG-0737.

5.4.7 Residual Heat Removal System

The design of the residual heat removal system (RHRS) for Beaver Valley Unit 2 has been reviewed in accordance with SRP 5.4.7 and Branch Technical Position RSB 5-1 of NUREG-0800. Conformance with the acceptance criteria, except as noted further below, formed the basis for the staff's conclusion that the design of the RHRS is acceptable provided that the RHRS pumps are fully qualified for continuous operation in the containment environment.

The RHRS has two independent cooling trains, which are designed for a pressure of 600 psig and a temperature of 400°F. Each train has a 4000-gpm pump and a heat exchanger that is designed to transfer 29 million Btu/hr to the component cooling water. The pumps, heat exchangers, and isolation and control valves are all located inside of containment. Each train of this RHRS is powered by an essential, separate, power supply. In the event of a failure of a power supply the licensee states that it is possible to switch the power source for the operation of isolation valves from the failed power supply to the functioning one. There are safety grade flow meters and low flow alarms connected to each of the two trains.

This RHRS operates in the following modes:

(1) <u>Cooldown</u>

Removes heat from the RCS after the system pressure and temperature have been reduced to approximately 400 psig and 350°F, respectively, by the steam and power conversion system. Under normal conditions, with two trains operating, it will take about 24 hours to get the reactor coolant temperature down to 140°F. If there is only one train operating it will take about 31 hours to get the reactor coolant temperature down to 212°F.

(2) Cold Shutdown

Removes fission product decay heat to maintain cold shutdown conditions.

(3) Refueling

Transfers water between the refueling cavity and the refueling water storage tank (RWST) at the beginning and end of the refueling operations.

(4) Startup

Acts as an alternate letdown path to control RCS pressure. In this mode the RHRS is connected to the chemical volume control system (CVCS) via the low pressure letdown line.

5.4.7.1 Functional Requirements

RSB 5-1 stipulates that the design of a plant shall be such that it can be taken to cold shutdown by using only safety grade systems and that these systems shall satisfy GDC-1 through 5. In this regard Section 5.4.7.2.5 of the FSAR states that the entire RHRS for Beaver Valley Unit 2 is designed as Safety Class 2 with the exception of the portions that form a part of the RCS pressure boundary which are designed as Safety Class 1. Compliance with GDC 1-5 criteria is as follows:

GDC-1, quality assurance aspects of safety grade systems, is evaluated in SER Section 17.1.

GDC-2, design bases for safety grade systems, is evaluated in SER Section 3.2.

GDC-3, fire protection of safety grade systems, is evaluated in SER Section 9.5.1.

GDC-4, environmental and missile protection design for safety grade systems, is evaluated in SER Sections 3.11 and 3.5.

GDC-5 is complied with because these RHRS's are not shared.

To comply with the redundancy criteria of GDC 34 the RHRS has two independent trains. Leak detection for the RHRS is discussed in Section 5.2.5 of this SER. Isolation valve and power supply redundancy are discussed under separate topics in this section. The staff has reviewed the description of the RHRS and the piping and instrumentation diagrams to verify that the system can be operated with or without offsite power and assuming a single failure. The two RHR pumps are connected to separate buses that can be powered by separate diesel generators in the event of loss of offsite power. Thus a single failure, such as that of a pump, valve, or heat exchanger, will still allow the operation of one train. However, in the inlet of each train there are two motor-operated valves (MOVs) for isolating the RHRS from the higher pressure RCS. The two MOVs in each train are connected to separate, Class 1E, electrical buses. Thus a failure of one of the electric buses could prevent water flow in both RHRS trains. To circumvent this single failure mode, the FSAR states that the electric power source for the MOV in each train that is not powered by the same b as powers the pump can be transferred to the other bus. The acceptability of this transfer method is discussed in Section 7.6 of this SER.

GDC 19 states that a control room shall be provided from which actions can be taken to maintain the plant in a safe condition under accident conditions, including loss-of-coolant accidents. SRP 5.4.7 stipulates that the control of the RHRS be such that the cooldown function can be performed from the control room assuming a single faiure of any active component, with only either onsite or offsite electric power available. Any operation required outside of the control room is to be justified by the applicant.

The applicant states in FSAR Section 5.4.7.2.7 that the RHRS is designed to be fully operable from the control room for normal operation and in Section 5.4.7.2.3 that the RCS can be taken from no-load temperature and pressure to cold conditions with only onsite or offsite power available assuming the most limiting single failure. It is also stated in Section 5.4.7.2.3 that as a backup to the isolation valves on th ECCS accumulators there are redundant, Class 1E, solenoid operated valves to ensure that any accumulator may be vented, should it fail to be isolated from the RCS.

The applicant states in FSAR Section 5.4.7.2.6 that in the event of such a failure, RHRS operation could be initiated by defeating the failed interlock by manual actions either at the solid state protection system cabinet or at the the affected motor control center. This could cause considerable delay in initiating RHRS operation. The applicant states that during this delay the auxiliary feedwater system (AFWS) and the steam generator PORVs could be used to continue the cooldown of the plant. As described in FSAR Section 10.4.9.2 there are secondary, Category I water supplies for the AFWS. The ultimate one is the Service Water System (SWS). Once this is connected, the AFWS could be used for core cooling for on indefinetly long period of time. In the event of a large break LOCA, the ECCS in conjunction with the recirculation coolers could be used to continue to cooldown of the plant while these manual actions were being taken outside of the control room. On this basis we find this action outside of the control room acceptable.

In FSAR Section 5.4.7.1 the applicant states that the RHRs is designed to reduce the temperature of the reactor coolant from 350°F to 140°F in approximately 24 hours. With only one train in service it will take approximately 31 hours to go from 350°F to 212°F. The cooldown time of 31 hours with one RHRS train is acceptable. With the stated 4-hour time for cooldown from standby to RHRS conditions the Beaver Valley Unit 2 plant can be brought to cold shutdown within a reasonable period of time with or without offsite power.

5.4.7.2 RHRS Isolation Requirements

The RHRS valving arrangement is designed to provide adequate protection to the RHRS from overpressurization when the reactor coolant system is at high pressure.

There are two separate and redundant motor-operated isolation valves (MOVs) between each of the two RHRS pump suction lines and the RCS hot legs. These valves are separately, diversely, and independently interlocked to prevent valve opening until the RCS pressure falls below 425 psig. If the valves are open, they are separately, diversely, and independently interlocked to close when the RCS pressure rises above 750 psig. Each one of the four RHRS suction

MOVs is aligned to a separate motor control center. One MOV in each suction line is powered from a separate power train. Thus a single failure will not prevent the isolation of the RHRS.

The possibility of water that is trapped between the two isolaton valves at a low temperature being heated and causing an overpressurization is discussed in FSAR Amendment 3. It is concluded that the maximum obtainable pressure would be 400 psi. We find this response acceptable.

There are a motor-operated isolation valve and a check valve in each of the RHRS discharge lines. The motor-operated valve is interlocked with a pressure signal to prevent its being opened whenever the RCS pressure is greater than 425 psig and to automatically close if the RCS pressure increases to 750 psig. The controls for the isolation of each discharge line are independent. The check valve is located in the emergency core cooling system.

The staff finds that the design of the RHRS isolation system satisfies the criteria of Branch Technical Position RSB 5-1 and is acceptable.

5.4.7.3 RHRS Pressure Relief Requirements

Overpressure protection for the RHRS is provided by a pressure relief valve in each inlet line. At its set pressure, this relief valve is designed to relieve the combined full water flow of two charging pumps. Fluid flowing through these valves goes into the pressurizer relief tank. The evaluation of the compliance of these valves with NUREG-0737 Item II.D.1 is included in SER Section 3.9.3.

In response to a question on what will alert the operator to the opening of the RHRS relief valves, the applicant responded in FSAR Amendment 3 that the operator would be alerted by either a high pressure alarm or a high level alarm from the pressurizer relief tank. An outline of the procedures the operator would follow for such an event was included in the response. We find this response acceptable.

For RHRSs with automatic isolation, Branch Technical Position RSB 5-1 criteria calls for adequate pressure relief capacity while the isolation valves are closing. The applicant states in FSAR Amendment 3 that additional pressure relief capacity is provided by the low-temperature overpressure protection system and that an evaluation to determine the adequacy of the RHRS overpressure protection system will be available by March 31, 1984. We will determine the adequacy of the RHRS pressure relief when this evaluation is received.

5.4.7.4 RHRS Pump Protection

The RHRS pumps are protected from operational overheating and loss of suction flow by miniflow bypass lines that assure flow to the pump suction. A throttling valve located in each miniflow line is adjusted and locked in place during initial system alignment to ensure required miniflow at all times. A pressure sensor in each pump discharge header provides a signal for an indicator in the control room. A high pressure alarm is also actuated by the pressure sensor. There are low flow alarms to alert the operator to turn off the pumps in the event the suction isolation valves close while the discharge isolation valves remain open.

Since both RHRS pumps are located inside of containment there is a question of whether or not this environment could cause a common mode failure. Moreover, the Equipment Environmental Qualification Table (3.11-1) in the FSAR for the RHRS does not include the RHR pumps. For a reliable system these pumps are going to have to be qualified for the containment environment and included in Table 3.11-1.

In its responses to questions, which are in FSAR Amendment 3, the applicant states that proper filling and venting procedures will be used to prevent water hammer in the RHRS and that just prior to its initiation, the RHRS will be cross-connected with the Chemical Volume Control System (CVCS) to pressurize the RHRS.

5.4.7.5 Tests, Operational Procedures, and Support Systems

The plant preoperational and startup test program provides for demonstrating the operation of the residual heat removal system in conformance with RG 1.68, as specified in SRP 5.4.7, Paragraph III.12.

The adequancy of the mixing of borated water added to the RCS under natural circulation and the ability to cooldown Beaver Valley Unit 2 with natural circulation will be verified by referencing the results of a natural circulation test at a similar plant. For this type of verification a detailed comparison of the two plants is required. This must include a comparison of the elevations of the major components.

As stated in FSAR Section 5A.3.2, the boron that is needed to offset the decay of xenon and the increase of reactivity during cooldown is provided by redundant, seismic Category I systems.

The staff has reviewed the component cooling water system to ensure that sufficient cooling capability is available to the RHRS heat exchangers. The acceptability of this cooling capacity and its conformance to GDC 44, 45, and 46 are discussed in Section 9.2.2 of this SER.

The applicant states that the RHRS is housed in a structure that is designed to withstand tornadoes, floods, and seismic phenomena, and that there are no motor-operated valves in the RHRS which are subject to flooding after a loss of coolant or steam line break accident. This area is addressed further in Section 3 of this SER.

Conformance with GDC 4 and the criteria in RG 1.46 for withstanding pipe whip inside containment is discussed in Section 3.6 of this SER. The entire RHRS is located inside the reactor containment.

The applicant, following SRP 5.4.7, Paragraph II.D.1, has demonstrated that suitable plant systems and procedures are available to place the plant in a cold shutdown condition with only offsite or onsite power available within a reasonable period of time following shutdown, assuming the most limiting single failure.

5.4.7.6 Conclusions

The RHR function is accomplished in two phases: the initial cooldown phase and the RHRS operation phase. In the event of loss of offsite power, the initial phase of cooldown is accomplished by use of the auxiliary feedwater system and the atmospheric dump valves. This equipment is used to reduce the reactor coolant system temperature and pressure to values that permit operation of the RHRS. The review of the initial cooldown phase is discussed in Section 10.3 of this SER. The review of the RHRS operational phase is discussed below.

The RHRS removes core decay heat and provides long-term cooling following the initial phase of reactor cooldown. The scope of review of the RHRS included piping and instrumentation diagrams, failure modes and effects analysis, and design performance specifications for essential components. The review has included the applicant's proposed design criteria and design bases for the RHRS and its analysis of the adequacy of and conformance to these criteria and bases.

Except for the above noted unresolved issues, the staff concludes that the design of the RHRS is acceptable and meets the relevant criteria of GDC 2, 5, 19, and 34. This conclusion is based on the following:

(1) As stated in SER Section 3.2, the applicant has met GDC 2 with respect to Position C.2 of RG 1.29 concerning the seismic design of systems, structures, and components whose failure could cause an unacceptable reduction in the capability of the RHRS.

(2) The applicant has met the criteria of GDC 5 with respect to sharing of structures, systems, and components by stating that the RHRS is not shared with another unit, i.e., Beaver Valley Unit 1.

6.3 Emergency Core Cooling System

The staff has reviewed the Beaver Valley Unit 2 emergency core cooling system (ECCS) in accordance with SRP 6.3 (NUREG-0800). Each of the four areas listed in the Areas of Review section of the SRP was reviewed according to the SRP Review Procedures. Conformance with the acceptance criteria, except as noted, below, formed the basis for concluding that the design of the facility for emergency core cooling is acceptable.

As specified in the SRP, the design of the ECCS was reviewed to determine that it is capable of performing all of the functions stipulated in the design criteria. The ECCS is designed to provide core cooling as well as additional shutdown capability for accidents that result in significant depressurization of the reactor coolant system (RCS). These accidents include mechanical failure of the RCS piping up to and including the double-ended break of the largest pipe, rupture of a control rod drive mechanism, spurious relief valve operation in the primary and secondary fluid systems, and breaks in the steam piping.

The principal bases for the staff's acceptance of this system are conformance to 10 CFR 50.46 and Appendix K to 10 CFR 50, and GDC 2, 5, 17, 27, 35, 36, and 37.

The applicant states that the criteria will be met even with minimum engineered safeguards available, such as the loss of one emergency power bus, with offsite power unavailable.

6.3.1 System Design

As specified in SRP 6.3.1.2, the design of the ECCS was reviewed to determine that it is capable of performing all of the functions stipulated in the design criteria. The ECCS design is based on the availability of a minimum of two accumulators, one high head safety injection (HHSI)/charging pump and one low head safety injection pump (LHSI) for the injection phase, and one HHSI/charging stipulated in GDC 19. The applicant states in FSAR Section 1.8 that the instrumentation in Beaver Valley Unit 2 is sufficient to allow the operating staff to ascertain plant conditions during and following a LOCA. The evaluation of this aspect of the post accident monitoring system (PAMS) is in Section 7.5 of this SER. The evaluations of other aspects of the PAMS are in Sections 6.2.5, 9.3.2, and 11.5.

As specified in SRP 6.3, Section III.3, the available net positive suction head (NPSH) for all the pumps in the ECCS (HHSI/charging, LHSI, and recirculation spray) has been shown to provide adequate margin by calculations performed to meet the safety intent of RG 1.1, "Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps."

As stipulated in SRP 6.3, Section III.11, the valve arrangement on the ECCS discharge lines has been reviewed with respect to adequate isolation between the RCS and the low-pressure ECCS. All lines to the RCS have at least two check valves in series with a normally closed isolation valve. This arrangement is acceptable.

Containment isolation features for all ECCS lines, including instrument lines (GDC 56 and the criteria in RG 1.11, "Instrument Lines Penetrating Primary Reactor Containment") are discussed in Section 6.2.4 of this SER.

In response to the staff's questions on an inspection program, operator training, and emergency procedures for dealing with debris, vortices, air entrainment and other containment sump problems the applicant stated in FSAR Amendment 3 that a response would be provided in a later amendment. This item will be considered open until that time.

The effects of primary coolant sources outside containment (NUREG-0737, Item III.D.1) are discussed in Section 13.5.2 of this SER.

During normal operation, the ECCS lines will be maintained in a filled condition. Suitable vents are provided and administrative procedures will require that ECCS lines be returned to a filled condition following events such as maintenance that require draining of any of the lines. Maintaining these lines in a filled condition will minimize the likelihood of water hammer occurring system startup.

The safety injection lines are protected from intersystem leakage by relief valves in both suction header and discharge lines. Intersystem leakage detection is described in Section 5.2.5 of this SER.

As specified in SRP 6.3, Section II.B, no ECCS components are shared between units. This meets GDC 5.

6.3.2 Evaluation of Single Failures

As specified in SRP 6.3, Section II, the staff has reviewed the system description and piping and instrumentation diagrams to verify that sufficient core cooling will be provided during the initial injection phase with and without the availability of offsite power, assuming a single failure. The cold leg accumulators have normally open motor-operated isolation valves in the discharge lines. One accumulator is attached to each of the RCS cold legs. These isolation valves will have control power removed to preclude inadvertent valve movement that could result in degraded accumulator performance.

Two active injection systems are to be available, each with two pumps operable. The pumps in each system are connected to separate power buses and are powered from separate diesel generators in the event of loss of offsite power, in accordance with GDC 17. Thus, at least one pump in each injection train would be actuated in the event of a loss of offsite power and failure of one diesel to start. The high-head injection systems contain parallel valves in the suction and discharge lines, thus ensuring operability of one train even if one valve fails to open. The low-head injection systems are normally aligned so that valve actuation is not required during the injection phase.

The engineered safety features actuation system (ESFAS) is designed to automatically perform the short-term injection phase; no operator actions are required. Two separate and redundant actuation trains are provided. Each actuation train is assigned to a corresponding electrical power train to ensure that, in the event of a single failure in the actuation logic, at least one emergency diesel generator, one LHSI, and one HHSI/charging pump would receive an actuation signal. There are also provisions for manual actuation, monitoring, and control of the ECCS on the main control board. This complies with SRP 6.3 and is acceptable.

After a LOCA the ESFAS will automatically initiate the transfer from the injection phase to the cold leg recirculation phase. However, the following operator actions are required to complete the transfer:

1. Open the cold leg isolation valve in the redundant high-head safety-injection flow path.

2. Close the isolation valves in both the common suction and discharge headers of the HHSI pumps to separate the redundant flow paths.

These operator actions are acceptable.

In this phase two of the four recirculation spray pumps, which are located in separate cubicles outside of containment, are automatically aligned to pump the water that will collect in the containment sump to the cold legs as well as to the inlets of the HHSI/charging pumps. The two operable charging pumps have separate flow paths to hot leg connections. This provides the capability for backflushing through the core to prevent boron precipitation. Since recirculation spray coolers are used to transfer the decay heat to the service water, it also provides subcooled water to terminate boiloff. This meets the criteria of SRP 6.3 Section III.6.

To ensure a long term cooling capability, leak detection and a method for isolating the leak is required. The applicant states that means are provided to detect and isolate leaks in the emergency core cooling flow path within 30 minutes. In a study of the system, the applicant found that the largest, sudden, potential leak is the failure of a recirculation spray pump shaft seal and that this would result in a leak rate of less than 50 gpm. This maximum leak would be detected by alarms which indicate the loss of accumulator pressure on the seal water. The applicant states that if this leaking pump is isolated within 30 minutes the ECCS will still meet the minimum core cooling requirements. The staff finds this system acceptable. The evaluation of the complete Equipment and Floor Drainage System is in Section 9.3.3 of this SER.

Flooding of ECCS components inside containment following a LOCA has been evaluated. The applicant states that all motor-operated valves which have to change positions after the injection phase are located to prevent their vulnerability to flooding and that those valves whose spurious repositioning could result in the loss of the ECCS function have their power removed.

Based on its review of the design features, [redacted], the staff concludes that the ECCS complies with the single-failure criterion of GDC 35.

6.3.3 Qualification of Emergency Core Cooling System

The ECCS design to seismic Category I criteria, in compliance with RG 1.29, and its location in structures designed to withstand a safe-shutdown earthquake and other natural phenomena, per the criteria of GDC 2, and the equipment design to Quality Group B, in compliance with RG 1.26, are discussed in Section 3.2 of this SER.

The ECCS protection against missiles inside and outside containment by the design of suitable reinforced concrete barriers, which include reinforced concrete walls and slabs (conformance to GDC 4), is discussed in Section 3.5 of this SER. The protection of the ECCS from pipe whip inside and outside of containment is discussed in Section 3.6 of this SER.

The active components of the ECCS designed to function under the most severe duty loads, including safe-shutdown earthquake, are discussed in Sections 3.9 and 3.10 of this SER. The ECCS design to permit periodic inspection in accordance with ASME Code, Section XI, which constitutes compliance with GDC 36, is discussed in Section 6.6 of this SER. This meets the criteria set forth in SRP 6.3, Paragraph III.23.c.

The ECCS is connected to one subsystem that serves another function. The centrifugal HHSI/charging pumps are normally aligned to the chemical and volume and control system (CVCS) for maintaining the required amount and chemistry of water in the RCS and for supplying water to the seals of the reactor coolant pumps. On an ECCS actuation signal, the system is aligned for ECCS operation and the CVCS function is isolated. This normal system use does not impair its capability to function in the ECCS mode.

6.3.4 Testing

The applicant has committed to demonstrate the operability of the ECCS by subjecting all components to preoperational and periodic testing, per the criteria of RG 1.68, "Preoperational and Initial Startup Test Programs for Water-Cooled Power Reactors," RG 1.79, "Preoperational Testing of Emergency Core Cooling System for Pressurized Water Reactors," and to GDC 37.

6.3.4.1 Preoperational Tests

One of these tests is to verify system actuation: namely, the operability of all ECCS valves initiated by the safety injection signal, the operability of all safeguard pump circuitry down through the pump breaker control circuits, and the proper operation of all valve interlocks.

Another test is to check the cold leg accumulator system and injection line to verify that the lines are free of obstructions and that the accumulator check valves and isolation valves operate correctly. The applicant will perform a low-pressure blowdown of each accumulator to confirm that the line is clear and check the operation of the check valves.

The applicant will use the results of the preoperational tests to evaluate the hydraulic and mechanical performance of ECCS for delivering the flow for emergency core cooling. The pumps will be operated under both miniflow (through test lines) and full-flow (through the actual piping) conditions.

By measuring the flow in each pipe, the applicant will make the adjustments necessary to ensure that no one branch has an unacceptably low or high resistance. As part of the ECCS verification, the applicant will analyze the results to ensure there are sufficient total line resistance to prevent excessive runout of the pumps and adequate NPSH under the most limiting system alignment and RCS pressure. The applicant will verify that the maximum flow rate from the test results confirms the maximum flow rate used in the NPSH calculations under the most limiting conditions and will also confirm that the minimum acceptable flow used in the LOCA analysis is met by the measured total pump flow and the relative flow between the branch lines.

The staff concludes that the preoperational test program conforms to the recommendations of RGs 1.68 and 1.79 and is acceptable pending successful completion of the program. Additional discussion of the preoperational test program is in Section 14 of this SER.

6.3.4.2 Periodic Component Tests

Routine periodic testing of the ECCS components and all necessary support systems at power will be performed. Valves that actuate after a LOCA are operated through a complete cycle. Pumps are operated individually in this test on their miniflow lines except the charging pumps which are tested by their normal charging function. The applicant has stated that these tests will be performed in accordance with ASME Code, Section XI.

6.3.5 Performance Evaluation

The ECCS has been designed to deliver fluid to the RCS to limit the maximum fuel cladding temperature following transients and accidents that require ECCS actuation. The ECCS is also designed to remove the decay and sensible heat during the recirculation mode. 10 CFR 50.46 lists the acceptance criteria for an ECCS. These criteria include the following:

(4) The applicant has met the criteria of GDC 27 with regard to providing combined reactivity control system capability to ensure that under postulated accident conditions and with appropriate margin for stuck rods, the capability to cool the core is maintained, and the applicant's design meets the guidelines of RG 1.47.

(5) The applicant has met the criteria of GDC 35 in regard to abundant cooling capability for ECCS by providing redundant safety-grade systems that meet the recommendations of RG 1.1.

(6) The applicant has met the criteria of GDC 36 with respect to the design of ECCS to permit appropriate periodic inspection of important components of the system.

(7) The applicant has met the criteria of GDC 37 with respect to designing the ECCS to permit testing of the operability of the system throughout the life of the plant, including the full operational sequence that brings the system into operation.

(8) The applicant has provided an analysis of the ECCS performance using an approved analysis model that meets the criteria of Appendix K to 10 CFR 50 and has shown the system performance meets the acceptance criteria of 10 CFR 50.46. This includes a demonstration that the peak cladding temperature, maximum hydrogen generation, and long-term cooling, as calculated with an acceptable evaluation model, are in accordance with these criteria.

13 CONDUCT OF OPERATIONS

13.3 Emergency Planning ~~Preparedness Evaluation~~

13.3.1 Introduction

The Beaver Valley Power Station (BVPS) operated by Duquesne Light Co., is located on the banks of the Ohio River in Shippingport Borough, Beaver County, western Pennsylvania, about 25 miles northwest of Pittsburgh, PA and 5 miles east of Liverpool, Ohio. The plume exposure EPZ includes portions of three counties in three States: Beaver County, PA, Columbiana County, Ohio, and Hancock County, West Virginia.

Beaver Valley Unit 1 was first operational in October 1976, and Unit 2 has a projected construction completion date of December 1985. The Beaver Valley site adjoins the Shippingport Atomic Power Station, a light water breeder reactor demonstration plant operated by Duquesne Light Company (DLC) for the Department of Energy.

Shippingport has an emergency preparedness plan for emergency conditions at the Shippingport Station. Both the Shippingport emergency plan and the BVPS emergency plan provide for communications interface between the plants, primarily for notification and implementation of onsite protective actions at either plant in response to an emergency condition at the other plant.

Previously, the staff had reviewed and commented on an earlier version of the Beaver Valley Emergency Plan. (NRC letter, G. Smith to J. J. Carey, April 28, 1982, Emergency Preparedness Appraisal, Appendix C.) The current plan is Issue 7, Rev. 0, dated December 21, 1982. As stated in Duquesne Light Company letter of April 28, 1983, the plan requires revision to incorporate BVPS Unit 2 as an operating unit vice a construction site.

The acceptance criteria used as the basis for the staff's review of the BVPS emergency plan are specified in SRP Section 13.3, "Emergency Planning" (NUREG-0800, July 1981) and include the planning standards of 10 CFR 50.47(b), the requirements of Appendix E to 10 CFR 50, and the specific criteria of NUREG-0654/FEMA-REP-1, Revision 1, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," dated November 1980. The criteria of NUREG-0654 have been endorsed in RG 1.101, Revision 2, "Emergency Planning and Preparedness for Nuclear Power Reactors" dated October 1981, and thus have the same status as a Regulatory Guide.

Evaluation of the state of emergency preparedness for the BVPS facility also involves the review of State and local radiological emergency response plans by the Federal Emergency Management Agency (FEMA). The Standard Review Plan states that the FEMA findings on offsite plans are reviewed by the NRC and that a full-scale exercise is conducted at the facility, demonstrating that the applicant and the State and local organizations are capable of taking adequate protective actions should a radiological emergency occur. The FEMA findings have not yet been developed; however, the FEMA review of offsite plans and subsequent submittal of findings and determinations to the NRC must be complete before authorization of operation above 5% of rated power. Similarly, a full-scale exercise must be conducted before operation above 5% of rated power will be permitted. The findings and determinations of FEMA on the adequacy of the State and local emergency response plans, and the overall conclusion of the NRC on the state of emergency preparedness for BVPS will be presented in a future supplement to the SER.

Section 13.3.2 of this report lists each planning standard of 10 CFR 50.47(b), followed by an evaluation of the applicable portions of the applicant's plan that relate principally to that particular standard. Section 13.3.3 of this report provides the staff's conclusions.

13.3.2 Evaluation of the Applicant's Onsite Emergency Plan

13.3.2.1 Assignment of Responsibility (Organization Control)

Standard

Primary responsibilities for emergency response by the nuclear facility licensee and by State and local organizations within the emergency planning zones (EPZ) have been assigned, the emergency responsibilities of the various supporting organizations have been specifically established, and each principal response organization has staff to respond to and to augment its initial response on a continuous basis.

Emergency Plan Evaluation

The Beaver Valley Power Station (BVPS) Emergency Plan identifies those State, local, and Federal response organizations which have response roles in the event of an accident. Since the plume exposure and ingestion Emergency Planning Zones (EPZs) incorporate portions of Pennsylvania, Ohio, and West Virginia, three State emergency response agencies have primary response roles; Pennsylvania Emergency Management Agency (PEMA), Ohio Disaster Services Agency (ODSA), and West Virginia Office of Emergency Services (WVOES). Similarly, the Beaver County Emergency Management Agency, Columbiana County Disaster Services Agency, and the Hancock County Emergency Services Agency serve as the lead county response agencies in Pennsylvania, Ohio and West Virginia respectively.

A concept of operations for each of these organization and their relationship to the total effort is specified. The interrelationships are illustrated in Table 3.1, Figure 5.3, and in Figure 5.6 of the emergency plan. These Figures are applicable only to Unit 1, and will be revised for two unit operations.

The Emergency Director, initially the Shift Supervisor who is succeeded by the Station Superintendent, is identified as the person who will assure overall direction and control of the Duquesne Light emergency response.

Twenty-four-hour-per-day emergency response, including manning of communications links is provided by the on-shift crew. This crew can be augmented as required in an emergency.

Written agreements from Federal, State, and local agencies and other support organizations having an emergency response role within the EPZs are appended to the Plan. The Plan describes the role of each of the agencies with which there are agreements and its relationship to the role of the plant.

The Plan describes a corporate level support organization which would be responsible for assuring continuity of resources for protracted 24-hour operations. This organization encompasses all of the Nuclear Division as required; consisting mainly of Nuclear Operations, assisted by Nuclear Safety and Licensing, Nuclear Engineering, and Nuclear Support Services. This organization will perform emergency duties essentially identical with its normal duties.

13.3.2.2 Onsite Emergency Organization

Standard

Onshift facility licensee responsibilities for emergency response are unambiguously defined, adequate staffing to provide initial facility accident response in key functional areas is maintained at all times, timely augmentation of response capabilities is available, and the interfaces among various onsite response activities and offsite support and response activities are specified.

Emergency Plan Evaluation

The onsite emergency organization of plant personnel for all shifts and its relation to the responsibilities and duties of the normal staff complement are specified. Plant staff emergency assignments for managers and key coordinators are described. The On-shift Supervisor is designated as the Emergency Director and has the authority to initiate emergency actions and recommend protective measures to offsite officials until relieved of Emergency Director duties by a designated Senior Management Official (Station Superintendent, Chief Engineer, or Maintenance Supervisor). The responsibilities, lines of succession, and functions which cannot be delegated are also described. Table 5.1 specifies the position or title and major tasks to be performed by the persons assigned to the functional areas of emergency activity. The plan indicates the staffing levels which can be augmented within 60-120 minutes. Table 5.1 of the plan has been revised to correspond with Table B-1 of NUREG-0654. Results of a survey conducted by BVPS on travel times indicates that augmentation by the emergency organization is in line with the guidelines of Table B-1.

The corporate management, administrative, and technical support personnel who will augment the plant staff are specified for those functional areas of emergency response. Section 6 of the revised plan states that the Shift Supervisor/ Emregency Director, upon classifying the condition as Alert or higher, will assure that key Emergency Coordinators are notified using the beeper paging system and/or telephonic communications. These Emergency Coordinators will initiate additional call-out of personnel as needed. Implementing Procedure EPP/IP 1.6, "Emergency Operations Facility Organization and Operation", was revised to describe the staffing, augmentation, and operation of the EOF.

The contractor and private organization who may be requested to provide technical assistance to and augmentation of the emergency response organization are specified, as are the services to be provided by local agencies, including police, ambulance, medical, hospital, and fire fighting organizations.

The interfaces between and among the onsite functional areas of emergency activity and the offsite emergency organization made up of corporate support, local services support, and State and local government response organizations are specified. A block diagram is provided in Figure 5.4 of the Plan. Copies of letters of agreement with these organizations as well as letters of agreement with Westinghouse Electric Corporation, Teledyne Isotopes, and the Institute of Nuclear Power Operations are appended to the Plan.

13.3.2.3 Emergency Response Support and Resources

Standard

Arrangements for requesting and effectively using assistance resources have been made, arrangements to accomodate State and local staff at the licensee's near-site Emergency Operations Facility have been made, and other organizations capable of augmenting the planned response have been identified.

Emergency Plan Evaluation

The BVPS plan describes the types of response expected to be provided by the Federal agencies, such as NRC, DOE Federal Radiological Monitoring and Assessment Plan (FRMAP), National Weather Service (NWS), and FEMA. The primary and secondary method of notification for each is specified, but the person who is authorized to request Federal assistance is not named.

The BVPS plan states that, since FRMAP resources are to be used for offsite response, the emergency plans of Pennsylvania, West Virginia, and Ohio have made provisions for the use of FRMAP resources. To provide access to the plant release and meteorological data, space will be made available in the EOF for a liaison from FRMAP as well as liaison personnel from each jurisdiction within the EPZ.

The plan does not identify the specific State, local, and licensee resources available to support the Federal response, such as, airports, transportation, office space, communication, etc. The plan contains provision for the dispatch of DLC liaison personnel to the primary governmental EOCs upon request.

The BVPS plan describes onsite laboratories, such as the dosimetry lab and sample preparation and counting facility; the radiological lab, and the radiological monitoring van (a portable field laboratory). Offsite laboratory support is available from the Shippingport Atomic Power Station adjacent to BVPS; Bettis Atomic Power Labs approximately one hour by car; Teledyne, the environmental contractor; Radiation Management Corp., Philadelphia; NUS Corp., Pittsburgh; and Interex Corp., Natich, Maine. In addition to these, emergency support and assistance is available from INPO who maintains a roster of personnel and an inventory of material, equipment, and services. The NSSS supplier is Westinghouse who has agreed to provide emergency engineering assistance on a 24-hour/day, 7 day/week basis. Additional industry support is available from the Central Area Power Coordinating Organization (CAPCO), whose members own or control several nuclear power plants.

The following items require resolution:

1. The plan should specify the persons, by title, who are authorized to request Federal assistance (Cla).

2. The plan should specify licensee, State, and local resources available to support the Federal response (Clc).

13.3.2.4 Emergency Classification System

Standard

A standard emergency classification and action level scheme, the basis of which includes facility system and effluent parameters, is in use by the nuclear facility licensee, and State and local response plans call for reliance on information provided by facility licensee for determinations of minimum initial response measures.

Emergency Plan Evaluation

The BVPS plan provides for a graded scale of response for distinct classifications of emergency conditions, action within those classifications, and criteria for escalation to a more severe classification. This classification system is compatible with the classification scheme used by the emergency/

disaster response agencies in all three risk counties and risk States. The Plan uses four categories, Unusual Event, Alert, Site Area Emergency, and General Emergency. The categories and the initiating events within each category are described in Section 4 of the Plan, and are consistent with the criteria of Appendix 1 to NUREG-0654.

The Emergency Action Levels (EALs) are outlined in Section 4 and Table 4.1 of the plan, and detailed in the emergency procedures to include specific instrument readings, plant system and effluent parameters, and equipment status indications characteristic of a spectrum of off-normal conditions and accidents corresponding to most initiating conditions of each emergency class.

The EAL sets appear adequate except that the following initiating conditions are missing or are deficient as noted. (The example initiating conditions are listed in Appendix 1 of NUREG-0654.)

Unusual Event

- Initiating Condition (IC) 9 - Loss of Engineered Safety Feature or fire protection system requiring Tech Spec. shutdown.

  This initiating condition is listed in Table 4.1 but Tab 13 omits discussion of the fire protection system.

° Initiating Condition 13d - Hurricane

The EAL, in Table 4.1 of the plan, and Tab 22 of the procedure, discuss tornadoes only. There is no reference to hurricanes or other high winds.

Initiating Condition 14e - Turbine rotating component failure causing rapid shutdown.

This IC is not addressed.

° Initiating Condition 15 - Other plant conditions.

This IC is not addressed.

° Initiating Condition 17 - Rapid depressurization of secondary side.

This IC is not addressed. Tab 7 addresses only Main Steam Line break.

Alert

° Initiating Condition 2 - Steam Generator tube failure with loss of offsite power.

This IC is not addressed in either Tab 6 or 7.

° Initiating Conditon 15 - Radiological effluents greater than 10 times Technical Specifications.

The monitors and instruments are specified in Tab 20, however, the alarm set-point is 100 times Technical Specifications vice 10 times.

° Initiating Condition 17d - Hurricane winds near design level.
None of the EALs address hurricane winds.

° Initiating Condition 18e - Turbine failure causing casing penetration.
This IC is listed as an Unusual Event which is overly conservative.

° Initiating Condition 19 - Other plant conditions.
This IC is not addressed.

Site Area Emergency

° Initiating Condition 3 - Rapid failure of Steam Generator tubes with loss of offsite power.
This IC is not addressed.

° Initiating Condition 9 - Transient requiring operation of shutdown system with failure to scram.
This IC is partially addressed as an alert condition (IC 11), but is not addressed as a Site Area Emergency.

° Initiating Condition 15c - Substained winds in excess of design levels. Table 1 of EPP/J-1 lists EAL of "winds in excess of design levels" but Tab 22 discusses only tornado winds.

° Initiating Condition 17 - Other conditions. This IC is not addressed.

General Emergency

° Initiating Condition 5b - Loss of Feedwater and Condensate system followed by failure of emergency feedwater systems. Table 1 of EPP/I-1 lists this IC but there is no discussion in any of the Tabs.

° Initiating Condition 5d - Loss of onsite and offsite power with total loss of emergency feedwater make-up capability for several hours. This IC is listed in Table 1 of EPP/I-1 but is not addressed in any of the Tabs.

13.3.2.5 Notification Methods and Procedures

Standard

Procedures have been established for notification, by the licensee, of State and local response organization and for notification of emergency personnel by all response organizations; the content of initial and followup messages to response organizations and the public has been established; and means to provide early notification and clear instruction to the populace within the plume exposure pathway EPZ have been established.

Emergency Plan Evaluation

The emergency communications procedures contain instructions and forms for initial contact and notification, and forms to be used when the agency calls back for follow-up information and verification.

The Shift Supervisor, upon classifying the event as an Alert emergency or higher, will ensure that key Emergency Coordinators are notified, as needed, using the beeper paging systems or telephonic communications. The key Emergency Coordinators will initiate additional call-out of personnel as needed. The emergency notification procedure, EPP/IP-1.1, contains instructions for notification of offsite authorities and emergency response agencies, phone lists, and an Initial Notification Form. The Initial Notification Form contains the basic information recommended in NUREG-0654, and requests that appropriate individuals of the response organization contact the station for additional information. The dissemination of public information to the news media has been done per EPP/IP-9.1, "Emergency Public Information Plan BVPS". The licensee has installed a public alert and warning system consisting of sirens mounted along public highways and at various fire stations throughout the 10-mile EPZ.

13.3.2.6 Emergency Communications

Standard

Provisions exist for prompt communication among principal response organizations to emergency personnel and to the public.

Emergency Plan Evaluation

The emergency plan specifies five independent systems for outside communications to Federal, State and county authorities, corporate management, and offsite support/response groups. These systems are commerical telephone system, the utility PAX system, dedicated "hot lines", the utility system operator direct lines, and the utility's industrial radio system. Onsite, the plant alarm system, station paging system, and a two-way alarm system between BVPS and Shippingport provide communication and notification for station personnel. These communication links are illustrated in Figures 7.2.a and b of the plan. The plan provides 24-hour-per-day activation of the State and local emergency response network through the Pennsylvania Emergency Management Agency (PEMA). The various primary and backup systems have redundant power supplies. A telephone link is provided between the plant and the Beaver County hospitals with a radio link between the hospitals and ambulances. Periodic testing of the communication systems will be conducted.

13.3.2.7 Public Information

Standard

Information is made available to the public on a periodic basis on how it will be notified and what its initial actions should be in an emergency; the principal points of contact with the news media for dissemination of information during an emergency (including physical location or locations) are established in advance; and procedures for coordinated dissemination of information to the public are established.

Emergency Plan Evaluation

The BVPS, in cooperation with State and county authorities, will develop and periodically disseminate emergency planning instructional material to residents and transients in the EPZ. This material will include (1) basic information on radiation, (2) public notification system, (3) public response to warning signals, (4) evacuation routes and procedures, (5) sheltering procedures, and (6) ingestion pathway protective actions. The methods to be utilized to ensure that emergency planning information is transmitted to residents and transients in the EPZ are (1) yearly ads in the local newspapers summarizing actions to be taken by the residents, (2) printed instructions and evacuation maps to be distributed to EPZ residents, (3) printed instructions to be included in the local telephone directory, and (4) printed instructions and evacuation maps to be distributed to motels, hotels, and recreation areas.

The Manager, Public Information Department will provide a point of contact for the news media. A nearsite emergency news center for use by the news media will be established at the Willows Motel about 3 miles from the site for events classified higher than an Alert. The DLC corporate headquarters and the William Penn Hotel will be used as an alternate news center in case the Willows Motel is unavailable due to radiological conditions. Corporate headquarters will be the point of contact for Unusual Event and Alert emergencies. The Vice President, Nuclear Division, or designee, will serve as the company spokesperson.

The Public Information Department will maintain a representative at the Technical Support Center (TSC) and Emergency Operation Facility (EOF) to ensure that correct and proper information is provided for public release.

The Customer Services Department will be staffed and prepared to answer calls from the news media and general public and to deal with rumors and incorrect information that may develop during the emergency.

Programs will be conducted annually by DLC to acquaint news media representatives with the content and implementation of the BVPS emergency plan, and the public notification system. In addition, information concerning radiation and points of contact for release of public information during an emergency will be provided.

The following item requires resolution:

> The printed instructions and evacuation maps for the public shall be developed, submitted for FEMA and staff review, and distributed to residents and transients within the 10-mile EPZ. (G1)

13.3.2.8 Emergency Facilities and Equipment

Standard

Adequate emergency facilities and equipment to support the emergency response are provided and maintained.

Emergency Plan Evaluation

The licensee has established a Technical Support Center (TSC) and a nearsite Emergency Operations Facility (EOF) in the Emergency Response Facility (ERF). An Operations Support (Assembly) Center with adequate capacity has been established in the Process Instrument and Rod Position Instrument Area. An offsite EOF has been established at the company's South Heights district office that is about 10 miles from the site. The permanent ERF, designed to satisfy the functional requirements of the TSC and the EOF, also contains an Emergency Control Center, a Dosimetry Lab, Sample Preparation and Counting Facility, and Decontamination Facility, as described in DLC's response to Generic Letter 82-33 (Supplement 1 to NUREG-0737). The permanent ERF is complete and operational except for the Safety Parameter Display System (SPDS). The SPDS is scheduled for installation during the forth refueling outage of Unit 1, starting in September 1984. Activation and staffing of the ERF will be done in a timely manner, following the guidelines of Table 2 of Supplement 1 to NUREG-0737 (Table B-1 of NUREG-0654). On an interim basis the staff finds the ERF facility adequate for the purpose.

As indicated in Supplement 1 to NUREG-0737, the staff will conduct a post-implementation appraisal of the adequacy of the applicant's completed emergency response facilities against the requirements in Supplement 1 to NUREG-0737 in accordance with a schedule to be developed between the applicant and the NRC (see 13.3.4, Item III.A.1.2. of this report).

Onsite and offsite monitoring and analysis systems and equipment have been established and are identified in the plan. The central point for the receipt and analysis of field monitoring data and coordination of sample media has been established at the EOF.

Meteorological instrumentation and procedures have been, or are being established including provisions for obtaining representative current meteorological information from the National Weather Service at Moon Township. The routine inspection, inventory, maintenance and calibration of emergency equipment and supplies is satisfactorily addressed in the plan.

The revised plan, and implementing procedure EPP/IP-1.6, "EOF Organization and Operation ", provides for augmentation of the EOF organization to perform the functions of overall emergency management, radiological/environmental assessment, and protective action recommendations.

13.3.2.9 Accident Assessment

Standard

Adequate methods, systems, and equipment for assessing and monitoring actual or potential offsite consequences of radiological emergency condition are in use.

Emergency Plan Evaluation

The revised plan, and Table 1 of emergency procedure EPP/I-1, present an emergency classification system and EALs, consistent with Appendix 1 of NUREG-0654. The procedure, EPP/I-1, supplements the table with Tabs for each initiating condition identifying the instruments for each parameter with alarm setpoints or emergency action levels for each. See Section 13.3.2.4 for the review of the EALs.

In the event of a known or projected release of radioactive material in quantities or concentrations greater than the Beaver Valley Power Station Technical Specifications, immediate and continuous assessment, including dose projection, is performed by on-duty shift personnel. Following activation of the Technical Support Center, dose projection activities are performed by the Environmental Assessment and Dose Projection Coordinator and assigned assistants at the TSC. Upon declaration of a Site Area or General Emergency, this function transfers to the Emergency Operations Facility (EOF). Responsibilities and functions assigned to these personnel are identified in the Plan. Activation of the emergency facilities is described in the plan. The training of personnel assigned dose projection functions is identified.

The plan and procedures describe dose commitment methods that rely on manual calculation. DLC is developing computer equipment and methodology to upgrade these dose calculations. The manual methods will be retained as back-up to the computer method.

13.3.2.10 Protective Response

Standard

A range of protective actions has been developed for the plume exposure pathway EPZ for emergency workers and the public. Guidelines for the choice of protective actions during an emergency, consistent with Federal guidance, are developed and in place, and protective actions for the ingestion exposure pathway EPZ appropriate to the locale have been developed.

Emergency Plan Evaluation

Onsite protective actions, including criteria and methods, are described in the plan. The primary protective action is evacuation of non-essential personnel and the use of protective equipment and clothing for those personnel who are required to perform emergency activities. Provision is made for increasingly larger areas of evacuation commensurate with existing conditions. Other onsite protective actions include the use of respiratory protection equipment, anti-contamination clothing, thyroid prophylaxis, and the administration of an effective radiological controls program.

Measures have been established to provide for personnel accountability in the event of an evacuation in accordance with Emergency Implementing Procedures. These measures are based on security identification badges and/or computerized access security system (card-key). The plan does not specify a time limit for initial accountability per the criteria of NUREG-0654, nor continued accountability of members of the emergency organization.

Offsite protective actions are addressed in the plan. Such actions are primarily the responsibility of State and local emergency organizations, but may be based on recommendations by the BVPS Emergency Director (Emergency/Recovery Manager for Site Area or General Emergencies). These offsite organizations may invoke any emergency actions which they deem appropriate, according to assessment of the individual situation, and at any level of radioactive material release or projected offsite dose. The key element which ensures compatibility of the BVPS Plan and offsite emergency plans is the provision for initial notification and continuing status reports to the State and local agencies, conveying current release and dose projection information.

The plan, in the Time Evacuation Study, gives time estimates for evacuation of individuals in the various planning sectors of the EPZ, maps showing major roadways through the EPZ, radiological sampling and monitoring locations, and population distribution by sectors throughout the EPZ.

Radiological monitoring procedures are detailed in the Radiological Control Manual (RCM). During site evacuations, personnel and vehicle surveys are performed on the site exit road adjacent to the Switchyard Relay Building, using portable survey instruments. Decontamination of vehicles will be done with fire hoses on the graveled area in the switchyard adjacent to the fire hydrants. Individuals will be returned to the Station, or to the Shippingport Visitors Center for decontamination. In the event of an immediate evacuation, personnel will be directed to proceed by personal automobiles to the designated remote assembly area for personal and vehicle monitoring.

The following item requires resolution:

> The plan does not indicate if initial accountability can be accomplished within 30 minutes, or the methodology to be used to maintain accountability on a contined basis. (J.5)

13.3.2.11 Radiological Exposure Control

Standard

Means for controlling radiological exposures, in an emergency, are established for emergency workers. The means for controlling radiological exposures shall include exposure guidelines consistent with EPA Emergency Workers and Life-Saving Activity Protection Action Guides.

Emergency Plan Evaluation

The BVPS Radiation control Manual and the Radiation Protection Procedures establish the radiation protection program at the Beaver Valley Power Station by providing criteria, guidelines, and instructions for maintaining the radiation exposure of station personnel as low as reasonably achievable (ALARA) and within Federal standards (10 CFR 20). Specifically, the BVPS Radiation Protection Program provides for exposure control, exposure monitoring, access control, identifying radiological areas and materials, respiratory protection, contamination control, and radioactive material handling. Administrative controls (radiation workpermits, radiation clearance, and ALARA measures) will remain in force during an emergency. If necessary operations require personnel exposure in excess of normal limits or if normal work practices result in unacceptable delay, the Radiological Controls Coordinator may waive or modify established exposure control criteria and methods, but the Emergency Director or Emergency/Recovery Manager are the only individuals who may authorize doses in excess of 10 CFR 20 limits. Procedures are listed in the plan to provide a 24-hour-per-day capability to determine the radiation dose received by emergency workers, issue and process dosimetry devices and maintain dose records.

Onsite control of access to contaminated areas, and control of access to onsite food and drinking water has been established. Provisions have been established for decontamination of personnel, materials and equipment, for disposal of contaminated waste, and return contaminated areas and items to normal use. The Licensee has established the capability to monitor and decontaminate relocated (evacuated) personnel and provide clean clothing as required. Decontamination kits are stocked with the various decontamination materials required for the various kinds of contamination including radioiodine on the skin.

13.3.2.12 Medical and Public Health Support

Standard

Arrangements are made for medical services for contaminated injured individuals.

Emergency Plan Evaluation

At least two first aid personnel, trained in Red Cross Multi-Media are onsite at all times. First Aid Kits are available at several locations onsite, and a first aid room is available. The licensee has made arrangements by written agreement with Aliquippa Hospital, Medical Center of Beaver County, Wald & Spritzer Associate/University of Pittsburgh RERP, and Presbyterian-University Hospital for medical assistance to injured personnel who also may be contaminated.

These organizations can be contacted directly or through the Beaver County Emergency Medical Services (via Beaver County Communications Center). Emergency Medical Services Radio provides communications between the Beaver County Communications Center, the ambulances, and the Beaver Valley hospitals. Ambulance emergency supply kits, per Appendix D of the plan, are available from storage in the First Aid Room at the station.

Personnel dosimetry for ambulance personnel is provided by the station. Contaminated patients are accompanied by a radiation control person who is responsible for appropriate contamination control measures to minimize the spread of contamination to the ambulance, the hospital, and hospital personnel. Back-up transportation can be provided by a suitable company vehicle, or a private vehicle (on a voluntary basis). A letter of agreement for transport of injured/contaminated personnel has not been included in Appendix D of the plan, although Medic-Rescue is listed in the table of contents.

The following items require resolution:

1. The plan lacks a letter of agreement for ambulance service. (A.3)

2. Licensee shall certify annually as to the currency of the letters of agreement. (P.4)

13.3.2.13 Recovery and Reentry Planning, Postaccident Operations

Standard

General plans for recovery and reentry are developed.

Emergency Plan Evaluation

Provisions are made for establishing a recovery organization which is commensurate with the scope and magnitude of an emergency condition. These provisions include the assignment of qualified individuals to fill recovery organization positions as may be appropriate. Termination from a severe emergency involving offsite consequences will be through joint evaluation by the utility, the three States involved, and the NRC.

Criteria for termination of the emergency condition and establishment of the primary recovery organization are detailed in the plan. All emergency response and support organizations shall be notified of the termination of the emergency, and/or the initiation of recovery operations, using the same procedures used for initial notification. If the emergency resulted in damage to the plant, the Duquesne Light Nuclear Division will be activated as the recovery organization under the direction of the Manager of Nuclear Operations. The Nuclear Division is structured into functional areas and staffed by personnel competent in the various disciplines necessary for emergency recovery conditions. The Nuclear Division maintains offices either onsite or nearsite.

The BVPS plan contains provisions for periodically estimating total population exposure resulting from radioactive releases during the emergency.

13.3.2.14 Exercises and Drills

Standard

Periodic exercises are (will be) conducted to evaluate major portions of emergency response capabilities, periodic drills are (will be) conducted to develop and maintain key skills, and deficiencies identified as a result of exercise or drills are (will be) corrected.

Emergency Plan Evaluation

An exercise appropriate to a Site Area or General Emergency, and which simulates conditions resulting in offsite radiological releases which would require protective response by offsite authorities shall be conducted at least once per calendar year for the Beaver Valley Power Station. This exercise shall test the integrated capability and a major portion of the basic elements of the Emergency Preparedness Plan. The scenario will be varied from year to year such that all major elements of the Plan and the emergency organizations are tested within a five-year period. Consistent with the ability of offsite agencies to participate, this exercise should be scheduled to commence between the hours of 1800 and 2400, and between 0000 and 0600 once every six years.

Scenarios for the joint exercises will be a cooperative effort between all participants, and, to the extent possible, will allow free-play for decision-making by the participants.

Each exercise will be observed and critiqued by qualified observers from Federal, State, and/or local governments. Critiques of all scheduled exercises will be held soon after the completion of the exercise, with all observers having the opportunity to provide input. An overall exercise report will be compiled and distributed to all primary participants. The exercise critique report shall document the significant deficiencies observed during the exercise. The Emergency Planning Supervisor is responsible for recommending corrective actions for each deficiency, submitting the recommendations to the Onsite Safety Committee for review and to the Station Superintendent for approval. The Emergency Planning Supervisor will also prepare necessary changes to the Emergency Preparedness plan and/or Emergency Implementing Procedures for review and implementation by the Onsite Safety Committee.

The plan provides for drills as pre-planned simulations in which the participants are "walked" or "talked" through one or more procedures, or aspects of the emergency plan to provide individuals with hands-on training in a controlled situation. Drills will be evaluated by the drill instructor, who will normally correct erroneous performance on-the-spot.

Each exercise or drill will be conducted to: (1) ensure that the participants are familiar with their respective duties and responsibilities, (2) verify the adequacy of the BVPS Emergency Preparedness Plan and the methods used in the Emergency Implementing Procedures, (3) test communications networks and systems, (4) check the availability of emergency supplies and equipment, (5) verify the operability of emergency equipment, and (6) verify the adequacy of interrelationships with offsite agency plans.

The following drills will be conducted according to schedule:

Fire Emergency Drill	Semi Annual
Medical Emergency Drill	Annual
Radiation Emergency Drill	
Onsite Airborne	Semi Annual
Onsite Liquid	Annual
Offsite Air and Liquid	Annual

Communications drills will be conducted regularly per the plan on a frequency schedule to ensure continued operability of the systems.

13.3.2.15 Radiological Emergency Response Training

Standard

Radiological emergency response training is provided to those who may be called upon to assist in an emergency.

Emergency Plan Evaluation

The licensee provides training and annual retraining in the plan and procedures for all permanent plant personnel. This training includes assignment of duties and responsibilities, location and use of assembly areas, and familiarization with alarms and communications systems. In addition, those personnel having specific response roles as part of the onsite emergency organization are given specialized training in accordance with their expected duties. These areas include emergency response coordination and direction, accident assessment, radiological monitoring, repair and damage control, rescue, and first aid.

The plan provides training for non-emergency onsite personnel, and those individuals working onsite but outside the Protected Area. Training consists (as a minimum) of instructions on warning signals, assembly areas, and evacuation routes.

The applicant will provide training and annual retraining for those offsite organizations whose services may be required in an emergency, such as fire, police, medical support, and rescue personnel. The training will be consistent with the organization's emergency functions. The training program for members of the applicant's emergency organization will include practical drills, as discussed in Section 13.3.2.14 above.

13.3.2.16 Responsibility for the Planning Effort: Development, Periodic Review, and Distribution of Emergency Plans

Standard

Responsibilities for plan development and review and distribution of emergency plans are established, and planners are properly trained.

Emergency Plan Evaluation

The BVPS Superintendent has overall responsibility and authority for maintenance of an appropriate emergency preparedness stature at BVPS. The Station Superintendent is assisted by the Emergency Planning Supervisor who is assigned the primary responsibility for the emergency plan. The Emergency Planning Supervisor remains current by attending appropriate seminars and training courses. The Emergency Planning Supervisor is responsible for the development and update of the Emergency Plan, coordination with onsite and offsite response organizations, and ensures the correspondence of the BVPS Emergency Plan with the interfacing offsite plans. These plans are listed in Section 2.5 of the BVPS Plan. The plan has a table of contents, and includes, in Appendix C, a listing of implementing procedures. The Onsite Safety Committee uses the review of the combined exercise report as an annual review of the Emergency Preparedness program. See Planning Standard 13.3.2.14, for discussion of the review, management controls for evaluation and correction, and distribution of the critique report.

13.3.3 Conclusions

Based on a review of the Beaver Valley Power Station Emergency Plan for conformance with the specific criteria in NUREG-0654/FEMA-REP-1, which addresses each of the planning standards of 10 CFR 50.47(b) and with the requirements of Appendix E to 10 CFR 50, the staff concludes that, upon satisfactory correction of those items requiring resolution and those items requiring commitment by Duquesne Light Company, as identified in Section 13.3.2 of this report and summarized below, the Beaver Valley Emergency Plan will provide an adequate planning basis for an acceptable state of onsite emergency preparedness.

- o Revise the emergency plan to include Unit 2 as an operating unit vice a construction site.

- o The plan should specify the persons, by title, who are authorized to request Federal assistance.

- The plan should specify licensee, State, and local resources available to support the Federal response.

- Correct the deficiencies in the EAL sets as listed in Section 13.3.2.4.

- The printed instructions and evacuation maps for the public shall be developed and submitted for staff review.

- The plan should specify methodology for initial accountability (to be accomplished within 30 minutes), and the methodology to be used to maintain accountability on a continued basis.

- The plan lacks a letter of agreement for ambulance service.

- Licensee shall certify annually as to the currency of the letters of agreement.

After reviewing the findings and determinations made by FEMA on the adequacy of State and local emergency response plans, and after reviewing the revisions to the applicant's Emergency Plan, a supplement to this report will provide the staff's overall conclusions as to whether the state of onsite and offsite emergency preparedness provides reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency.

13.3.4 TMI Action Items

III.A.1.2 Upgrade Emergency Support Facilities

Supplement 1 to NUREG-0737, "Requirements for Emergency Response Capability" issued by Generic Letter No. 82-33, dated December 17, 1982, states that the NRC will conduct post-implementation reviews of emergency response facilities (ERF's), and provides all licensees and applicants with the requirements and guidance against which the ERF's will be evaluated.

Generic Letter No. 82-33 requested that by April 15, 1983, each licensee and applicant develop and submit to the NRC its own plant-specific schedule for completion of the ERF's, including a description of the plans for phased implementation and integration of all emergency response activities. Final staff evaluation of the operational capability of completed ERF's (i.e., TSC, OSC, and EOF) will be conducted as part of the post-implementation review of emergency response capabilities against the requirements contained in Supplement 1 to NUREG-0737. Accordingly, the schedule for the post-implementation appraisal of the ERF's will be established after these facilities have been completed.

III.A.2 Improving Emergency Preparedness - Long Term

The objective of this item was for each nuclear facility licensee to upgrade its emergency preparedness effort at each nuclear facility in order to provide reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency. This task involved three phases: (1) submittal of upgraded emergency plans consistent with the revised emergency planning regulations effective November 3, 1980 and the guidance of NUREG-0654, FEMA-REP-1, Revision 1; (2) submittal of implementing procedures, and (3) implementation of radiological response plans. Particular emphasis in this task was given to the upgrade of meteorological assessment capabilities. The previous guidance on meteorology found in Appendix 2 to NUREG-0654 and in Revision 1 to Regulatory Guide 1.23 has been superseded by Supplement 1 to NUREG-0737. Review efforts under this task will be conducted in accordance with the rule on emergency planning, the guidance in NUREG-0654, and Supplement 1 to NUREG-0737. The results of this review are reported in Section 13.3 of the SER. The capability of licensees to implement their emergency plans will be assessed during an onsite appraisal as part of the preoperational inspection.