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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 containment systems, structural and geotechnical engineering, power systems, and reactor systems. Enclosure 1 is a list of the open items resulting from this review and Enclosure 2 is the staff evaluation which should be incorporated into the BVPS-2 draft SER.

With this transmittal, staff input into the draft SER has been completed and all significant issues for this review phase have now been identified. Therefore, we request that you perform an assessment of the time required to respond to all of the draft SER open items and provide us with a schedule for those responses. Following receipt of your response schedule, we plan to reassess the BVPS-2 licensing schedule. You will be advised of the outcome of this reassessment.

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,

Original signed by:
Victor Nerses

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

Enclosure: As stated

cc: See next page

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Enclosure 1

Additional Open Items (CSB, SGEB, PSB, RSB)
for BVPS-2 FSAR Review

(166) Barometeric pressure for containment depressurization analysis (6.2.1.1)

(167) Mass and energy release analyses (6.2.1.1, 6.2.1.2, 6.2.1.3, 6.2.1.4)

(168) Subcompartment analyses (reactor cavity, steam generator and pressurizer) (6.2.1.2)

(169) Containment sump design (6.2.2)

(170) Post-accident hydrogen monitoring system (6.2.5)

(171) Type C testing exclusion of valves (6.2.6)

(172) Longitudinal sections and parameters of Category 1 buried pipelines (2.5.4.1, 2.5.4.2)

(173) Stability analyses (2.5.4.3.1, 2.5.4.3.4, 2.5.5.1, 2.5.5.2)

(174) Foundation data for main intake structure (2.5.4.3.2)

(175) Measured, estimated and allowable settlement data (2.5.4.3.3)

(176) Differential settlements (2.5.4.3.3)

(177) Settlement monitoring program (2.5.4.3.3)

(178) Densification of soils (2.5.4.3.4)

(179) Soil damping valves (2.5.4.3.5)

(180) Soils effective strength parameters (2.5.5.3)

(181) Accuracy of SIDES program (2.5.5.3)

(182) Offsite power systems (8.2.2.2, 8.2.2.3)

(183) Independance between offsite and onsite power sources (8.2.2.4)

(184) Automatic load tap changer (8.2.2.5)

(185) Testing of offsite power transfer (8.2.3.1)

(186) Voltage analysis for safety-related loads (8.3.1.1)

(187) Diesel generator testing (8.3.1.3, 8.3.1.5, 8.3.1.8, 8.3.1.9, 8.3.1.10)

(188) Compliance with BTP-PSB-2 (8.3.1.4)

(189) Diesel generator loading (8.3.1.6)

(190) Compliance with IEEE Standard 387-1977 (8.3.1.11)

(191) Power removal for selected safety valves (8.3.1.12)

(192) Automatic reclosure of breakers after manual trip (8.3.1.14)

(193) Replacements for Class 1E loads (8.3.1.15)

(194) Accident loading capacity of the diesel generator (8.3.1.16)

(195) Connecting non Class 1E loads with Class 1E loads (8.3.1.17, 8.3.1.18, 8.3.1.19)

(196) Compliance with GDC 2 and 4 (8.3.3.1.1 thru 8.3.3.1.4)

(197) Compliance with GDC 17 (8.3.3.3.1 thru 8.3.3.3.7)

(198) Electrical independence between power supplies (8.3.3.5)

(199) Compliance with GDC 50 (8.3.3.7.1, 8.3.3.7.2)

(200) Information on evaluations of individual events (15)

(201) Turbine trip event (15.2.3)

(202) Reactor coolant pump rotor seizure (15.3.3/15.3.4)

(203) Inadvertant boron dilution during refueling (15.4.6)

(103)* Evaluation of steam generator tube rupture (15.6.3, Table 15.1)

(108) TMI items II.K.1.5, II.K.1.10 and II.K.3.17, (15.9.1/15.9.2, 15.9.10)

(109) TMI item II.K.2.13 (15.9.3)

(110) TMI item II.K.3.2 (15.9.6)

(111) TMI items II.K.3.5, II.K.3.30 and II.K.3.31 (15.9.8, 15.9.12, 15.9.13)

*Previously identified in the draft SER

Enclosure 2

6.2 CONTAINMENT SYSTEMS

The Beaver Valley Power Station, Unit 2 Containment Systems include the containment structures and associated systems, SUCH AS THE ~~including~~ containment heat removal systems, containment isolation system, and containment hydrogen control system, that function to prevent or control the release of radioactive fission products which might be released into the containment atmosphere following a postulated loss of coolant accident (LOCA), secondary system pipe rupture, or fuel handling accident.

The staff has reviewed the INFORMATION RELATING TO THE ~~applicant's~~ design, design bases and safety analyses for the containment and the containment systems provided in the FSAR. The acceptance criteria used as the basis for our evaluation are contained in Section 6.2.1, "Containment Functional Design," 6.2.2, "Containment Heat Removal Systems," 6.2.4, "Containment Isolation System," 6.2.5, "Combustible Gas Control in Containment," and 6.2.6, "Containment Leakage Testing," of the Standard Review Plan (SRP), NUREG-0800. These acceptance criteria include the applicable General Design Criteria (GDC) of Appendix A to 10CFR Part 50, Regulatory

Guides, Branch Technical Positions, and industry codes and standards, as specified in the above cited sections of the SRP.

6.2.1 Containment Functional Design

6.2.1.1 Containment Structure

The containment structure for Beaver Valley, utilizes the subatmospheric containment concept, and houses the Nuclear Steam Supply System (NSSS), including the reactor coolant system (RCS), associated auxiliary systems and certain components of the plant engineered safety feature systems. It is a steel-lined reinforced concrete structure with an internal free volume of about 1,800,000 cubic feet. The maximum and minimum internal design pressures of the containment structure are 45 psig, and 8 psia, respectively, and the design temperature is 280 F. (See also Section 3.8 of the SER).

During normal operation, the containment structure is maintained at a subatmospheric pressure (i.e., about 9 to 12 psia). In the event of a high energy line break accident, the containment would be depressurized and a subatmospheric condition re-established within 60 minutes; this condition would be maintained for at least 30 days following an accident.

Maximum Pressure/Temperature and Depressurization Analyses

The applicant has performed containment response analyses for a spectrum of postulated reactor coolant system and secondary system pipe ruptures to verify the containment functional design; i.e., the acceptability of the containment design pressure and containment depressurization criterion, and establish the pressure and temperature conditions for environmental qualification of safety-related equipment located inside containment. The containment functional analyses include the peak containment pressure analysis and the containment depressurization analysis.

With respect to the peak containment pressure analysis, the loss of coolant accidents (i.e., RCS pipe breaks) analyzed by the applicant include a spectrum of hot leg and cold leg (pump suction and pump discharge) breaks, up to and including the double-ended rupture of the largest reactor coolant line. The spectrum of secondary system pipe breaks analyzed by the applicant include double-ended and split breaks of the main steam line at different reactor power levels (i.e., 102%, 70% and 30% of full power, and the hot shutdown condition). A single failure analysis is not necessary for the peak containment pressure evaluation since the peak pressure for each case analyzed occurs before active engineered safety feature systems can influence the results. The design basis accident for peak containment pressure (containment integrity DBA) was determined to be the double-ended guillotine break in the hot leg (HLDER). The peak containment pressure calculated by the applicant (using the Stone and Webster LOCTIC computer code, was 44.7 psig, which is below the containment design pressure of 45 psig. The applicant also performed a sensitivity study and found that the initial conditions which result in the highest peak calculated pressure are the maximum initial containment pressure (11.6 psia), maximum initial containment temperature (105 F) and maximum initial containment dewpoint (105 F), i.e., relative humidity. These are the limiting valves that will be allowed by the Technical Specifications.

The staff has performed a confirmatory analysis of this design basis accident using the CONTEMPT-LT/28A computer code. The results of the staff's analysis are in good agreement with the applicant's results.

For the secondary system pipe break analysis, the applicant analyzed a spectrum of main steam line break accidents covering different double ended ruptures and split breaks of the main steam line, and reactor operating power levels from hot shutdown to full power. For the DER, the forward flow area (effective break area) is limited to 1.4 FT^2 by a flow restrictor in the main steam line. Two different single active failures were considered, namely, the failure of a main steam isolation valve to close and the failure of an emergency bus to energize (causing the failure of one ESF train which results in minimum containment heat removal capability). Redundant valves are provided for automatic isolation of the main feedwater lines. The highest containment pressure, 41.2 psig, was calculated for a full DER at 30% power, with a MSIV failure, and with an initial containment pressure of 11.6 psia and initial containment dry bulb and dewpoint temperatures of 105° F. The highest containment temperature, 333° F, was calculated for a 0. 707 ft^2 split break at 30%

power, assuming either a MSIV failure or emergency bus failure, and with an initial containment pressure of 9.11 psia, initial dry bulb temperature of 105° F and initial dewpoint temperature of 55° F.

With respect to the containment depressurization analysis, only pump suction ruptures were determined to be of concern since they produce the highest energy flow rates during the post-blowdown period. The design basis accident for maximum depressurization time and subatmospheric peak pressure (containment depressurization BBA) was found to be the double-ended rupture of the pump suction line (PSDER), with miminum ESF (loss of offsite power and emergency diesel generator failure resulting in the loss of one engineered safety feature train, i.e., one charging pump, one safety injection pump, one quench spray pump and two containment recirculation pumps with associated coolers). The applicant also performed a sensitivity study and found that the initial conditions which result in the maximum depressurization time are: initial containment pressure of 9.85 psia, initial containment temperature of 85° F, initial containment dewpoint of 85° F, service water temperature of 86° F, and refueling water storage tank temperature of 50° F. These are the limiting values that will be allowed by the Technical Specifications. The applicant calculated a maximum containment depressurization time of 3480 seconds, which is within the design limit of 3600 seconds, and a subatmospheric peak pressure -0.08 psig.

The staff is unable to conclude on the acceptability of the applicant's containment depressurization analysis at this time because the applicant has not stated the barometric pressure used in the analysis. The applicant will be required to discuss and justify the barometric pressure for the plant site. This matter will remain an open item pending the receipt of additional information.

The staff's review of the applicant's containment response analysis has included the postulated reactor coolant system and secondary system pipe breaks, initial conditions, input parameters and assumptions. However, the methodology used to calculate the mass and **energy release rate data for the LOCA and MSLB accident has not been reviewed due to a lack of information (see Section 6.2.1.3 and 6.2.1.4 of the SER). Therefore, the staff can not conclude on the acceptability of** the applicant's analysis at this time. This will be an open item until further information is provided by the applicant regarding the calculation of the mass and energy release data.

Protection Against Damage from External Pressure

The containment structure is designed to withstand the external (differential) pressure load due to a postulated inadvertent actuation of the containment quench spray system during normal plant operation. The maximum pressure differential is based on the difference between the maximum barometric pressure and the minimum attainable internal containment pressure. The applicant calculated a minimum internal pressure of 8.0 psia for this postulated event.

The staff has reviewed the applicant's analysis and HAS found that the applicant's assumptions regarding initial containment conditions and containment quench spray system operation tend to minimize the containment pressure (e.g., minimum initial air partial pressure, maximum initial containment temperature and final containment temperature ~~which~~ equal~~s~~ TO the minimum RWST temperature). The applicant, ~~further~~ assumed a barometric pressure of 14.36 psia, which is the maximum expected barometric pressure for the Beaver Valley 2 site. Based on the conservative analysis performed by the applicant, the staff concludes that the containment external (differential) pressure design basis is acceptable.

6.2.1.2 <u>Subcompartment Analyses</u>

Subcompartment analyses are required to determine the acceptability of the design differential pressure loadings on containment internal structures from high energy line ruptures. The applicant has performed the necessary subcompartment analyses for the reactor cavity, ~~a~~ steam generator compartments and the pressurizer compartment, where high energy line ruptures are postulated to occur. The applicant has developed models for each subcompartment, with a selected pipe break location, type and size, and initial conditions, that result in maximum differential pressure loads on the subcompartment walls.

The mass and energy release rate data used in the subcompartment analyses were calculated using the SATAN-VI computer program (WCAP-8306). The acceptability of using SATAN-VI for this purpose is currently under separate staff review. This matter will remain an open item until such time that pending staff information needs under the Westinghouse Topical Report Review are satisfied.

The applicant used the THREED computer program to analyze the pressure transients in the reactor cavity, the steam generator compartment and the pressurizer compartment. The staff's confirmatory analysis is based on the COMPARE-MOD 1A computer code.

A separate discussion and review of the analyses of the reactor cavity, steam generator and pressurizer compartments are presented below.

Reactor Cavity Analysis

The reactor cavity is a heavily reinforced concrete structure that performs the dual function of providing reactor vessel support and radiation shielding. For the reactor cavity analysis the applicant postulated a 150 in^2 cold leg, limited displacement rupture (LDR) at the reactor vessel nozzle. The staff has reviewed the applicant's analysis and concurs in the selection of the design basis pipe break, contingent upon the acceptability of the mechanically constrained limit on the pipe break size. (See Section 3.6 of the SER).

The reactor cavity subcompartment model employed by the applicant was developed to account for all important obstructions to flow. This model is consistent with the recommendations concerning nodalization that are presented in NUREG/CR-1199, "Subcompartment Analysis Procedures Report." We have examined the applicant's nodal model and find it acceptable. is ~~modeled~~ in accordance with current NRC guideline as specified in NUREG/CR-1199, therefore, is , AND,

The applicant calcuated a peak differential pressure load on the reactor cavity wall of 115.9 psid, for the design basis 150 in^2 LDR. ~~All assumptions utilized by the applicant in the reactor cavity analysis have been reviewed and found to be acceptable.~~ The selection of the break size location and the use of restraints to limit the break area are discussed in Section 3.6. The assumed initial conditions are chosen to maximize the differential pressure responses. In addition, the staff performed a confirmatory analysis using the COMPARE-MOD 1A computer code,wh~~en~~ich confirmed that the applicant's result is conservative. However, the design basis value of the differential pressure load on the reactor cavity wall is not documented in the FSAR; therefore, the staff can not confirm that the reactor cavity wall design basis is satisfied. This will be an open item pending the receipt of additional information from the applicant.

The applicant has not provided in the FSAR an analysis of the forces and moments on the reactor vessel due to the differential pressure across the vessel caused by a reactor coolant system pipe break within the reactor cavity. This matter will be an open item pending the receipt of additional information from the applicant.

Steam Generator Subcompartment Analyses

Steam generator cubicle 2 was selected as the representative steam generator cubicle since all three steam generator cubicles are similar in design. The applicant analyzed three RCS breaks in the steam generator compartment to evaluate loads on the subcompartment walls and component supports. Main steam lines are not routed through the steam generator cubicles and are, therefore, not considered in the analysis. The three pipe ruptures analyzed include a 360-in^2 LDR at the steam generator outlet nozzle, a 180-in^2 LDR at the reactor coolant pump (RCP) outlet nozzle, and a 70.7-in^2 longitudinal intrados split break at the steam generator inlet elbow. These breaks were chosen from the nine breaks in the applicant's sensitivity study as being limiting cases which envelop conditions resulting from all nine breaks. The staff has reviewed the spectrum of postulated breaks analyzed by the applicant and finds them acceptable.

The applicant's nodalization scheme of the steam generator subcompartment was developed to take into account all significant physical obstructions to flow. The staff has reviewed the applicant's model and finds it acceptable. The results of the applicant's analyses predict a peak differential pressure of 12.9 psid for the design basis 707-in^2 longitudinal intrados split break. However, the design basis value of the differential pressure Load on the steam generator wall is not documented in the FSAR. This will be an open item pending the receipt of additional information from the applicant.

Pressurizer Subcompartment Analyses

The applicant considered three breaks for the pressurizer cubicle, and the pressurizer relief tank cubicle; namely a spray line DER in the upper pressurizer cubicle, a surge line DER at the pressurizer nozzle and a surge line DER in the pressurizer relief tank cubicle. The applicant's nodalization models of the pressurizer subcompartment were developed to take into account all critical restrictions to flow. The staff has reviewed the applicant's models and the spectrum of postulated breaks and finds them appropriately conservative and acceptable.

The results of the applicants analysis of the spray line DER in the upper pressurizer cubicle gave a peak differential pressure of 18.07 psid across the pressurizer nodel boundary surface. However, the design basis value of the differential pressure load on the pressurizer cubicle walls is not documented in the FSAR. This will be an open item ~~to~~ pending the receipt of additional information from the applicant..

6.2.1.3 Mass and Energy Release Analyses for Postulated LOCA

The applicant calculated the mass and energy release rate data for reactor coolant system pipe breaks at three break locations including the hot leg piping between the reactor vessel and steam generator, the cold leg piping at the pump suction, and the cold leg piping at the pump discharge. The results indicate the pump suction break is the worst case for long term containment depressurization, and the hot leg break is the worst case for containment peak pressure. The staff has reviewed the applicant's spectrum of breaks, the description of the LOCA transient models and the single failure considerations, and finds them acceptable. The method used by the applicant to compute the mass and energy release rates from reactor coolant pipe breaks for the containment functional analyses is described in a reference Westinhouse letter that is currently under staff review. At this time, we are not in a position to conclude on the acceptability of the blowdown methodology. This matter will remain an open item pending the completion of the staff's review.

6.2.1.4 Mass and Energy Release Analyses for Postulated Secondary System Pipe Ruptures

The applicant has computed the mass and energy release rates for postulated main steam line breaks using the MARVEL Computer Code (WCAP-8843, 1977). However, the mass and energy release data for the MSLB analysis were not documented in the FSAR. The staff has requested this information for review, and to facilitate the staff's confirmatory analysis. This matter will remain an open item pending the receipt of additional information.

6.2.1.5 Minimum Containment Pressure Analysis for Emergency Core Cooling System Performance Capability Studies

Appendix K to 10CFR Part 50 requires that the containment pressure used for evaluating core cooling effectiveness during reactor vessel reflood shall not exceed a pressure calculated conservatively for this purpose. The calculation must include the effect of operation of all installed containment pressure reducing systems and processes. The corresponding reflood rate in the core will then be reduced because lessened containment pressure reduces the resistance to steam flow in the reactor coolant loops and increases the boiloff rate from the core.

The applicant has performed the required containment back-pressure calculation, as addressed in Section 6.2.1.5 of the FSAR, using the methods and assumptions described in "Westinghouse Emergency Core Cooling System Evaluation Mode-Summary," WCAP-8339, Appendix A, for the limiting case LOCA, the double-ended cold leg guillotine break (C_D = 0.4) (i.e, the break found to produce the highest peak clad sing temperature). Mass and energy release rates for this break were calculated, using the method described in Section 15.6.5 of the FSAR. This method is evaluated separately in Chapter 15 of this SER.

The staff has reviewed the applicant's input parameters used in the minimum containment pressure analysis including initial containment conditions, containment net free volume, containment active heat removal, passive heat sinks, heat transfer to passive heat sinks, and found them to be acceptably conservative, and in conformance with BTP CSB 6-1.

6.2.1.6 <u>Summary and Conclusions</u>

The staff has evaluated the Beaver Valley, Unit 2 containment functional design with respect to the acceptance criteria in SRP Section 6.2.1.1.A, 6.2.1.2, 6.2.1.3, 6.2.1.4, and 6.2.1.5 and concluded that

General Design Criteria 13, 16, 38 and 50 have been met with the following exceptions:

1. Staff acceptance of the applicant's containment depressurization analysis is contingent on the applicant's justification of the barometric pressure for the Beaver Valley site.

2. The method used by the applicant to compute the mass and energy release rates from postulated reactor coolant system pipe breaks for the containment analyses and for the subcompartment analysis remaining to be approved by the staff.

3. The mass and energy release data for postulated main steam line breaks have not been documented in the FSAR. Staff acceptance of the applicant's main steam line break analysis is contingent upon the receipt of this information.

4. There are two open items concerning the staff's review of the applicant's subcompartment analysis. First, subcompartment design pressure differentials for the reactor cavity, and steam generator and pressurizer compartments have not been documented in the FSAR. Second, the applicant has not provided an analysis of the forces and moments on the reactor vessel due to the differential

pressure caused by a RCS break within the reactor cavity. Staff acceptance of the applicant's subcompartment analysis is contingent upon the receipt of this information.

6.2.2 Containment Heat Removal Systems

The function of the containment heat removal systems is to remove heat from the containment atmoshpere to limit, reduce and maintain at acceptably low levels, the containment temperature and pressure following a loss of coolant accident or main steam line break. In addition to heat removal provided by passive means such as heat transfer to containment structures and components, the Beaver Valley 2 design includes active containment heat removal systems (CHRS). The active CHRS includes two spray systems; namely, the quench sp~~r~~ray system (QSS) and the recirculation spray system (RSS); the containment air coolers are not included in the CHRS. The CHRS is designed to depressurize the containment to a sub-atmospheric condition within one hour. For a discussion of the fission product removal function of the CHRS, see SER Section 6.5.

The QSS is composed of two redundant 100 percent capacity trains each containing a quench spray pump, a chemical injection system and riserpipe leading to two spray headers.

The two trains connect to the two common 360-degree spray headers in parallel with risers 180 degrees apart. There are a total of 159 Spraco model 1713A nozzles on the two quench spray ring headers; 120 nozzles on the lower header and 39 nozzles on the upper header. Each quench spray pump is rated at 3000 gpm of spray flow to the spray headers. Both spray pumps operating together can supply approximately 4500 gpm to the spray headers. The QSS is designed to spray cold borated water into the containment from the refueling water storage tank (RWST) no later than 83 seconds after receipt of a containment isolation Phase B signal (CIB). Sodium hydroxide (NaOH) solution from the chemical additive tank (CAT) is added to the quench spray by means of the chemical injection system upon receiving a CIB signal. Once the quench spray discharge has ended, flow from the chemical injection pump is automatically diverted to the containment sump.

The RSS is designed to provide additional depressurization of the containment and to maintain the containment at a subatmospheric condition in the long term following the accident. The RSS consists of two 360 degree spray ring headers and four pumps and heat exchangers. Each spray ring header contains 292 SPRACO model 1713A nozzles, and is fed by two risers, with each riser originating from one of the recirculation coolers.

The two redundant recirculation spray pumps that feed each header are each supplied with emergency power from separate diesel generators. Each RSS pump takes suction from the containment sump at approximately 3480 gpm (50% heat removal capacity). The RSS is capable of operating in the post-accident environment to maintin a sub-atmospheric pressure for 30 days following a high energy line break.

The RSS pumps are started automatically about 628 seconds after receipt of a CIB signal, and the spray becomes EFFECTIVE about 714 seconds after the CIB signal. When the water in the DWST reaches a predetermined low level, the flow from two of the RSS pumps is automatically diverted to the cold leg recirculation mode OF by ECCS.

The CHRS satisfies the provisions of Regulatory Guide 1.26, "Quality Group Classifications for Water, Steam and Radioactive-Waste Containing Components of Nuclear Power Plants," and 1.29, "Seismic Design Classifications," for engineered safety features. The applicant has provided testing information (FSAR Section 14.2, Test Programs in compliance with the basic intent of Regulatory Guide 1.68 "Initial Test Programs for Water Cooled Nuclear Power Plants" demonstrating the ability of the quench spray system and recirculation spray system to function following a postulated single active failure.

Regulatory Guide 1.82, "Sumps for Emergency Core Cooling and Containment Spray Systems," provideds design guidelines for containment sumps that are to serve as sources of water for ECCS and the containment spray system following a LOCA. The guidelines address redundancy, location and arrangement criteria, as well as debris screen provisions to ensure adequate pump performance. The staff has review the Beaver Valley 2 sump design against this guidance.

A single containment sump has been provided, and is enclosed by a protective screen assembly that has a total screen area of about 150 ft^2. Furthermore, the containment sump is divided at the center line by screening and vertical bars so that a failure of either half would not adversely affect the other half. The redundant recirculation pump suctions are located in seperate halves of the sump. Therefore, even though the single sump design is not in accordance with Regulatory Guide 1.82 recommendations, the staff has concluded that adequate measures have been taken to assure that the RSS function will not be lost.

The protective screen assembly provides three stages of screening, namely, vertical trash bars, a coarse mesh screen (3/4" opening) and a fine mesh screen (3/32" opening). The fine mesh screen opening is smaller than the smallest coolant passage gap in the reactor core and smaller than a spray nozzle orifice. The screen assembly rises vertically approximately 5 feet above the containment floor, and is arranged so that no single failure could result in the clogging of all suction points of the recirculation spray system. Following a LOCA, the top of the screen assembly would be under about 10 feet of water. System design allows for 50 percent blockage of the sump screening without loss of function. However, the applicant should further justify the acceptability of 50 percent blockage assumption by specifying the types (and quantity of each type) of insulation used within the Beaver Valley 2 containment, and discussing the susceptibility of the insulation of become dislodged by virtue of its proximity to high energy line piping.

The applicant has conducted containment sump model testing at the Alden Reserach Laboratory, but has not reported the results to the staff. The staff has learned, however, that the sump model used differs from the sump design shown in the FSAR. The staff has requested the applicant to provide the results of the Alden sump tests and discuss the significance of the results relative to the performance of the as-built, Beaver Valley 2 sump. This information has not been received. This matter will remain an open item pending the receipt of the Alden test report and an accompanying discussion of the applicability of the results to the as-built Beaver Valley 2 sump.

The staff has reviewed the net positive suction head (NPSH) calculations submitted by the applicant. The analysis shows the NPSH available to the reciculation pumps during both the spray mode and the low head safety injection mode is always greater than the required NPSH. The applicant has complied with the provisions of Regulatory Guide 1.1, "Net Positive Suction Head for Emergency Core

Cooling and Containment Heat Removal Systems", with one exception. Regulatory Guide 1.1 states that containment heat removal systems should be designed so that adequate NPSH is provided to system pumps assuming maximum expected temperatures of pumped fluids and no increase in containment pressure from that present before the postulated LOCA. Instead, the applicant calculated the NPSH available, as stated in the FSAR, Section 6.3.2.3.2, using a saturated sump model (i.e., the containment atmospheric pressure is conservatively assumed to be equal to the vapor pressure of the liquid in the sump, ensuring that credit is not taken for containment pressurization during the transient). The staff has previously found the saturated sump model to be conservative and, therefore, acceptable.

The staff has reviewed the information in the applicant's FSAR and in responses to staff requests for addtional information concerning the containment heat removal systems to assure conformance to the acceptance criteria contained in SRP Section 6.2.2. The staff finds that the containment heat removal systems satisfy the requirement of General Design Criteria 38, 39, and 40, and the provisions of Regulatory Guide 1.1 on an acceptable alternative basis as defined above. However, there are several issues in Regulatory Guide 1.82 which the applicant has not adequately addressed, and for which additional information is needed before the staff can conclude on the acceptability of the sump design. In considering the location of the sump within the containment, the applicant should discuss the potential for whipping pipes, high velocity jets of water or steam, or direct streams of water (which may contain entrained debris) to adversely affect the integrity or performance of the sump protective screen assembly. The applicant should also address the acceptability of the water velocity at the fine mesh screen, based on one-half of the available free area to account for blockage. The acceptability of the materials used in the construction of the sump screen assembly, and the inservice inspection requirements for the sump components, as well as the provisions made to facilitate such inspections, should also be addressed.

6.2.3 Secondary Containment Functional Design

The Beaver Valley 2 design does not include a secondary containment.

6.2.4 Containment Isolation System

The function of the containment isolation system (CIS) is to allow the normal or emergency passage of fluids through the containment boundary while preserving the ability of the boundary to prevent or limit the escape of fission products that may result from postulated accidents. In general, for each fluid system penetration at least two barriers are required between the containment atmosphere or the reactor coolant system and the outside atmosphere, so that failure of a single brrier will not prevent isolation of the containment.

Containment isolation for Beaver Valley 2 is accomplished in two phases. The containment isolation Phase A (CIA) signal isolates all non-essential system lines penetrating the containment, and is initiated by any of the following: (1) high containment pressure (Hi-1 setpoint); (2) low compensated steam line pressure; (3) pressurizer low pressure; or (4) manual actuation. The containment isolation Phase B (CIB) signal isolates the component cooling water supply and return lines for the reactor coolant pumps (RCPs) and control rod drive mechanism (CRDM) shroud coolers, and the service water lines to the containment recirculation air coolers. The CIB signal is initiated by high containment pressure (Hi-3 setpoint) or by manual actuation. The containment isolation signals which initiate containment isolation functions are summarized in Table 6.2.4-1. The applicant has documented that each system line having automatic containment isolation valves, which must be immediately isolated following an accident, is isolated by one of the signals in Table 6.2.4-1. Although the Phase B isolation signal is not actuated by diverse parameters, it is acceptable because the affected lines are considered important to the safe shutdown of the plant and are capable of remote manual isolation. The staff concludes that adequate diversity has been provided with regard to the different monitored parameters which actuate containment isolation.

TABLE 6.2.4-1
CONTAINMENT ISOLATION SIGNALS
AND ACTUATION PARAMETERS

Containment Isolation Phase A signal

a. High Containment Pressure (Hi-1)

b. Low Compensated Steam Line Pressure

c. Pressurizer Low Pressure

d. Manual Actuation

Containment Isolation Phase B Signal
a. High Containment Pressure (Hi-3
b. Manual Actuation

~~Containment Isolation Phase B Signal~~
Safety Injection Signal
a. High Containment Pressure (Hi-1)

b. Low Compensated Steam Line Pressure

c. Pressurizer Low Pressure

d. Manual actuation

Main Steam Isolation Signal

a. High Steamline Pressure Rate

b. High Containment Pressure (Hi-2)

c. Low Steamline Pressure

d. Manual Actuation

Feedwater Isolation Signal

a. Steam Generator Hi-Hi Water Level

b. Safety Injection Signal

c. Low TAVG and Reactor Trip

Containment Vacuum System Isolation Signal

a. Containment Isolation Phase A Signal (Hi-1)

b. Manual Actuation

The staff has reviewed the applicant's containment isolation system design bases and containment isolation provisions as documented in Table 6.2-60 of the FSAR, for conformance to General Design Criteria (GDC) 54, 55, 56 and 57 and Regulatory Guide 1.11, "Instrument Lines Penetrating Primary Reactor Containments". The applicant's containment isolation system design is summarized as follows:

(1) There are at least two barriers between the atmosphere outside containment and the atmosphere inside containment (or the RCS) on each system line penetrating the containment.

(2) The two barriers consist of one of the following arrangements:

a. two normally closed manual valves with administrative control, one inside containment and the other outside containment;

b. two automatic isolation valves, one inside containment and the other outside containment, a simple check valve may not be used as the automatic isolation valve outside containment;

c. one automatic isolation valve inside containment and one normally closed manual valve under administrative control outside containment (or the reversed arrangement);

d. a sealed system (closed system) inside containment and one isolation valve outside containment, which is either automatic, remote manual, or manual under administrative control.

(3) Isolation valves of the ESF related systems, which are essential to mitigate the effects of an accident, remain open or move to their open position post-accident. These valves are remote manually controlled and operated from the control room.

(4) Motor operated valves (MOV) are used for system lines which are part of an ESF related system, and fail "as is" on loss of power supply. Solenoid operated valves are used when greater reliability post-accident and a safe-failure position are required. All power operated valves are designed to fail in the position that provides greater safety upon loss of power or control air.

(5) Mechanical and electrical redundancy are provided by designing two isolation barriers between the RCS or atmosphere inside containment and the atmosphere outside containment with two separated IE power sources.

(6) Containment purge system isolation is accomplished with ~~uses~~ two 42-in. butterfly valves, which are only open during plant cold shutdown and close~~d~~ automatically within 10 seconds upon receipt of a high radiation signal.

(7) The containment isolation system is designed to meet the single failure criterion.

(8) The closure time for each containment isolation valve is less than 60 seconds. System lines which have no post-accident function are provided with air-operated valves (AOV) with a closure time of 10 seconds.

The applicant's containment isolation provisions are reviewed against the requirements of GDC 54, 55, 56, and 57 (Appendix A to 10 CFR Part 50) and the supplementary guidance of SRP 6.2.4, where applicable. Staff~~'s~~ review has confirmed that the containment isolation system meets the explicit requirements of GDC 54, 55, 56, and 57 with the following exceptions:

(1) The containment vacuum pump and hydrogen recombiner suction lines are provided with two solenoid-operated isolation valves in series <u>outside</u> containment. Therefore, the containment isolation provisions differ from the explicit requirements of GDC 56. However, the isolation valves are located as close as possible to the containment, and the associated system piping is designed in accordance with the break/crack exclusion criteria of Branch Technical Position MEB 3-1. Furthermore, the valves are hermetically sealed, precluding the need to encapsulate the valves. Since the lines are used post-accident, for containment atmosphere sampling and hydrogen control, locating the valves outside containment improves the functional reliability of the valves. Therefore, the staff finds the isolation provisions for these lines to be acceptable alternatives to the explicit requirements of GDC 56.

(3) The emergency core cooling system safety injection lines and reactor coolant pump (RCP) seal injection lines are equipped with weight-loaded check valves inside containment and motor-operated valves (MOV), outside containment which do not receive a containment isolation signal to close. The safety injection lines discharing to the hot and cold legs of the reactor coolant system and the RCP seal injection lines are important to safe shutdown or are part of an engineered safety feature system.

Provisions have been made to detect possible leakage from these lines outside containment, thereby allowing remote manual instead of automatic isolation valves. The staff, therefore, finds that the containment isolation provisions for these lines are acceptable alternatives to the explicit requirements of GDC 55.

(4) The quench spray pump discharge and recirculation spray pump discharge lines are provided with a normally open, remotely-controlled, motor operated valve outside containment and a weight-loaded check valve inside containment. The isolation valves in the containment depressurization (quench and recirculation spray) systems open upon receipt of a CIB signal, if not already open, with the exception of the caustic addition line to the containment sump which automatically opens after the quench spray discharge has stopped. The recirculation spray pump suction lines are provided with a single, normally open, remotely-controlled, motor operated valve outside containment since it is not practical to locate a second valve inside containment where it would be submerged following a LOCA; these valves do not

receive an automatic isolation signal for closure. Therefore, the containment isolation provisions for these lines differ from the explicit requirements of GDC 56 regarding their actuation and number.

These lines are part of ESF systems, and are required to be open to perform their post-accident safety function. The ESF systems are closed outside containment, and are safety grade. Therefore, the staff finds the use of remote-manual instead of automatic isolation valves acceptable. In addition, the single isolation valve outside containment in the recirculation spray pump suction lines is acceptable because system reliability is improved with a single valve and the piping between the outside of the containment wall and the isolation valve, as well as the valve, are contained within a leak-tight encapsulation.

The staff has also reviewed information provided by the applicant to demonstrate compliance with the provisions of NUREG-0737 Item II.E.4.2, "Containment Isolation Dependability". As previously described, the applicant has complied with the provisions regarding diversity in parameters sensed for initiation of containment isolation, and has considered the functional the functional requirements of all systems penetrating containment and has made acceptable provisions for isolation of systems not required for mitigation of the consequences of an accident or safe shutdown of the plant. The applicant also made provions that resetting of a containment isolation signal will not result in the automatic reopening of containment isolation valves. In addition, the applicant has designated all system lines penetrating the containment as essential or non-essential systems by appropriate signals. Therefore, the staff concludes that the applicant has complied with the provisions of NUREG-0737 Item II.E.4.2.

The applicant has stated that all containment isolation barriers as well as electrical and control components required for initiation are protected from missiles and the effects of natural phenomena to ensure their performance under all anticipated environmental conditions. The staff, therefore, finds that the containment isolation system meets the requirements of GDC 1, 2, and 4. The containment isolation system also meets the provisions of Regulatory Guide 1.29, "Seismic Design Classification", and 1.26, "Quality Group Classifications and Standarsds for Water-, steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants.".

In summary, the staff has reviewed the information in the applicant's FSAR and in response to NRC Questions concerning the containment isolation system to assure conformance to all of the acceptance criteria contained in SRP Section 6.2.4. The staff concludes that the Beaver Valley 2 containment isolation system meets the requirements of General Design Criteria 1, 2, 4, 16, 54, 55, 56, and 57, and is, therefore, acceptable.

6.2.5 Combustible Gas Control System

Following a loss of coolant accident, hydrogen may accumulate as a result of (1) metal-water reaction between the zirconium fuel cladding and the reactor coolant, (2) radiolytic decomposition of the water in the reactor core, (3) radiolytic decomposition of the water collected on the sump floor, (4) hydrogen released from the pressurizer gas space and reactor coolant, (5) corrosion of metals by the alkaline solution used for containment spray. The function of the combustible gas control system (CGCS) is to monitor and control the potential hydrogen accumulation within the containment atmosphere below 4-volume percent following a design basis accident.

In the event of a LOCA, two redundant, independent, full capacity electric hydrogen recombiners will be available outside containment to control the containment hydrogen concentration. Each recombiner has a capacity of 50 SCFM and is designed to Seismic Category I criteria. One hydrogen recombiner is permanently installed in the safeguards area; the other recombiner will be transferred from Beaver Valley, Unit 1 and installed in the safeguard area following an accident. (In addition to the two safety related hydrogen recombiners provided, a non-safety grade containment purge system is available to purge the containment atmosphere as an aide to cleanup.) Each hydrogen recombiner system includes flow control capability, a blower, a temperature-controlled electric preheater, a thermal recombiner, and an air blast heat exchanger. The safeguards area is a Seismic Category I concrete structure located adjacent to the containment. The penetrations, and components within the safeguard area are protected against tornados and missiles. The hydrogen recombiners and all associated valves are remote manually controlled from panels located in the safeguards area, outside of the recombiner cubicles, to allow access and minimize exposure of personnel. The staff has reviewed the hydrogen recombiner system design concept and finds it acceptable.

Two redundant, independent hydrogen analyzers are installed in the cable vault area to monitor the hydrogen concentration in the containment atmosphere. The analyzers are also used to check the efficiency of recombiner operation. The hydrogen analyzer is classified as Class IE, Seismic Category I and functional tested with a calibrated gas sample. Indicators are provided in the main control room to monitor hydrogen concentration. Annunciation is also provided in the main control room for hydrogen analyzer/recombiner local panel trouble. Based on the staff's review, the post-accident hydrogen monitoring system meets the requirements of NUREG-0737 Item II.F.1, Attachment 6, "Containment Hydrogen Monitor", and the single failure criterion. However, the applicant has not required a sufficiently complete description of the operating characteristics of the hydrogen analyzer to be installed.

The applicant has analyzed the potential hydrogen generation within the containment using the guidelines provided in Regulatory Guide 1.7, and calculated the hydrogen concentration for both one and two recombiner operation. The analysis shows that a single recombiner, initiated when the containment hydrogen concentration reaches 3.1 volume percent (i.e., approximately 4 days post-accident), is sufficient to maintain the hydrogen concentration in the containment atmosphere below the lower flammability limit of 4 volume percent. The design of the Beaver Valley, Unit 2 containment is similar to the Beaver Valley, Unit 1 and Surry containments, which use recombiners. The staff has previously confirmed, using the COGAP computer code, that there is sufficient time before the containment hydrogen concentration reaches 3.1 volume percent to manually initiate the post-accident hydrogen recombiners, and that a single recombiner can acceptably control the hydrogen concentration in containment below 4.0 volume percent.

The applicant has stated in the FSAR that the containment design allows air to circulate freely. Furthermore, all cubicles and compartments within the containment are provided with openings near the top as well as openings in the floor to allow air circulation. The applicant has also performed an analysis to demonstrate that adequate mixing of the hydrogen in the containment atmosphere will be ensured by the turbulence created by the containment spray system and thermal convection. Therefore, sufficient mixing of hydrogen in containment will occur to prevent stratification and to eliminate areas of potential stagnation. The staff finds that adequate passive and/or active design measures have been incorporated into the containment design to ensure adequate hydrogen mixing within containment and, therefore, the applicant's hydrogen mixing provisions are acceptable.

In summary, the staff has reviewed the information in the applicant's FSAR and in response to our questions concerning the combustible gases control system to assure conformance to all of the acceptance criteria contained in SRP Section 6.2.5. The staff concludes that the applicant's combustible gas control system meets the requirements of GDC 41, 42 and 43, satisfies the design and performance requirements of 10 CFR 50.44, the provisions of Regulatory Guide 1.7 and the requirements of NUREG-0737 Item II.F.1, Attachment 6, except for the following item. The applicant has not discussed in sufficient detail the performance characteristics of the actual post-accident hydrogen monitoring system to be installed. Therefore, this will remain an open item pending the receipt of additional information.

6.2.6 Containment Leakage Testing Program

The containment design includes the provisions and features required to satisfy the testing requirements of Appendix J to 10 CFR Part 50. The design of the containment penetrations and isolation valves permit preoperational and periodic leakage rate testing at the pressure specified in Appendix J to 10 CFR 50.

The staff has reviewed the containment leakage testing program contained in the FSAR and in the response to NRC Questions, and finds them acceptable with the following exception. The applicant proposes to exclude certain valves from Type C testing (including the safety injection system penetrations and recirculation spray system penetrations). The justification for excluding penetrations from

Type C testing will be evaluated in conjunction with the staff review of the facility Technical Specifications.

Other than the exception mentioned above, the proposed reactor containment leakage testing program complies with the requirements of Appendix J to 10 CFR Part 50. Such compliance provides adequate assurance that containment leak-tight integrity can be verified periodically throughout service lifetime on a timely basis to maintain such leakage within the limits of the Technical Specifications.

Maintaining containment leakage rates within such limits provides reasonable assurance that, in the event of any radioactivity releases within the containment, the loss of the containment atmosphere through the leak paths will not be in excess of acceptable limits specified for the site. Compliance with the requirements of Appendix J constitutes an acceptable basis for satisfying the requirements of General Design Criteria 52, 53 and 54.

2.5.4 Stability of Subsurface Materials and Foundations

2.5.4.1 Site Foundation Conditions

2.5.4.1.1 General Site and Plant Description

The Beaver Valley Power Station Unit 2 (BVPS-2) is located on the south bank of the Ohio River approximately 25 miles ~~southwest~~ northwest of Pittsburgh, Pa. The major structures of BVPS are built on the highest of three Pleistocene terraces that are composed predominantly of alluvial deposits. These deposits were derived from the in-place weathering of local materials which were transported by glacial outwash by the ancestral Ohio River drainage system during the Pleistocene period. Sequential deposition, erosion and subsequent deposition formed the terraces at the site. The surface of the upper terrace slopes gently toward the Ohio River from about elevation 760 feet to 735 ft. The soils of this terrace consist predominantly of interbedded sands, gravels, and silty sands and gravels. A steep natural slope originally separated the upper terrace from the intermediate terrace and a gentle natural slope separates the intermediate terrace from the lower (floodplain) terrace. (FSAR Fig. 2.5.4-1). The intake structure is located north of the main structure on the floodplain of the Ohio River. The near surface soils of the intermediate terrace with original ground surface at el 685 to 700 ft and the present flood plain with original ground surface el 675 ft consist of medium stiff to soft clays and silts. These recent river silts and clays extend to approximately el 655 ft and are underlain by sand and gravels down to bedrock at about el 620 ft. Parts of the intermediate terrace are overlain by fill placed during the construction of Shippingport Atomic Power Station (SAPS) and BVPS-Unit 1.

The bedrock in the general area of the site consists of interbedded sandstones, shales, coal seams, and occasional limestones. The rock underlying the plant site is a dark gray carbonaceous shale that dips gently southeastward. It is slightly weathered for the first few feet; the weathering effects rapidly decrease with depth.

The seismic Category I structures, systems and components (SSC) for the BVPS Unit 2 that were reviewed are listed in FSAR Table 3.2-2 and include: reactor containment building, auxiliary building, fuel and decontamination buildings, diesel generator building, service building, main steam and cable vault, safeguards area, refueling water storage tank, primary demineralized water storage tank, primary intake structure, buried pipelines, pipe tunnels and emergency outfall structure. Figure 2.4.2 of this SER shows a general layout of the plant facilities.

The original ground surface in the main plant area ranged from about elevation 735 ft to 760 ft (msl). The final plant grade is at el 735 ft. The bottom of excavation for the power block structures was above el 665 ft, except for a local area within the containment cofferdam. All Category I structures are founded either on natural terrace of gravelly sand and sandy gravel or on select granular backfill. The groundwater level at the site is el 665 ft, the same as the normal level of the adjacent Ohio River.

Both normal cooling water and emergency cooling water are obtained from the Ohio River and pumped from the primary intake structure through two 30" diameter service water supply lines, as shown in FSAR Figures 2.5.4.54. These pipelines are supported on select granular backfill up to the Valve Pit. The intake structure is founded in the lower terrace section and is directly adjacent to the Ohio River and about 600 ft from the main plant area. The founding elevation of this structure varies between el 634.5 and 640.5 ft.

2.5.4.1.2 Properties of Subsurface Materials

(a) Field Investigations

The subsurface conditions at the site were investigated by drilling exploratory borings, installing piezometers, and performing geophysical surveys. Approximately 300 borings were drilled for the construction of SAPS, BVPS-1 and BVPS-2. The applicant also used borings made by others for a sludge pipeline system and for the BVPS emergency response facility. Figure 2.5-1 of this report shows a generalized subsurface profile based on the data derived from the borings. A subsurface profile across the northern portion of Reactor Containment Structure is shown in Fig. 2.5-2.

In addition to the original subsurface investigations described above, three sets of borings were drilled to verify the effectiveness of soil densification performed during construction in the following locations:

(1) northern half of the containment building and extending east and west beneath most Category I structures,

(2) northern part of the area along the 30-inch service water lines from the intake structure to the Valve Pit, and

(3) two areas on the east and the west sides of the intake structure.

The question of effectiveness of these soil densification programs was discussed with the applicant at some length by the staff at the PSAR stage and the staff was generally satisfied with the applicant's documentation of the data supporting their claim of effectiveness of densification. However, the staff has now requested the applicant to provide confirmatory analyses for the areas near the intake structure and along the service water lines as discussed in Section 2.5.5 of this SER. The applicant has agreed to furnish longitudinal sections of all Category I pipelines (1) from the Valve Pit No. 1 to the main plant structures, and (2) from the main plant area to the Emergency Outfall structure. These sections should show the soil profile and the static and dynamic soil properties used in the pipe stress analysis, such as the subgrade modulus, shear wave velocity, shear modulus, etc.

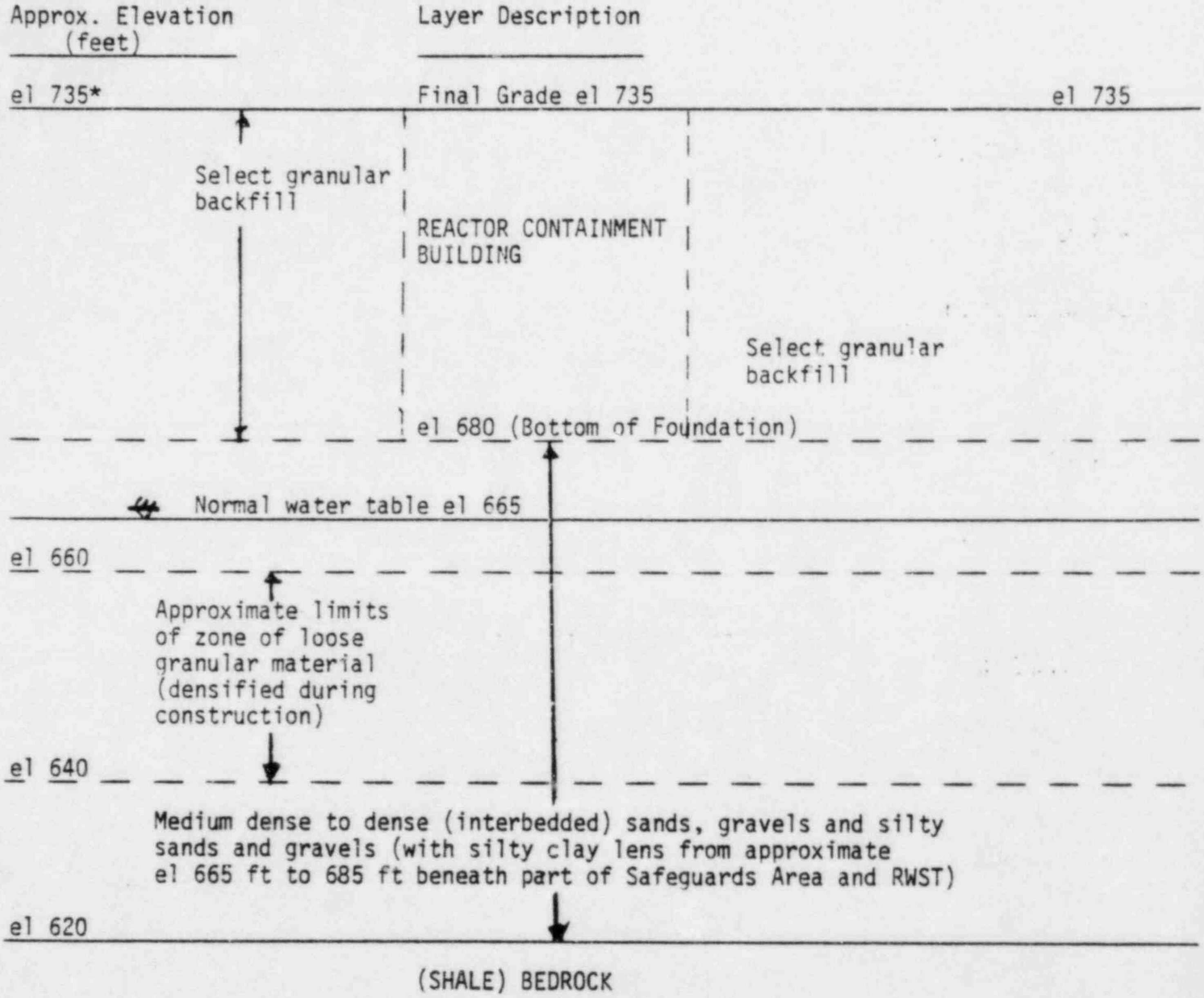
Six piezometers were installed for studying the groundwater table locations at the site. In 1968 and again in 1977, the applicant's consultant (Weston Geophysical Engineers) conducted geophysical surveys at the site to measure the in situ compression and shear wave velocities of the foundation soil and rock. As seen in FSAR Fig. 2.5.4-17, the seismic compression (P) wave velocity of the undisturbed in situ soil in the general vicinity of the densified zone measured in 1968 ranges from about 1,500 ft per second (fps) at el 730 to about 2000 fps at el 680 ft, which is above the ground water table. The geophysical survey conducted in 1977 indicated a P-wave velocity of the in situ soil densified by PIF technique ranging from 2000 fps at el 685 ft to about 2500 fps at approximately el 665 ft. The corresponding shear (S) wave velocities range from about 900 fps to 1050 fps in the 1968 survey and from 700 to 1000 fps in the 1979 survey.

There are anamolies in the 1977 seismic survey shown in FSAR Fig. 2.5.4-17 concerning the elevation of the ground water table in 1977 and the P-wave velocity below the water table. The applicant has confirmed that the groundwater table in 1977 was at approximately el 665 ft and not at el 652 as shown in that figure. The applicant has agreed to revise the FSAR, and the anamolous value of 3000 fps for the P-wave velocity between el 652 and el 665. The correct P-wave velocity below the water table at el 665 ft must be about 5000 fps (which is the P-wave velocity of water).

The measured P-wave and S-wave velocities of the (shale) bedrock below el 620 are 12000 fps and 6000 fps in the general vicinity of the densified zone.

(b) <u>Subsurface Profile</u>

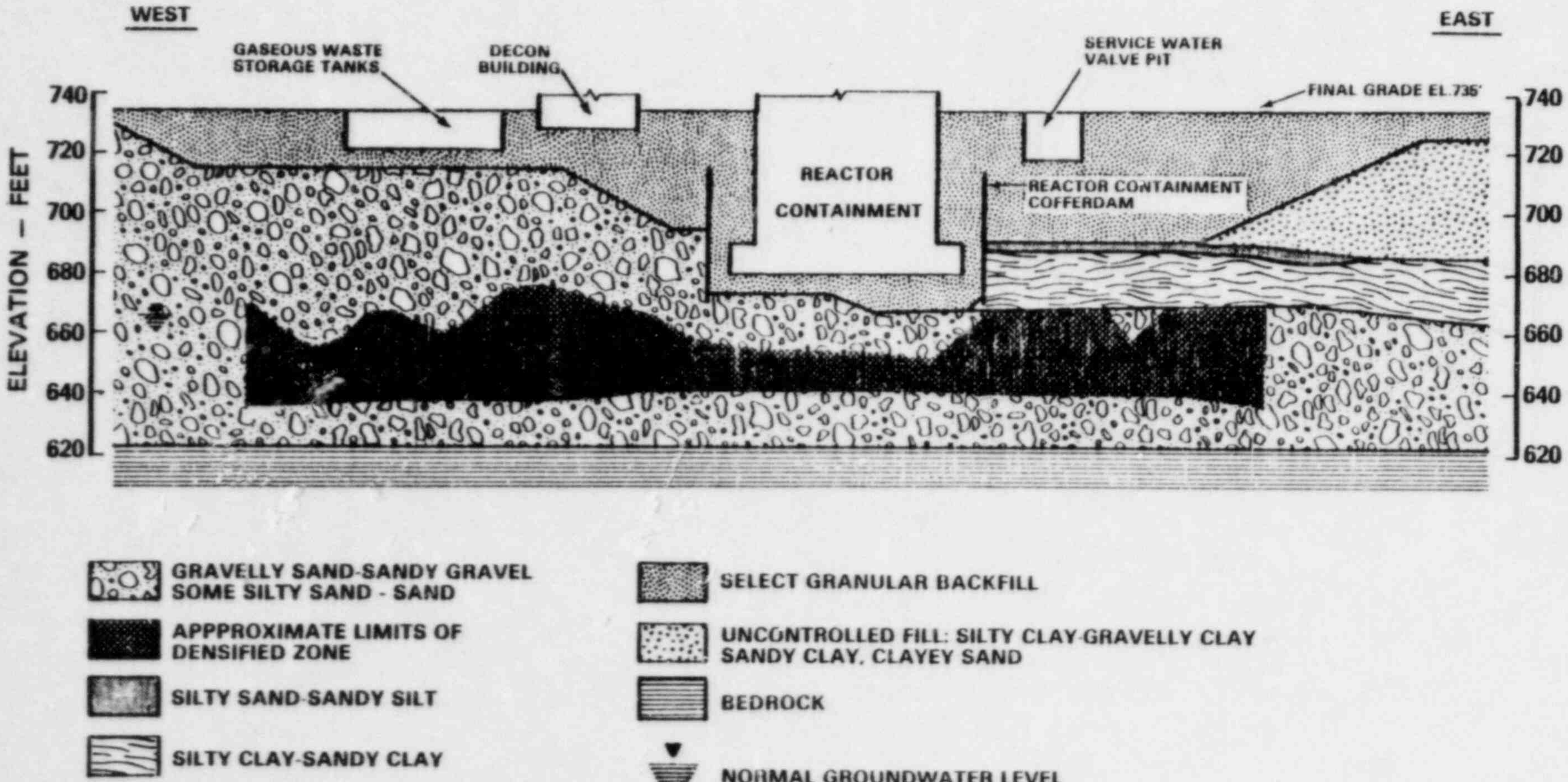
As shown in Fig. 2.5-1, the subsurface profile at the main plant area consists of about 115 feet of medium dense to dense granular soils (interbedded sands, gravels, and silty sands and gravels), underlain by shale bedrock at about el 620 ft. A zone of loose granular material was discovered in the containment excavation between about el 640 ft and el 660 ft and was densified under Category I structures using the pressure injected footing technique



*NOTES:

1. Original grade ranged from el 760 to 735 ft.
2. Backfill depth varies from structure to structure.
3. Zone of loose granular material existed only at the northern portion of the power block area.

Fig. 2.5-1 Generalized Subsurface Profile at the BVPS-2 Power Block Area



NOTE: This section represents the conditions existing at the northern portion of containment; The silty clay layer does not exist at other sections through the structure.

Figure 2.5-2 Subsurface Profile Across Containment Structure (REF: FSAR Fig. 2.5.4-3)

as discussed in Reference 1. A lens of very stiff, silty clay was also noted at about el 679 ft during the excavation for the northern portion of the Reactor Containment. (The containment mat foundation is founded at about el 680 ft.) This silty clay lens was not found during the original subsurface investigation. It extends eastward to areas under the northern portions of the Safeguards area and the Refueling Water Storage Tank (RWST). At the RWST, the top surface of this silty clay layer is at approximate el 688 ft; it is about 20 ft thick at the northern edge of the Safeguards area and about 10 ft thick at the northern edge of the RWST. This clay layer thins to the south and is not present at about the east-west center line of the Safeguards area. FSAR Table 2.5.4-1 lists the boring logs that provide the data from which the subsurface profiles at the site were determined.

The soil profile in the intake structure area consists of clay and silty clay from ground surface at el 675 ft to about 655 ft, and sands and gravels with lenses of loose materials susceptible to liquefaction from el 655 ft to bedrock at el 620 ft.

(c) Laboratory Investigations

The applicant performed laboratory tests to determine the engineering and index properties of the intermediate and lower terrace lenses of silts and clays. FSAR Appendix 2.5D presents the results of these laboratory investigations. Soil classification tests and consolidation tests were performed on undisturbed block samples of the stiff silty clay lens that was encountered just below the reactor containment base mat. The average value of liquid limit of the silty clay was 50; the average value of the plastic limit was 23 and was equal to the natural water content, thus indicating precompression of this clay. From the consolidation tests the following average values were derived for consolidation settlement calculations:

Compression index	= 0.12
Recompression index	= 0.02
Overconsolidation ratio (OCR)	= 1.3 to 2.4
Coefficient of consolidation (C_v) =	
less than preconsolidation pressures	5×10^{-3} to 1.8×10^{-2} cm^2/sec
greater than preconsolidation pressures	2.5×10^{-3} cm^2/sec
Coefficient of secondary consolidation	= 5×10^{-4} to 2×10^{-3}

The consolidated undrained triaxial compression (CIU) test results indicated for the silty clay an effective friction angle of 25.7° assuming that effective cohesion intercept was zero. The unconsolidated undrained (UU) triaxial compression tests showed the undrained shear strength to be approximately 4.3 ksf.

The results of the grain size analyses performed on samples of the in situ sands and gravels and the results of the inplace density tests on the soils at the Reactor Containment foundation elevation are also given in FSAR Appendix 2.5-D. The following are the average properties of the in situ sands and gravels:

Dry unit weight	= 117 pcf
Specific gravity	= 2.65
Void ratio	= 0.4
Saturated unit weight below groundwater table (G.W.T)	= 136 pcf
Total unit weight above G.W.T. assuming an average water content of 7%)	= 125 pcf

The engineering properties of the in situ sands and gravels were not determined by laboratory tests because undisturbed samples of these granular materials could not be obtained. The applicant estimated, for design purposes, the engineering properties of these materials by using accepted empirical correlations of these properties to subsurface conditions determined by test borings, geophysical surveys, and field testing (Reference 7).

Standard Penetration Tests (SPT) were performed in conjunction with borings outside of the area densified by the pressure-injected footing (PIF) technique. The applicant used the relationship between relative density and SPT blow counts (N) suggested in Reference 5 and determined that the in situ sands and gravels could be classified as medium dense to dense. Although the relative density of these materials indicate angles of internal friction, ranging between 33 and 40 degrees, the applicant has conservatively chosen an angle of 30 degrees for design purposes.

For the purpose of estimating static settlements of buildings, the applicant used an empirical relationship (Reference 3) for the low strain shear moduli of the in situ sand and gravel. These shear moduli values compared reasonably well with moduli calculated from in situ seismic velocity measurements. For example, at the elevation of 680 ft the empirical relationship indicated a value of about 4.5×10^3 ksf compared to the measured value of about 4.1×10^3 ksf for the low strain shear modulus. (FSAR Fig. 2.5.4-~~18~~17).

But for the applicant's inability to obtain and test undisturbed samples of granular materials in the main plant area, his testing program is in accordance with Regulatory Guide 1.138, "Laboratory Investigations for Engineering Analysis and Design of Nuclear Power Plants". Where undisturbed samples of granular materials were not obtained and tested, as recommended by the SRP Section 2.5.4, the applicant has determined the engineering properties of these materials by acceptable procedures. Therefore, the staff has concluded that the applicant has adequately investigated and analyzed the subsurface conditions and established appropriate subsurface material properties for foundation design and soil-structure interaction analysis.

2.5.4.1.3 Groundwater Conditions

The applicant recorded the groundwater level readings in four temporary observation wells installed at the bottom of the excavation for the reactor containment foundation and the Ohio River elevations in the Spring of 1976. A comparison of these readings showed that there was essentially no time lag between the elevation of the Ohio River and the groundwater level in the observation wells. Falling head permeability tests conducted in three of the observation wells gave coefficient of permeability values ranging from a minimum of 0.9×10^{-3} to a maximum of 3.9×10^{-3} cm/sec. Six piezometers were installed at different locations of the site in 1977 as part of the settlement monitoring program. The groundwater levels recorded in the piezometers show good correlation with the Ohio River elevations. Based on these observations, the applicant has assumed, and the staff agrees, that the groundwater level at the plant site area agrees with the various stages of the Ohio River. Thus, the design basis groundwater level is elevation 665 ft during normal water level in the Ohio River, 690 ft during a 25-yr flood, and 705 ft during standard project flood. Section 2.4 of this SER contains a more detailed discussion of the groundwater conditions and also discusses the design basis for subsurface hydrostatic loading.

2.5.4.2 Excavation and Backfill

2.5.4.2.1 Excavation and Foundation Treatment

Dewatering was not required during excavation because the bottom of all excavations in the main plant area were above the normal ground water level of el 665 ft except for a local area within the containment cofferdam (see FSAR Fig. 2.5.4-19).

Waste material had been placed over portions of the intermediate terrace and floodplain at the BVPS-2 site during construction of the Shippingport Atomic Power Station and the BVPS, Unit 1. This material was removed from the BVPS-2 area by excavating to el 690 ft north and east of the Reactor Containment area and replaced by compacted select granular fill to plant grade of 735 ft. The excavation for the containment structure was made to el 679 ft within a steel sheetpile cofferdam driven to el 671 ft. On the east and west sides of the containment cofferdams, the excavation was carried to el 700 ft and 703 ft respectively, while on the south and northwest sides the excavation was to el 715 ft. The excavation slopes were generally 1.5 horizontal to 1 vertical.

The suitability of the materials beneath the excavated foundation levels for Category 1 structures, buried piping, and duct lines was verified by performing in-place density tests using Washington densometer (ASTM D2167) and/or nuclear densometer (ASTM D2922) and by removing any soft spots at the bottom of excavations and backfilling with compacted fill.

After excavating the containment area to the required depth, a foundation documentation program was carried out by establishing a 25-ft square grid over the floor of the excavation and performing in-place density tests at each grid intersection. Bag samples of soil were also obtained for classification purposes at the grid intersection points. As a result of the foundation documentation program, the applicant found a stiff silty clay lens along the northern perimeter of the containment excavation at el 679 ft. The silty clay lens had a chord length of about 100 ft and a maximum width of about 30 ft. In order to remove the silty clay lens the containment excavation was deepened to about el 674 ft over the area where the silty clay lens occurred. It was observed that the silty clay lens extended below el 674 ft. As the applicant's investigations indicated that the silty clay posed no engineering problems, the applicant stopped further excavation of this material and filled up the over-excavation with lean concrete backfill (Ref. 1). The use of lean concrete in place of the approved granular backfill was questioned by the Region I inspector in Inspection and Enforcement Report No. 50-412/76-02 dated May 26, 1976. Furthermore, since the presence of the silty clay was not reported in the PSAR, the applicant was asked to investigate the extent of this material at the site.

Six borings were drilled within the reactor containment cofferdam to determine the thickness of the silty clay that was left beneath the lean concrete plug. This investigation revealed that a zone of loose granular material existed from approximately el 640 to 660 ft under roughly the northern portion of the containment and extended east and west beneath most of the Category I structures. The extent of the unacceptably loose zone was defined from exploratory borings. A significant number of borings had corrected standard penetration test (N_1) values less than 10 determined by the Gibbs and Holz (Reference 2) relationship. The applicant densified the loose materials by the pressure injected footing (PIF) technique as reported in Reference 1. The staff reviewed this densification program during the construction stage and found it acceptable. The purpose of this densification was to preclude liquefaction of the loose granular material and dynamic settlement of structures during the SSE.

Soil densification by the PIF technique was accomplished by first conducting a feasibility investigation in which 24 PIF's were installed in four test-panels and the resulting degree of soil densification verified by using conventional boring and sampling techniques. The PIF is basically a type of compaction pile (with a concrete shaft) that derives increased bearing capacity by densifying the soils around an expanded base. At the BVPS-2 site, a modified PIF technique was used that densified the loose soil by both volume displacement and dynamic energy input. The material used for the shaft was ordinary portland cement concrete. The concrete shafts were, however, not continued up to the bottom of the foundation mat to preclude a rigid connection between the PIF and the overlying mat foundation as shown in Fig. 2-3 of Reference 1. Concrete was used in the shafts only in the loose zone from about el 640 ft to 660 ft, and compacted granular material was used to backfill the shafts from about el 660 ft to about el 680 ft. The PIF concrete shafts were spaced at a 7.5 ft triangular grid pattern as shown in Fig. 4-1 of Reference 1. The volume of concrete injected into a PIF shaft is approximately 6.3 cubic ft per ft depth of the shaft so the equivalent diameter of the shaft is about 2.83 ft.

Having been satisfied with the results of the feasibility study, the applicant intsalled a total of 1271 PIF's througnout the affectd site area between September 1976 and August 1977. As seen from FSAR Fig. 2.5.4-15, soil densification by the PIF technique was done below the following major Category I structures: reactor containment, fuel and decontamination buildings, auxiliary building, diesel generator building, safeguards building and refueling water storage tank. Only about one-half of the areas beneath the reactor containment, auxiliary building, and diesel generator building were densified.

The effectiveness of the soil densification was demonstrated by drilling a total of 164 verification borings. The results of the verification borings indicated that significant increases in the SPT (N_1) values were achieved as compared with the N_1 values obtained before densification. In the containment area test panels, the corrected mean blow count, less one standard deviation, in the loose zone was 9.4, and the corrected mean blow count, less one standard deviation, after densification was 24.3. Similar increases in the blow counts were observed in the test panels outside the containment area.

The applicant removed the silty clay layer found in the reactor containment and the concrete plug referred to earlier. This has been verified by the NRC Region I inspector in the IE Inspection Report No.

50-412/77-03 dated April 11, 1977. Subsequent borings in the main plant area revealed that the silty clay layer extended under the northern-most one-half of the Safeguards Building and the Refueling Water Storage Tank foundations. The maximum thickness of the silty clay layer was approximately 12 ft under the northern edge of the Safeguards Building (Reference 1).

A sheet pile cofferdam was driven to bedrock to facilitate construction of the intake structure. Two rows of sheetpile walls (that are tied together) extend along the river to the east and west of the intake structure. The river bottom directly in front (and north) of the structure was dredged to el 645 ft with an average side slope of approximately 3.5 to 1.

2.5.4.2.2 Backfill

Well graded sand and gravel (SW and GW) was used as the select granular backfill beneath and adjacent to Category I structures. The backfill material conformed to the following grain size requirements:

Sieve Size	Percent Passing by Dry Weight
6 (inches)	100
No. 200	0-15 (nonplastic fines)

The fill material was placed in loose lifts of 6 to 12 inches and compacted to a minimum of 95% of the maximum dry unit weight obtained from compaction tests performed in accordance with ASTM D1557, Method D, with a minimum required in-place dry density of 130 pcf.

For design purposes, the following soil properties were used for the select compacted fill, based on laboratory test results:

Dry unit weight	= 130 pcf
Specific gravity	= 2.65
Void ratio	= 0.27
Saturated unit weight	= 144 pcf
Total unit weight above water table (moisture content 5%)	= 136 pcf
Angle of internal friction	= 36°

The low strain shear moduli of the select granular fill were estimated using equations available in Reference 3. The dynamic properties of the fill material are discussed in Section 2.5.4.3.4, Seismic Loading.

2.5.4.3 Stability of Foundations

2.5.4.3.1 Design Criteria

The applicant used state-of-the-art procedures to analyze the foundation stability of Category I structures and systems. The minimum safety factor for bearing capacity used for the design of these facilities was 3.0 for all loading conditions.

The staff requires that applicant must also consider the loading combination of OBE and standard project flood in all stability analyses, as recommended in SRP 2.4.4. We expect to report our evaluation of this matter in the final SER.

2.5.4.3.2 Bearing Capacity

All Category I structures are founded on reinforced concrete mat foundations. FSAR Table 2.5.4.4 gives the approximate plan dimensions, the applied foundation loads, and the ultimate bearing capacity of each foundation. Table 2.5-1 of this SER gives the plan dimensions, mat

TABLE 2.5-1

Foundation Data for Major Category I Structures

	Approximate Dimensions of Contact Area (ft)	Approximate mat elevation (ft)	Approximate Bearing Pressure Static (ksf)	Dynamic (ksf)
Auxiliary building	120 x 146	703.0	5.7	10.6
Control room extension	65 x 81	703.0	3.5	5.6
Decontamination building	33 x 33	729.5	6.3	11.5
Diesel generator building	81 x 83	713.0	3.1	5.9
Fuel building	44 x 110	717.3	6.3	11.5
Reactor containment	142 dia.	681.0	7.5	12.4
Safeguards area	60 x 96	714.5	3.2	4.7
Service building	55 x 186	724.5	4.0	4.6

(Reference: FSAR Table 2.5.4-4)

elevations, and approximate bearing pressures for the foundations of major Category I structures. Since the mat foundations are embedded in dense sands and gravels, the ultimate bearing capacity is quite high,ranging from 33 ksf for the decontamination building to 129 ksf for the auxiliary building. The calculated static foundation stresses range from 2.5 ksf to 7.5 ksf - the upper value being the foundation pressure beneath the Reactor Containment Building. Therefore, the factor of safety against a bearing capacity failure is typically very high.

In response to OL question 241.9, the applicant has informally furnished a revised copy of FSAR Table 2.5.4-4 incorporating the dynamic foundation loads therein. The foundation stresses including the effects of dynamic loads range from 3.8 ksf to 12.4 ksf. The applicant has not revised the factors of safety shown in that table, although the proposed revision ~~will~~ should not alter the above conclusions regarding the high safety factors against a bearing capacity failure. The applicant is expected to docket the revised FSAR Table 2.5.4-4 with corrected safety factors.

The information concerning the foundation dimensions and the bearing capacity of the main intake structure are not included in Table 2.5.4-4. The applicant has been requested to include the foundation data concerning the intake structure in revised FSAR Table 2.5.4-4.

2.5.4.3.3 Settlement

Foundation soils supporting structures and components in the main plant area consist of compacted select coarse grained fill and medium dense to dense in situ coarse grained materials. A layer of fine grained silty clay underlies the foundation soils beneath the northern portions of the Safeguards area and recirculating water storage tank (RWST). The applicant has calculated the total static and potential dynamic settlements of the Category I structures in the main plant area using the soil properties discussed in Section 2.5.4.1.2 above.

The applicant has evaluated the potential dynamic settlements of the Category I structures using the procedures suggested by Reference 4, the magnitude of these dynamic settlements range from 0.09 in (Reactor Containment) to 0.16 in. (Service Building). These dynamic settlements are not significant compared to the static settlements as seen from the following discussion.

The total static settlement of structures founded on granular soils was assumed to consist of two components: an elastic (immediate) settlement, and a time-dependent settlement taken equal in magnitude to the elastic component. The settlement of the clay layer underlying the Safeguards area and the RWST was determined using the one-dimensional consolidation theory. The total settlement of structures with a clay layer beneath them was estimated by adding the consolidation settlement of the clay layer to the elastic settlement of the in situ sand and compacted fill.

Based on the settlement data given in the FSAR Figs. 2.5.4-20, and 2.5.4-46, Table 2.5-2 gives a comparison of the estimated and measured total settlement at the corners of a few Category I structures. This table also gives the percentage of structural loading of these structures as of January 10, 1984 provided by the applicant during the geotechnical audit by the staff in January 1984. The applicant has been requested to docket this data, including a comparison of up-to-date measured settlements with predicted values, in a tabular form in the forthcoming amendment of the FSAR. The applicant needs to provide the total loading including the equipment loading, etc. in addition to civil, steel and concrete loading.

The differential settlements between adjacent structures, and across the structure foundation, can be estimated from the predicted total settlements given in FSAR Fig. 2.5.4-20. The applicant has stated that the Category I structures are not specifically designed for differential

Table 2.5-2 Settlements of Corners of Major Category I Structures

Structure	Settlement, inches Predicted	Settlement, inches Measured (1/1/84)	Percent Loading* (1/10/84)
Reactor Containment	1.7 (NE) 1.6 (SE) 1.7 (SW) 1.5 (NW)	1.38 (NE) 1.06 (SE) 1.06 (SW) 1.00 (NW)	95.3
Safeguards Area	1.6 (NE) 0.9 (SE) 1.2 (SW) 2.0 (NW)	1.12 (NE) 0.58 (SE) 0.60 (SW) 0.91 (NW)	97.8
Fuel Building	1.8 (NE) 1.8 (SE) 1.4 (SW) 0.8 (NW)	0.14 (NE) (not available) 0.14 (SW) 0.23 (NW)	94.7
Auxiliary Building	2.0 (NE) 1.6 (SE) 1.5 (SW) 0.8 (NW)	0.66 (NE) 0.58 (SE) (not available) 0.59 (NW)	97.3
Service Building	1.2 (NE) 1.1 (SE) 1.4 (SW) 1.4 (NW)	0.28 (NE) 0.24 (SE) 0.50 (SW) 0.78 (NW)	93.3
Diesel Generator Building	0.6 (NE) 0.4 (SE) 1.2 (SW) 1.1 (NW)	0.17 (NE) 0.14 (SE) (not available) 0.17 (NW)	92.8
Control Room Extension	1.0 (at the center of the building)	-0.10 (NE) -0.04 (SE) -0.20 (SW) -0.12 (NW)	91.5
Refueling Water Storage Tank	1.0 (NE) 0.6 (SE) 0.8 (SW) 1.0 (NW)	Not available (N/A) N/A N/A N/A	Not available (N/A) N/A N/A N/A

*NOTE: The applicant has indicated in his informal submittal that this percentage represents loading due to civil, steel and concrete components only. The measured settlements as of 1/1/84 are preliminary and will be confirmed by the applicant in the forthcoming amendment of the FSAR.

settlements; however, the applicant reviews the measured settlements periodically as part of a settlement monitoring program to detect any unusual movements of structures.

The applicant has analyzed the piping systems between adjacent structures for a minimum differential settlement of 0.5 inch. If this resulted in an overstress in pipes, the applicant analyzed the pipes for the predicted differential settlements, giving credit for the observed settlement that had occurred up to the time when piping connections were made between structures.

While the applicant is monitoring the settlements of all Category I structures as stated above, there is no program to monitor the settlement of buried pipelines. Since the pipelines have already been buried in the soil without installing the required instrumentation for settlement monitoring, the applicant may adopt an alternative procedure to demonstrate the safety of buried pipes against the effects of differential settlements. The applicant can make an analytical evaluation of the expected differential movements of buried pipes and determine if the pipes are capable of withstanding such differential movements without exceeding the allowable pipe stresses. Such an analytical evaluation and demonstration of the adequacy of the buried pipelines against the effects of differential settlements will be an acceptable alternative for settlement monitoring of buried pipes.

Longitudinal sections of pipelines (shown in FSAR Figs. 2.5.4-52 and 2.5.4-54 in response to the OL review question 241.2) indicate a considerable thickness of silty clay directly below compacted granular backfill. Also, as seen in FSAR Fig. 2.5.4-54, a steep gradient exists, in the embankment slope that contains the 30-inch pipelines that go from the present floodplain to the main plant area. The applicant must evaluate the differential settlement of these pipelines and include the results in the forthcoming amendment of the FSAR.

The applicant must include, in the forthcoming amendment of the FSAR, the following information.

1. Allowable total settlement and tilt of safety related structures.

2. Allowable differential settlements between structures, and those between structures and pipes buried in the soil. For the allowable settlement of buried pipelines, the applicant may provide an analytical evaluation of the ability of the pipes to withstand differential settlements without exceeding allowable pipe stresses.

3. A commitment to monitor the settlements of all Category I structures throughout the plant life.

2.5.4.3.4 Liquefaction Potential

The Category I structures at the main plant area are supported either on select compacted granular fill or medium dense to dense terrace sands and gravels; these materials are not susceptible to liquefaction. However, during additional boring exploration in the reactor containment excavation, a loose zone of potentially liquefiable granular material was discovered between approximately el 640 ft and 660 ft. This zone was effectively densified (as described in Section 2.5.4.1.2 above) to preclude liquefaction as demonstrated by the results of verification borings that indicated corrected SPT (N_1) values greater than 20.

The possibility, and the consequences, of liquefaction of the granular materials in the vicinity of the intake structure were thoroughly evaluated by the applicant (and reviewed by the NRC staff) at the construction permit stage as seen from the PSAR for BVPS-2, Amendment 13, dated February 28, 1974. Since liquefaction of these soils was considered likely, the applicant densified two areas west and east of the intake structure, each measuring 90' x 75', using the Terra Probe method. Areas immediately north of the intake structure and beneath the structure were not densified.

The effectiveness of the Terra Probe densification was evaluated by performing liquefaction analyses of the soils in the vicinity of intake structure using the data obtained by verification borings drilled in the densified areas. For analyzing the liquefaction potential of the soils beneath and north of the intake structure, borings drilled in the vicinity prior to densification (including the only preconstruction boring drilled beneath the intake structure) were used. The evaluation using the SSE indicated that the soils within the densified zones should not liquefy. The soils directly beneath the intake structure had a minimum factor of safety against liquefaction of 1.3 with the ground water level at el 665 ft (corresponding to normal river water level), and 1.1 with the ground water level at el 690 ft. The applicant has, thus, shown that the soils east and west of the intake structure, and beneath the structure, have some margin of safety against liquefaction for the combination of SSE and 25-year flood.

The applicant has also performed, but not yet docketed, a sliding stability analysis for the intake structure. In addition to this analysis, the applicant must also reevaluate and docket the liquefaction potential analysis of the soils beneath, and east, and west of the intake for the combination of OBE and a ground water level corresponding to the standard project flood (el 705 ft) as recommended by SRP 2.4.4.

The applicant has determined that the undensified area immediately north of the intake structure might liquefy under the SSE causing unanticipated stability problems. Therefore, the applicant has performed a static slope stability analyses for the dredged slopes (shown in FSAR Fig. 2.5.4-32) on the west and east sides of the intake structure, assuming that the liquefied soil north of the intake structure had weight but no shear strength. The results of the static slope stability analysis for both normal groundwater and 25-year flood conditions indicate that the dredged slopes are stable if the upper 10 ft of soil north of the intake structure liquefies. The applicant has also performed a dynamic slope stability analysis for the above side slopes, in response to an OL review question. Before docketing this analysis, the applicant must ensure that loading combinations include the OBE and Standard Project Flood and SSE and 25 year flood.

The areas immediately south of the intake structure and beneath and adjacent to the 30" service water supply (SWS) lines that run from intake structure to the main plant area were densified to the top of bedrock by vibroflotation as reported in the BVPS-2 PSAR Amendment 13 dated February 28, 1974. FSAR Fig. 2.5.4-16 shows the densified areas. Results of verification borings drilled in this area indicated that the relative density of the densified zones exceeded the minimum allowable value of 75% except for two out of 178 sand and gravel samples that showed less than 75 percent. The staff is reviewing the applicant's response to an OL review question regarding the adequacy of the width of densification of soils along the 30" SWS lines and will report the results in the final SER input.

2.5.4.3.5 <u>Seismic Loading</u>

Category I structures including buried piping have been designed for a SSE corresponding to a peak horizontal ground acceleration of 0.125g at the ground surface. The peak horizontal acceleration for the OBE was taken as 0.06g. Vertical accelerations were assumed to be two-thirds of the horizontal accelerations. Liquefaction potential evaluations were made assuming eight equivalent uniform stress cycles at these acceleration levels. Detailed discussion of the design earthquake is presented in Section 2.5.2 of this SER.

Soil structure interaction analyses of most of the Category I structures were performed using the lumped mass-spring method. Only the reactor containment and the fuel building were analyzed using the finite element analysis technique. The dynamic soil properties used in these analyses and in liquefaction analyses of soils were derived from the following sources:

1. In situ cross hole seismic wave velocity measurements for in situ soils.
2. Empirical relationship of Hardin and Drenevich (1972) for low strain shear modulus of compacted structural fill (Reference 3).
3. Variation of shear modulus and damping with strain levels was obtained from the published work of Seed and Idriss (1970). (Ref. 8).

As stated in Section 2.5.4.1.2, the engineering properties of the in situ sands and gravels were not determined by laboratory tests because of the inability to obtain undisturbed samples of these granular materials. In the soil structure interaction analysis of the Reactor Containment, the applicant varied the soil shear modulus value by ±30 percent, but has not so varied the damping value. The applicant has been asked to justify not varying the soil damping value by ±30% because of the uncertainty involved in the soil properties and the presence of thick clay lenses.

2.5.4.3.6 <u>Lateral Earth Pressures</u>

The static lateral earth pressure coefficients for the design of below-ground walls were obtained by accepted procedures (Reference 11). A value of 0.5 was used for the coefficient of lateral earth pressure at rest, K_0, for the in situ sands and gravel (Reference 11). The value of K_0 was chosen as 0.6 for compacted granular backfill to account for the increase in lateral pressure due to compaction. The design hydrostatic pressures acting on the structures is discussed in Section 2.4 of this SER.

The dynamic lateral earth pressures due to horizontal and vertical ground accelerations were determined using the procedures developed by Monabe-Okabe and described by Seed and Whitman (1970). (Ref. 9).

2.5.4.4 Instrumentation

Since the structures at the site are founded on soils and are likely to undergo settlement for a long period of time, the applicant plans to continue the settlement monitoring program that was started during construction. The instrumentation consists of permanent benchmarks and piezometers installed at several locations of the site, and settlement markers in each structure. During construction the settlements are measured by the applicant monthly, and compared with predicted settlements. After the completion of construction and when the settlement data indicate no significant additional settlements occurring, the frequency of measurements will be reduced. Details of the results of settlement monitoring are discussed in Section 2.5.4.3.3 above.

2.5.4.5 Conclusion

The staff has performed a review of the applicant's design criteria and the results of the applicant's field investigations, laboratory tests, and engineering analyses in accordance with the procedures established in SRP Section 2.5.4 and concluded that the plant foundations will safely support the seismic Category I structures and systems. This conclusion is subject to the applicant's furnishing of certain additional data and analyses described in the above subsections. The major items that need to be addressed by the applicant in the forthcoming amendment of the FSAR are the following:

1. Furnish longitudinal sections of Category I pipelines and ducts not already provided showing therein the soil profile and the elevations at which the pipes are laid. Locations of manholes and their foundation configuration should also be shown in these longitudinal sections;

2. Provide the actual values of the geotechnical parameters such as subgrade modulus, shear wave velocity and soil modulus, etc. used in the static and dynamic analysis of buried pipes;

3. Provide a table comparing the latest measured settlements with predicted and allowable total and differential settlements between structures and tilt of these structures; also provide an analytical evaluation to demonstrate the adequacy of buried pipes to withstand the effects of expected differential settlements.

4. Commit to monitor the settlement of Category I structures throughout plant life.

5. Justify not varying the soil damping value by ±30% in the soil-structure interaction analysis of Reactor Building while varying the shear modulus value by ±30%.

6. Docket the revised FSAR Table 2.5.4-4, including therein the corrected dynamic soil pressures and factors of safety against bearing capacity failure and also incorporating the data concerning the foundation for the main intake structure; also furnish the sliding stability analysis of the intake structure for all applicable loading conditions.

7. Revise the FSAR ~~that~~ Section 2.5.4.7.3 discuss~~es~~ the ~~anamoly~~ anomaly in the reported values of the groundwater level and the soil shear wave velocity below groundwater levels in FSAR Fig. 2.5.4-17.

2.5.5 Stability of Slopes

The applicant has analyzed the stability of slopes for the following three areas:

1. the riverward slope that supports the 30" service water pipelines leading to the intake structure;
2. the dredged side slopes east and west of the intake channel located in front of the intake structure; and
3. the slopes at the Emergency Outfall Structure (EOS).

2.5.5.1 Slopes Near Intake Structure

The static and dynamic stability of the riverward slope were analyzed by two methods: the simplified Bishop Method (using a circular arc failure surface) and the Morganstern-Price Method (that allows for an arbitrary shaped failure surface). These stability analyses were performed using the computer program LEASE II that employs a pseudo-static approach for dynamic stability analysis. The horizontal seismic coefficient is taken as 0.125 for the SSE. The vertical seismic coefficient is taken as 0.083. In response to the OL review question 241.18, the applicant has considered additional failure surfaces through the silty clay layer as shown in FSAR Fig. 2.5.4-57. This figure also shows the soil properties of various layers, used in the analysis. The ground water table is taken at el 705 corresponding to the standard project flood. The minimum safety factor of 1.29 was obtained in this dynamic slope stability analysis. The applicant will docket the results of this analysis in the forthcoming amendment of the FSAR. We will report our evaluation in the Final SER.

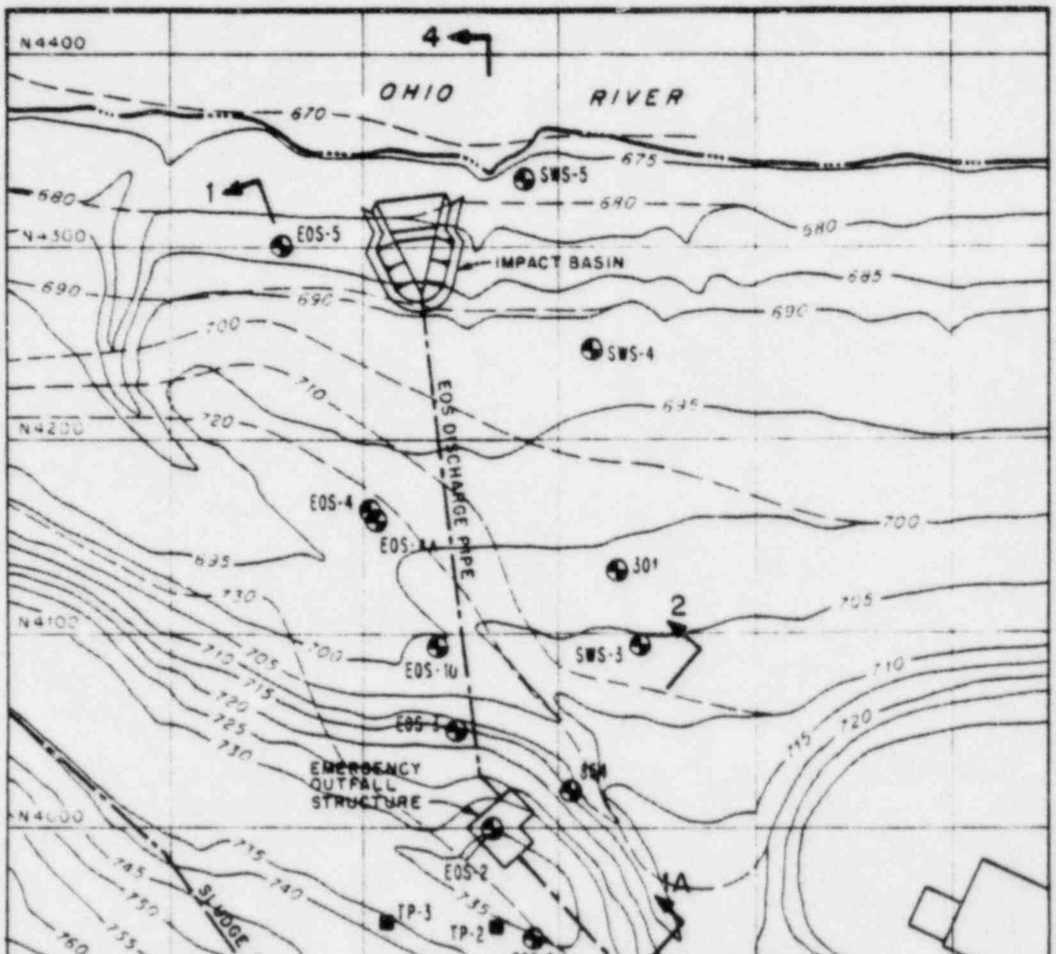
2.5.5.2 Dredged Slopes of the Intake Channel

The stability analysis of the side slopes east and west of the intake channel in front of the intake structure has been discussed in Section 2.5.4.3.4 above. The staff will review the dynamic stability analysis of these slopes when furnished by the applicant with the forthcoming amendment of the FSAR.

2.5.5.3 Slopes Near the EOS

The Emergency Outfall Structure, (EOS), constructed at the far western part of the site, provides missile protection for the emergency discharge point of the service water system, and raises the discharge point above the PMF level (el 730 ft). As shown in Fig. 2.5-3, two sets of slopes exist in the vicinity of the EOS: the steep valley wall about 150 ft to the southwest of the EOS, called the colluvial slope, and the terrace directly northeast of the EOS, called the riverward slope.

Subsurface profiles for these slopes were developed using 11 borings (EOS series) drilled by the applicant's consultant and some borings (PL-series) performed by others in this general area. Four piezometers were installed in the soils under the colluvial slope. The groundwater levels followed quite closely the levels of the Ohio River. Because of the complex character of the soil deposits in this area, it is difficult to develop a soil profile showing specific continuous soil types between adjacent borings. Generally, on the steep valley walls, the bedrock surface is overlain by coarse colluvium (sandy gravel) derived from the weathering of the parent sandstone bedrock at higher elevations. The coarse colluvium is, in turn, overlain by fine colluvium (sandy clay) derived from the weathering of shales, claystones, and limestones. Bedrock strata in the site area are essentially flatlying with a slight regional dip (less than 5°) to the southeast. At the base of the valley and extending north to the river is seen an interfingering of the colluvial slopes with the glacial outwash and alluvial soils deposited by the Ohio River. The terrace north of boring EOS-10 has been eroded



LEGEND

- BORINGS BY OTHERS
- BORINGS BY SWEC
- TEST PITS
- GROUND SURFACE CONTOURS-1972
- GROUND SURFACE CONTOURS-1982

Figure 2.5-3
Boring Location Plan
Emergency Outfall Structure
(FSAR Fig. 1-1 in Appendix 2.5E)

and more recent river deposits of silt and clay have replaced portions of the original granular outwash deposits.

Properties of the fine grained soils were determined by laboratory tests while those of the coarse grained soils were estimated from correlations with SPT blow counts and soil sample descriptions. The following tabulations gives the properties of different soil types used in the static and dynamic slope stability analyses. The applicant must justify the use of effective strength parameters (i.e. effective friction angle, $\bar{\phi}$, and effective cohesion, $\bar{c}$) in the dynamic stability analysis where permanent slope displacements are anticipated.

COLLUVIAL SLOPE

Soil Type	Total Unit Weight (pcf)	Friction Angle $\bar{\phi}$, degrees	Cohesion $\bar{c}$, psf
1. Fine colluvial soils	125	28	0
2. Coarse colluvial soils	135	40	0
3. Fine alluvial soils	120	30	0
4. Coarse alluvial soils	130	30	0

RIVERWARD SLOPE

Soil Type	Total Unit weight (pcf)	Friction Angle $\bar{\phi}$, degrees	Cohesion $\bar{c}$, psf
1. Fill (compacted granular)	136	36	0
2. Fill (uncontrolled)	120	30	0
3. Sand and Sandy Gravel	125	30	0
4. Silty clay	125	32*	0

*For the dynamic case, the silty clay friction angle was assumed zero, and cohesion values ranging from 200 psf to 2500 psf were used to represent undrained loading conditions.

The static stability analyses of the colluvial slope south of the EOS indicated safety factors greater than 1.5. However, the applicant has reported that minor surface sloughing of the upper slope (above el 780 ft) was possible since shallow circular surfaces of failure gave safety factors of 1.0.

The dynamic stability analysis of the colluvial slope indicated a minimum factor of safety of 0.8 for a circular arc within the fine grained colluvium. Therefore, the applicant assumed that some movement of the slope would occur in the event of the SSE, and estimated the plastic (permanent) displacement of the slope using a computer program SIDES based on the Newmark model (Reference 6). The acceleration time histories from the El Centro 1940 earthquake (N-S component) and the 1952 Kern County earthquake (S69E component of the Taft record) were used in this analysis. The cumulative displacement of the slope predicted by the SIDES program (Reference 10) was less than an inch. Although the magnitude of this movement is small the applicant has been requested to document the accuracy of the analysis by furnishing an independent verification of the SIDES program since it is a SWEC in-house program.

The applicant analyzed the static and dynamic slope stability of the riverward slope north of the EOS for the combined loading of SSE and the normal water level at el 665 ft. The safety factors in the static and dynamic cases were 1.6 and 1.2 respectively. The SRP, Section 2.4.4, recommends that analyses be made for two combined loading conditions, namely, SSE + 25 yr. flood (el 690 ft), and OBE + standard project flood (el 705 ft). The applicant has stated that groundwater levels in the clay layer of the riverward slope would not change substantially during the relatively short duration of the 25-yr flood. Therefore, the applicant has assumed that it is acceptable to consider failure surfaces through the clay layer with the groundwater level corresponding to the normal river water level at el 665 ft. rather than 690 ft. Because of the presence of cohesionless soil layers with greater permeability than that of clay layer in the riverward slope, the staff requires that the applicant perform stability analyses for the two loading conditions described above. We will report the results of our evaluation of this matter in the final SER.

2.5.5.4 Conclusion

Based on a review of the applicant's design criteria and the results of his analyses, the staff has concluded that the slopes at the site are generally stable for the loading conditions considered by the applicant. However, the applicant must reevaluate the stability of each of these slopes for two loading conditions, namely, (1) SSE + 25 yr flood and (2) OBE + standard project flood, as recommended by SRP Section 2.4.4. The applicant must also docket the stability analyses of all slopes where revised seismic coefficients have been used. The applicant must justify the use of effective strength parameters in the dynamic slope stability analysis. The applicant must also document an independent verification of the accuracy of the permanent displacement analysis results obtained by using the SWEC inhouse computer program, SIDES.

2.5.6 Embankments and Dams

There are no category I embankments or dams at the site that might affect the safety of the Category I structures.

References

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8 ELECTRIC POWER SYSTEMS

8.1 General

The bases for the staff's evaluation of the applicant's designs, design criteria, and design bases for the Beaver Valley electric power systems are set forth in the Standard Review Plan (SRP) (NUREG-0800) Section 8.1, Table 8-1, "Acceptance Criteria and Guidelines for Electric Power Systems." These acceptance criteria and guidelines include the applicable general design criteria (Appendix A to 10 CFR 50) and guidelines of branch technical position, regulatory guides, and NUREGs. The staff has determined that conformance to the applicable general design criteria and guidelines cited above would provide~~s~~ sufficient bases for acceptance of the electric power systems.

The following subsections provide the staff's evaluation of the offsite and onsite electric power system design and ~~how~~ whether it meets the requirements of the above cited acceptance criteria. As a part of the review process, the staff will also visit the site to view the installation and arrangement of electrical equipment and cables, to review confirmatory electric drawings, and to verify test results for the purpose of verifying the adequacy of the design and proper implementation of the design criteria. The confirmatory site visit will be completed prior to issuance of the license and if any problems are found, they will be addressed in a supplement to this report.

The conclusions in the following subsections are subject to acceptable implementation of design changes that, if any, may be required as a result of the staff's site visit.

8.2 Offsite ~~Electric~~ Power System

The safety function of the offsite power system (assuming the onsite power system is not functioning), is to provide sufficient capacity and capability to assure that the structures, systems, and components important to safety perform as intended. The objective of the staff review is to determine that the offsite power system satisfies the requirements of General Design Criteria 5, 17, and 18 and will perform its design function during all plant operating and accident conditions.

8.2.1 Compliance With General Design Criterion (GDC) 5

The applicant has met (except as noted) the requirements of GDC 5, "Sharing of Structures, Systems, and Components," with respect to sharing of circuits of the preferred power system.

The following items address problem areas revealed during the staff review and resolutions or status concerning them.

8.2.2 Compliance With General Design Criterion (GDC 17)

The applicant has met (except as noted) the requirements of GDC 17, "Electric Power Systems," with respect to the offsite power system's (a) capacity and capability to permit functioning of structures, systems, and components important to safety, (b) provisions to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit or loss of power from the onsite electric power supplies, (c) independence of circuits, and (d) availability of circuits.

The following items address the problem areas revealed during the staff review and resolutions or status concerning them.

8.2.2.1 Physical Independence of Offsite Circuits Between the Grid System and Switchyard

The original description and analysis relating to physical independence of the offsite power system's transmission lines between the Duquesne transmission grid system and the Beaver Valley switchyard, contained in Section 8.2.1.1 of the FSAR, ~~is~~ was limited to the following: The transmission lines converge on the switchyard by means of two or more widely separated routes. This description ~~is~~ was not sufficient for the staff to conclude that the transmission lines are adequately separated in accordance with the requirements of Criterion 17 of Appendix A to 10 CFR 50.

By amendment 3 to the FSAR, the applicant provided additional description with layout drawings of the subject physical separation of offsite transmission lines. Based on the additional description the staff concludes that the offsite transmission lines have adequate physical separation in accordance with the requirements of GDC 17 and are, therefore, acceptable.

8.2.2.2 Capability to Reestablish Power From the Offsite Power System

GDC 17 requires, in part, that each of the offsite circuits be designed to be available in sufficient time following a loss of all onsite alternating current power supplies and the other offsite electric power circuit, to assure that specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded. The description in the FSAR as to compliance with this part of GDC 17 is not sufficient to reach a conclusion of acceptability.

By amendment 3 to the FSAR the applicant did not provide the requested description. This item will be pursued with the applicant and the requests of the staff review will be reported in a supplement to this report.

8.2.2.3 Independence of Offsite Power Circuits Between the Switchyard and Class 1E System

The Beaver Valley design provides two immediate access offsite circuits between the switchyard and the 4.16 KV Class 1E busses. It is the staff position that these two circuits be physically separate and independent such that no single event can simultaneously affect both circuits in such a way that neither can be returned to service in time to prevent fuel design limits or design conditions of the reactor coolant pressure boundary from being exceeded. The physical separation and independence of these two circuits ~~has~~ had not been described or analyzed in the FSAR.

By amendment 3 to the FSAR, the applicant, in response to a request for information, has provided a description of the routing or physical separation and independence of these two circuits. Based on the description, the staff concludes that these circuits are physically separated, meet the above staff position, the requirements of GDC 17, and are acceptable.

In regard to physical separation and independence of controls and protective relaying associated with these circuits, the applicant, in response to a request for information, addressed controls and relaying for 138 KV circuit breakers, the station service transformers, and the 5 KV cable bus. Control and relaying for 5KV circuit breakers and Busses 2A, 2B, 2C, and 2D were not addressed in the applicant's response. This item will be pursued with the applicant and the results of the staff review will be reported in a supplement to their report.

The description of physical separation of offsite circuits has not been included in Section 8.2 of the FSAR in accordance with the guidelines of Regulatory Guide 1.70, Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants. This item will be pursued with the applicant and the results of the staff review will be reported in a supplement to this report.

8.2.2.4 Independence Between Onsite and Offsite Power Sources

Each of the 4.16 KV Class 1E buses at Beaver Valley is supplied power from preferred offsite and standby onsite circuits. It is the staff position that these circuits should not have common failure modes. Physical separation and independence of these circuits has not been described or analysed in the FSAR.

The applicant by amendment 3 to the FSAR did not provide a description or analysis that was requested. This item will be pursued with the applicant and the results of the staff review will be reported in a supplement to this report.

8.2.2.5 Use of Automatic Load Tap Changer

Section 8.3.1.1.1 of the FSAR indicates that the system station service transformer specified with an automatic load tap changer.

By amendment 3 to the FSAR, the applicant, in response to a request for information, indicated that the automatic load tap changer optimize voltage on the 4160 volt Class 1E buses for any plant load condition and power grid voltage variation. The applicant has further implied that the design is Class 1E and meets all the requirements of a Class 1E system. Design criteria with description and analysis as to the systems compliance with GDC 2, 4, 5, 17, and 18 has not been addressed in the FSAR. This item will be pursued with the applicant and the results of the staff review will be reported in a supplement to this report.

8.2.3 Compliance With General Design Criterion (GDC) 18

The applicant has met (except as noted) the requirements of GDC 18, "Inspection and Testing of Electric Power Systems," with respect to the capability to test systems and associated components during normal plant operation and the capability to test the transfer of power from the nuclear power unit, the offsite preferred power system, and the onsite power system. The following items address problem areas revealed during the staff review and resolutions or status concerning them.

8.2.3.1 Capability to Test Transfer of Power Between Normal and Preferred Offsite Circuits

The capability to test the transfer of power from the normal unit station service transformer to the station service transformer has not been specifically addressed in the FSAR.

By amendment 3 to the FSAR, the applicant, in response to a request for information, described the transfer circuitry, how it is tested during normal plant operation, and its compliance with GDC 18. Based on the description the staff concludes that the design is testable, meets the requirements of GDC 18, and is acceptable. It is the staff's concern, however, that periodic testing of the transfer may create transients in the plant if done during power operation. This concern will be pursued with the applicant and the results of the staff review will be reported in a supplement to this report.

The above description has not been included in Section 8.2 of the FSAR in accordance with the guidelines of Regulatory Guide 1.70, Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants. This item will be pursued with the applicant and the results of the staff review will be reported in a supplement to this report.

8.2.4 Evaluation Findings

The review of the offsite power system for the Beaver Valley Plant covered single line diagrams, station layout drawings, schematic diagrams, and descriptive information. The basis for acceptance of the offsite power system in the staff review was conformance of the design criteria and bases to the Commission's regulations as set forth in the General Design Criteria (GDC) of Appendix A to 10 CFR 50. The staff concludes that the plant design meets the requirements of GDC 5, 17, and 18, and conforms to applicable guidelines of regulatory guides and branch technical positions and is acceptable except as noted in the preceding sections.

8.3 Onsite Power Systems

The safety function of the onsite power system (assuming the offsite power system is not functioning) is to provide sufficient capacity and capability to assure that the structures, systems, and components important to safety perform as intended. The objective of the review is to determine that the onsite power system satisfies the requirements of GDC 2, 4, 5, 17, 18, and 50 and will perform its intended function during all plant operating and accident conditions.

The onsite power system consists of an alternating current (ac) power system and a direct current (dc) power system. Compliance with GDC 2, 4, 5, 18, and 50 as they relate to both ac and dc systems are evaluated in Section 8.3.3 of this report. Compliance with GDC 17 as it relates to ac systems is evaluated in Section 8.3.1 of this report and as it relates to dc systems is evaluated in Section 8.3.2 of this report.

8.3.1 Onsite AC Power System's Compliance With General Design Criterion (GDC) 17

The applicant has met (except as noted) the requirements of GDC 17 "Electric Power Systems," with respect to the onsite ac system's (a) capacity and capability to permit functioning of structures, systems, and components important to safety, (b) the independence, redundancy, and testability to perform their safety function assuming a single failure, and (c) provisions to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit or the loss of power from the transmission network.

The following items address the problem areas revealed during the staff review and resolution or status concerning them.

8.3.1.1 Voltage Analysis

"The voltage levels at the safety-related loads should be optimized for the maximum and minimum load conditions that are expected throughout the anticipated range of voltage variations of the offsite power sources. The applicant was requested to perform a voltage analysis and verification by actual measurement in accordance with the guidelines of positions 3 and 4 of branch technical position PSB-1 (NUREG-0800, Appendix 8A).

By amendment 3 to the FSAR, the applicant indicated that the requested analysis would not be completed before March 15, 1984. Review schedule for submittal of the analysis, verification of the analysis by actual measurement, and justification for voltages (as determined by analysis) not meeting the specific voltage supply tolerances specified by equipment manufacturers, will be pursued with the applicant and the results of the staff review will be reported in a supplement to this report.

8.3.1.2 Bypass of Diesel Generator Protective Trips

Sections 8.3.1.1.15 of the FSAR indicates that a number of tripping devices have been provided for each diesel generator. The majority of these tripping devices are bypassed when the diesel generator receives an emergency start signal. Tripping devices that are not bypassed include generator current differential, generator overexcitation, and engine overspeed protection. This design meets the guidelines of position 7 of Regulatory Guide 1.9 except for the generator overexcitation tripping device that is not bypassed.

By amendment 3 to the FSAR, the applicant, in response to a request for information, indicated that the design for generator overexcitation trip has two independent measurements with coincident logic for trip actuation. This design also meets the guidelines of position 7 of Regulatory Guide 1.9 and therefore is acceptable. Surveillance requirements for the protective trips that are bypassed will be included in the technical specifications. The design for the protective bypass will be confirmed as part of the staff drawing review/site visit.

8.3.1.3 Load Testing of the Diesel Generator

Section 8.3.1.1.16 of the FSAR indicates that safety related motors are designed with the capability of accelerating the driven equipment to its rated speed with 80 percent of motor nameplate voltage applied at the motor terminals. Section 8.3.1.1.15 of the FSAR indicated that the design of each diesel generator unit is such that at no time during the loading sequence does the voltage decrease to less than 75 percent of nominal.

By amendment 3 to the FSAR, the applicant, in response to a request for additional information, indicated that data extrapolated from diesel generator load tests implied that 79.3 percent versus 75 percent is the largest voltage drop to be expected during the diesel generator load sequence. Testing of the diesel generator using actual load and loading sequence to demonstrate the voltage will not drop below 80 percent will be pursued with the applicant and the results of the staff review will be reported in a supplement to this report.

8.3.1.4 Compliance to BTP-PSB-2

Section 8.3.1.1.15 of the FSAR describes the surveillance instrumentation provided to monitor the status of the diesel generator. In this regard, the applicant was requested to describe how the Beaver Valley design complies with the guidelines of Branch Technical Position PSB-2. By amendment 3 to the FSAR the applicant did not provide the requested description. This item will be pursued with the applicant and the results of the staff review will be reported in a supplement to this report.

8.3.1.5 Capability of Diesel to Accept Design Load After Prolonged No Load Operation

Section 6.4.2 of IEEE Standard 6.4.2 of IEEE Standard 387-1977 requires, in part, that the load acceptance test consider the potential effects on load acceptance after prolonged no load or light load operation of the diesel generator. The applicant was requested to provide the results of load acceptance tests or analysis that demonstrates the capability of the diesel generator to accept the design accident load sequence after prolonged no load operation.

By amendment 3 to the FSAR the applicant did not provide the requested test or analysis results. This item will continue to be pursued with the applicant and the results will be reported in a supplement to this report.

8.3.1.6 Diesel Generator Loading Above its Continuous Rating

Section 8.3.1.1.15 of the FSAR states that the maximum load imposed on the diesel generator is less than the continuous rating. The continuous rating has been defined to be 4238 KW. In contradiction, Table 8.3-3 of the FSAR states that the worst case loading is 4261 KW. 4261 is greater than the stated maximum load of 4238 KW imposed. Justification for this contradiction will be pursued with the applicant and the results of the staff review will be reported in a supplement to this report.

8.3.1.7 Compliance With IEEE Standard 387-1977

Table 1.8-1 of the FSAR indicates that the diesel generators have been selected, designed, and qualified following the guidance of IEEE Standard 387-1977 as augmented by Regulatory Guide 1.9 with the exception that the diesel generators were procured with the specification that they comply with the 1972 version of IEEE Standard 387. By amendment 3 to the FSAR, the applicant, in response to a request for information, stated that the diesel generators are in conformance with IEEE Standard 387-1977 and Regulatory Guides 1.9 and 1.108. Based on this statement of compliance, the staff concludes that even though the diesel generators may have been procured to 1972 guidelines, they have been designed, tested, and qualified to 1977 guidelines, and are, therefore, acceptable.

8.3.1.3 Diesel Generator Start and Load Acceptance Qualification Tests

Section 6.3.2 of IEEE Standard 387-1977 requires that a series of tests be conducted to establish the capability of the diesel generator unit to start and accept load within the period of time to satisfy the plant designs requirements. By amendment 3 to the FSAR the applicant documented that the diesel generator voltage and frequency were monitored, recorded, and verified when starting the unit and applying a 50 percent load for each of the 300 start-load tests in full compliance with IEEE Standard 387-1972. In regard to this item the following items will continue to be pursued with the applicant:

1. Testing to the 1972 versus 1977 versions of IEEE Standard 387,

2. Definition for specified frequency, voltage, and required time interval, and

3. Conformation of test results.

The results of the staff review of these items will be reported in a supplement to this report.

8.3.1.9 Diesel Generator Load Capability Qualification Test

Section 6.3.1 of IEEE Standard 387-1977 requires that one test be conducted to demonstrate the capability of the diesel generator to carry and reject rated loads. By amendment 3 to the FSAR, the applicant, in response to a request for information, indicated that these tests were not performed by the manufacturer but will be performed after installation of the diesel generators at the plant site. Confirmation of these test results will be pursued with the applicant and the results of the staff review will be reported in a supplement to this report.

8.3.1.10 Margin Qualification Test

Section 6.3.3 of IEEE Standard 387-1977 requires at least two margin tests to demonstrate diesel generator capability to start and carry loads that are greater than the most severe step load change within the plant design loading sequence. By amendment 3 to the FSAR, the applicant did not provide the requested description as to how the Beaver Valley testing meets the margin test requirements of Section 6.3.3 of IEEE Standard 387-1977. This item will be pursued with the applicant and the results of the staff review will be reported in a supplement to this report.

8.3.1.11 Description of Compliance With IEEE Standard 387-1977

A description as to how the Beaver Valley design complies with the guidelines of IEEE Standard 387-1977 as augmented by Regulatory Guide

1.9 and 1.108 has not been presented in the FSAR nor was the description provided in amendment 3 to the FSAR as requested. This item will be pursued with the applicant and the results of the staff review will be reported in a supplement to this report.

8.3.1.12 Design for Power Removal for Selected Safety Related Valves

Table 8.3-5 of the FSAR identifies valves from which power is to be removed in order to meet the single failure criterion. By amendment 3 to the FSAR, the applicant indicated that removal of a banana plug located in the control room provides the necessary power removal and will prevent inadvertent operation of the valves. Details of the design for power removal will be pursued with the applicant and the results of the staff review will be reported in a supplement to this report.

8.3.1.13 Electrical Interconnections Between Redundant Class 1E Buses

Section 8.3.1.1.4 of amendment 3 to the FSAR identifies a number of Class 1E loads that can be electrically connected to both redundant Class 1E power supplies. To prevent the electrical interconnection of redundant Class 1E power supplies, a key-interlocked manual transfer switch design is provided. Based on the description presented in the FSAR, the staff concludes that the design provides reasonable assurance that suffficient independence will be maintained between redundant electrical systems, meets GDC 17 and is, therefore, acceptable. The design will also be reviewed as part of the staff's confirmatory site visit. If problem areas are identified they will be reported in a supplement to this report.

8.3.1.14 Automatic Reclosure of 4160 Volt Circuit Breakers After Manual Trip

Section 8.3.1.1.3 of the FSAR implies that when a Class 1E 4160 volt circuit breaker is tripped manually while a safety injection signal is present, the breaker control scheme is such that automatic reclosure will occur.

In order to understand how this automatic reclosure design may affect operation of other safety systems, the following items will be pursued with the applicant: (a) details of the design for automatic reclosure, (b) the extent and purpose of the design, (c) justification for bypass of anti-pump design feature and (d) design provisions to preclude automatic reclosure during diesel generator operation or analysis which demonstrates that overload of diesel generator will not occur. The results of the staff review will be reported in a supplement to this report.

8.3.1.15 Design Provisions for the Use of Replacements for Class 1E Loads

Section 8.3.1.1.4 and Table 8.3-3 of the FSAR indicates that for a number of Class 1E loads, there is a replacement load provided to allow maintenance to be performed while satisfying the single failure criterion. The Beaver Valley design is such that the Class 1E load and its replacement may be connected to the same Class 1E power supply at the same time. It is the staff concern that this simultaneous connection of loads will exceed the capacity of the Class 1E power supplies. Identification of loads involved and design provisions to preclude simultaneous connection will be pursued with the applicant and the results of the staff review will be reported in a supplement to this report.

8.3.1.16 Connected Accident Loading Exceeds Capacity of the Diesel Generator

Section 8.3.1.1.7 of the FSAR states that the diesel generator units are designed and manufactured so that the capacity of each diesel generator unit is sufficient to start and accelerate all connected loads to their rated condition in the specified time sequence. Based on the connected loading presented in Table 8.3-3 of the FSAR and the diesel generator rating presented in Section 8.3.1.1.15 of the FSAR, it appears that the connected loading exceeds the rated capacity of 4238 KW. A detailed analysis of the loading and design provisions provided to preclude having the load exceed 4238 KW will be pursued with the applicant and the results of the staff review will be reported in a supplement to this report.

8.3.17 Design for Connecting Non Class 1E Standby Service Water Pumps on the Class 1E System

Section 8.3.1.1.8 of the FSAR indicates the following in regard to the non Class 1E standby service water pumps when there is a safety injection signal:

a. Non Class 1E loads are stripped and blocked from starting with the possible exception of the standby service water pump motors. If these motors are running, they will not be tripped.

b. During the automatic loading sequence of safety loads, the standby service water pumps will be blocked from starting until the automatic loading sequence is complete.

Clarification of the design for the loading of the non Class 1E standby service water pumps onto the Class 1E power supplies and its purpose will be pursued with the applicant and the results of the staff review will be reported in a supplement to this report.

By Table 8.3-2 of the FSAR, the design for the non-safety alternate intake structure exhaust fan load appears to be the same or similar as that of the standby service water pump load. Clarification for the loading of this non Class 1E load onto the Class 1E system and its purpose will also be pursued with the applicant and the results of the staff review will be reported in a supplement to this report.

8.3.1.18 Loading of RHR Pump onto the Diesel Generator

Table 8.3-3 of the FSAR indicates that the RHR pumps are not needed for DBA mode of operation and are not needed for four hours after a loss of offsite power or after loss of offsite power with a safety injection signal. Specific reference to RHR system description in the FSAR and justification for this power availability to RHR pumps will be pursued with the applicant and coordinated with the Reactor Systems Branch. The results of the staff review will be reported in a supplement to this report.

8.3.1.19 Automatic Reconnection of Non-Safety Loads After Loss of Offsite Power

Table 8.3-3 of the FSAR indicates that the non Class 1E pressurizer heater backup load is automatically reconnected to the Class 1E system after a loss of offsite power. The staff has been accepting design wherein non- Class 1E loads were reconnected manually after loss of offsite power as well as after an accident signal. Justification for non-compliance with the accepted practice will be pursued with the applicant and the results of the staff review will be reported in a supplement to this report.

8.3.1.20 Physical Independence

Physical independence criteria for the redundant onsite ac power system is the same as that for the onsite dc system and is, thus, addressed in Section 8.3.3 of this report.

8.3.2 Onsite DC System's Compliance With General Design Criterion (GDC) 17

The applicant has met (except as noted) the requirements of GDC 17, "Electrical Power Systems," (a) capacity and capability to permit functioning of structures, systems, and components to safety, (b) the independence, redundancy, and testability to perform their safety function assuming a single failure, and (c) provisions to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit or the loss of power from the transmission network.

The following items address the problem areas revealed during the staff review and resolution or status concerning them.

8.3.2.1 Physical Independence

Physical independence criteria for the redundant onsite dc power system is the same as that for the onsite ac system and is, thus, addressed in Section 8.3.3.3 of this report.

8.3.3 Common Electrical Features and Requirements

This section presents common electrical features and requirements of the onsite ac and dc power system which deal with distinct aspects of the onsite alternating current and direct current power systems. The common electrical features and requirements addressed in this section are as follows:

8.3.3.1 Compliance With General Design Criteria (GDC) 2 and 4

The applicant has met (except as noted) the requirements of GDC 2, "Design Basis for Protection Against Natural Phenomena," and GDC 4, "Environmental and Missile Design Bases," with respect to structures, systems, and components of the onsite ac and dc power system being capable of withstanding the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, and floods, missiles, and environmental conditions associated with normal operation and postulated accidents. The onsite power system and components (1) are located in seismic Category I structures which provides protection from the effects of tornadoes, tornado missiles, turbine missiles, and external floods, (2) have been given a quality assurance designation "Class 1E," (3) have been designated to be seismically and environmentally qualified, and (4) are to be designed to acccommodate or are to be protected from the effects of missiles and environmental conditions associated with normal operation and postulated accidents.

The following items address the problem areas revealed during the staff review and the resolution or status concerning them.

8.3.3.1.1 Submerged Electrical Equipment as a Result of a Loss-of-Coolant Accident

It is the staff's concern that following a loss-of-coolant accident, fluid (from the reactor coolant system and from operation of the emergency core cooling systems) may collect in the primary containment and reach a level that may cause certain electrical equipment located inside the containment to become submerged and thereby rendered inoperable. Both safety and nonsafety-related electrical equipment is of concern, because their failure may cause electrical faults that could compromise the operability of redundant emergency power sources or the integrity of containment electrical penetrations. In addition, the safety-related electrical equipment that may be submerged is also of concern if this equipment is required to mitigate the consequences of the accident for both the short-term and long-term emergency core cooling system functions and for containment isolation.

The staff's position, in regard to submerged equipment, is that all electrical equipment must be located above the maximum possible flood level or be qualified for submerged operation, or the lack of qualification must be justified.

By amendment 3 to the FSAR, the applicant provided a listing of safety class equipment that may become submerged as a result of a LOCA and are not designed and qualified for submergence. In justification of the lack of qualification, the applicant stated that the design of the Class 1E distribution system satisfies the isolation criteria by ensuring that the failure of the submerged equipment will not degrade the Class 1E power source. Clarification of the isolation criteria and how it ensures that Class 1E systems will not be degraded will be pursued with the applicant and the results of the staff review will be reported in a supplement to this report.

8.3.3.1.2 Design, Qualification and/or Protection of Class 1E Equipment From Natural Phenomena

Sections 8.3.1.2 and 8.3.2.2 of the FSAR states, in regard to compliance with General Design Criterion (GDC) 2 of Appendix A to 10 CFR 50, that Class 1E ac and dc systems are housed in structures that are designed to, and are capable of, withstanding the effects of natural phenomena such as earthquakes, tornados, hurricanes, and floods without loss of capability to perform its function.

Based on this statement of compliance, the staff is unable to conclude that all instrumentation, control, and electrical structures, systems, and components important to safety have been either designed and qualified to operate in an environment caused by natural phenomena or have been adequately protected from its effects.

By amendment 3 to the FSAR, the applicant did not provide the requested information for an expanded analysis of compliance with GDC 2. This item will continue to be pursued with the applicant and the results of the staff review will be included in a supplement to this report.

8.3.3.1.3 Protection of Class 1E Equipment From Dynamic Effects

In Section 8.3.1.2 and 8.3.2.2 of the FSAR, it has been stated, in regard to compliance with General Design Criterion (GDC) 4 of Appendix A to 10 CFR 50, that Class 1E ac and dc power systems are designed to accommodate the effects of the environmental conditions associated with normal operation and postulated accidents and that the structures, the ac and dc systems are housed in, are protected against internally-and externally-generated missiles, pipe whip, and jet impingement forces associated with pipe breaks such that safety functions will not be precluded. Based on this statement of compliance, the staff is unable to conclude that all instrumentation, control, and electrical structures, systems, and components important to safety have been appropriately protected against dynamic effects in accordance with the requirements of GDC 4.

By amendment 3 to the FSAR, the applicant did not provide the requested information for an expanded analysis of compliance with GDC 4. This item will be pursued with the applicant and the results of the staff review will be reported in a supplement to this report.

8.3.3.1.4 Protection of Class 1E Equipment From Fire Protection System Effluents

Section 8.3.1.4 of the FSAR indicates that fire suppression systems are installed in a number of plant areas at Beaver Valley that contain Class 1E systems and components. For the design basis event "fire protection system operation," it is the staff position that Class 1E systems and components located in areas with fire suppression systems should be capable and qualified to perform their function when subject to the effects of the subject design basis event (Section 4.2 and 4.7 of IEEE Standard 308-1974).

By amendment 3 to the FSAR, the applicant (in response to a request for information) provided a positive statement of compliance to the above stated position. Pending documentation in Section 8 of the FSAR, this item is considered resolved.

8.3.3.1.5 Bypass of Thermal Overload Protection

Section 8.3.1.1.11.2 of the FSAR, indicates that thermal overload protection is provided for continuous and intermittent duty motors.

By amendment 3 to the FSAR, the applicant, in response to a request for information, provided a description of their thermal overload protection bypass design for all motor operated valves that are required for safe shutdown. Accident signal contacts in parallel with the thermal overload relay contacts, provide the required design for bypass. The design meets the guidelines of position 1 of Regulatory Guide 1.106, meets the requirements of GDC 4, and is acceptable.

8.3.3.1.5 Design and Qualification of Safety Related Electric Equipment

Section 8.3.1.2.1 of the FSAR states that qualification of Class 1E equipment is addressed in Section 3.11 of the FSAR. By amendment 3 to the FSAR, the applicant stated, in response to a request for information, that all safety-related equipment is designed Class 1E, is included in a qualification program, and is designed and qualified to perform its safety function in normal and design basis event environments. Based on these statements, the staff concludes that Class 1E equipment will meet the design requirements of GDC 4, qualification requirements of 10 CFR 50.49, the guidelines of Sections 4.2 and 4.7 of IEEE Standard 308-1974, and therefore is accceptable.

8.3.3.2 Compliance With General Design Criterion (GDC) 5

The applicant has met (except as noted) the requirements of GDC 5, "Sharing of Structures, Systems, and Components," with respect to structures, systems, and components of the ac and dc onsite power systems.

8.3.3.3 Physical Independence - Compliance With General Design Criterion (GDC) 17

8.3.3.3.1 Use of Regulating Transformers as Isolation Devices

Table 8.3.2 and Section 8.3.1.1.17 of the FSAR indicates that there are six Class 1E isolating voltage regulation transformers allocated to the four vital bus systems. They serve to isolate either certain designated non-Class 1E loads from the Class 1E portion of the system or to isolate Class 1E train loads from the Class 1E channel portion of the system.

The FSAR further states that each of the isolating transformers is fully qualified and is designed such that a continuous bolted short circuit on the secondary winding will not be reflected on the primary winding. By amendment 3 to the FSAR, the applicant, in response to a request for information, indicated the following:

a. Oscillograph traces of transformer input current showed 101.6 to 109.4 percent of the transformer's full load rating current being input with the output terminals shorted.

b. The transformers were specified to limit input current to the transformer to 150 percent of its full load rating under short circuit.

c. The vital bus UPS system can supply the full burden of the transformer with a shorted secondary.

d. Output circuits are run in dedicated conduit from the transformer to the connected load

e. The non-Class 1E loads are composed of control and instrument circuits.

Based on the above information the staff is unable to conclude the acceptability of these transformers as isolation devices. Areas that require additional information or clarification include:

a. Duration of time to which the isolation transformer was tested with justification of its adequacy.

b. Qualification test report that demonstrates the capability of the transformers to withstand anytime during its design life the continuous bolted short circuit on its secondary winding.

c. Analysis that demonstrates the capability of the vital UPS system to supply its normal loads plus the 150 percent load specified for the shorted transformer.

d. Extent of compliance of the Non-Class 1E output circuits from the transformer to and including the load to all the requirements placed on Class 1E circuits.

This item will be pursued with the applicant and the results of the staff review will be reported in a supplement to this report.

8.3.3.2 Separation of Containment Electrical Penetrations

Section 8.3.1.4 (part 2, item 2b(5)) of the FSAR stated that containment electrical penetrations meet separation requirements of currently approved design procedures which comply with the intent of IEEE Standard 384-1981 for limited hazard areas. Section 5.5 of IEEE Standard 384-1974 (which is the currently approved NRC guideline for this subject) requires that redundant penetrations be widely dispersed around the circumference of the containment. Recent designs, approved by NRC on this subject, locate redundant electrical penetrations in different rooms or on opposite sides of containment. The Beaver Valley design, however, locates redundant penetrations in a single room in a 21 by 5 matrix with eight feet (center to center) between redundant penetrations. The Beaver Valley design does not meet the requirements nor the intent of IEEE Standard 384-1974 (or IEEE Standard 384-1981) as stated in the FSAR.

In response the applicant by amendment 3 to the FSAR, stated that containment electrical penetrations are physically separated over a 120-degree arc of the containment and are located on two distinct building elevations. This statement contradicts the above design description for Beaver Valley penetrations. This item will be pursued with the applicant and the results of the staff review will be reported in a supplement to this report.

8.3.3.3.3 Use of a Single Piece of Steel as a Barrier

Section 8.3.1.4 of the FSAR has been interpreted to mean that a single piece of steel or steel tray cover is to be installed as a barrier between raceways that are separated by less distance than allowed by Beaver Valley separation criteria. The objective of the barrier is to preclude failures of cables located in one raceway from causing failure of cables located in another raceway.

The applicant by amendment 3 to the FSAR and in response to a request for information stated that additional analysis and testing will be submitted on or before June 30, 1984. This item will be pursued with the applicant the results of the staff review will be reported in a supplement to this report.

8.3.3.3.4 Barrier Configurations

Section 8.3.1.4 (part 2, item 2a(9)) of the FSAR, stated that barriers will extend to the maximum extent practical beyond the area of exposure. The applicant was requested to identify each location where a barrier will extend less than 12 inches beyond the area of exposure and provide an analysis for each identified location that demonstrates the adequacy of the lesser separation.

In response the applicant by amendment 3 deleted item 2a(9) from the FSAR and stated that the requested information and analysis will be developed and submitted in a future amendment to the FSAR. This item will continue to be pursued with the applicant and the results of the staff review will be reported in a supplement to this report.

8.3.3.3.5 Separation Inside Panels, Cabinets, or Enclosures

Section 8.3.1.4 (part 2 item 2b(6)) of the FSAR stated that wiring within control switchboards and cabinets has been specified in currently approved design procedures to meet the intent of the independence requirements of IEEE Standard 384-1981. Based on this statement it appears that neither 6 inches of spatial separation or a barrier need be installed between redundant cables or between Class 1E and non-Class 1E cables inside panels or cabinets.

The applicant by amendment 3 revised the FSAR to state that wiring within control switchboards and instrumentation cabinets has been specified to meet the requirements of IEEE Standard 384-1974. The staff interprets this statement in the FSAR to mean that all redundant cables, wires, or circuits within cabinets or enclosures will be separated by 6 inch or a barrier. This meets staff guidelines, the independence requirement of GDC 17, and is acceptable.

Separation between Class 1E and Non Class 1E cables inside panels or enclosures has not been specifically addressed. This item will be pursued with the applicant and the results of the staff review will be reported in a supplement to this report.

8.3.3.3.6 FSAR Description of Physical Separation

IEEE Standard 384-1974, as augmented by Regulatory Guide 1.75 (Revision 2), provides minimum raceway separation guidelines acceptable to the staff for complying with the physical independence requirement of Criterion 17 of Appendix A to 10 CFR 50. These guidelines, however, have not been followed in the design of Beaver Valley Power Station, Unit 2. The unique Beaver Valley designs for separation of raceways was only partially described in Section 8.3.1.4 of the FSAR.

Description of separation at Beaver Valley and analysis for lesser separation has not been provided in amendment 3 to the FSAR. This item will be pursued with the applicant and the results of the staff evaluation will be reported in a supplement to this report.

8.3.3.3.7 Use of 12 Inches of Separation Versus the Recommended 36 Inches

Section 8.3.1.4 of the FSAR indicated that 12 inches of horizontal separation will be provided between redundant Class 1E cable trays located in general plant areas versus 3 feet required by Section 5.1.4 of IEEE Standard 384-1974.

The applicant, by amendment 3 to the FSAR, deleted reference for 12 inches of horizontal separation and stated in its place that physical independence of redundant Class 1E circuits throughout the plant is maintained by having redundant raceways physically separated to conform with minimum free air space requirements cited in IEEE Standard 384-1974 as augmented by Regulatory Guide 1.75. This design meets staff guidelines and physical independence requirements of GDC 17 and is acceptable. However, the applicant in contradiction also states that physical barriers, tests, and/or analysis are provided to assure the independence of redundant Class 1E circuits. The staff is unable to determine what specific design criteria exists for physical separation of circuits at any given area at Beaver Valley Unit 2. This item as well as the following listed items will be pursued with the applicant with the results of the staff review reported in a supplement to this report.

a. Section 8.3.1.4 of the FSAR indicated that approximately 2 feet of vertical separation will be provided between redundant Class 1E cable trays versus 3 or 5 feet required by Sections 5.1.3 and 5.1.4 of IEEE Standard 384-1974. The applicant by amendment 3 to the FSAR, deleted reference to the 2 feet of vertical separation.

b. Section 8.3.1.4 of the FSAR indicated that 6 inches of horizontal separation will be provided between Class 1E and non-Class 1E cable trays versus 12 inches or 3 feet required by Sections 5.1.3 and 5.1.4 of IEEE Standard 384-1974. The applicant, by amendment 3 to the FSAR, deleted all reference to specific design separation requirements between Class 1E and non Class 1E cables.

c. Section 8.3.1.4 of the FSAR indicated that 12 inches of vertical separation will be provided between Class 1E and non Class 1E cable trays versus 3 or 5 feet required by Sections 5.1.3 and 5.1.4 of IEEE Standard 384-1974.

8.3.3.4 Compliance With the Guidelines of NUREG-0737, "Clarification of TMI Action Plan Requirements"

Two TMI items relating to GDC 17 are identified in NUREG-0737. These items are II.E.3.1, "Emergency Power Supply for Pressurizer Heaters," and II.G.1, "Emergency Power for Pressurizer Equipment." The background, the NUREG position, and clarification of the positions are included in the NUREG report.

Emergency Power Supply for Pressurizer Heaters (II.E.3.1)

Description of compliance to each of seven clarifications associated with this TMI item have not been included in the FSAR as stated in response to a request for information. Description of compliance will be pursued with the applicant and the results of the staff evaluation will be reported in a supplement to this report.

Emergency Power for Pressurizer Equipment (II.G.I)

Similarly description of compliance to each of four clarifications associated with this TMI item have not been included in the FSAR. Description of compliance will be pursued with the applicant and the results of the staff evaluation will be reported in a supplement to this report.

8.3.3.5 Electrical Independence Between Power Supplies to Controls Located in Control Room and Remote Locations

Section 8.3.1.1.10 of the FSAR indicates that controls for the diesel generator and Class 1E circuit breakers are located in the control room and at remote locations. By amendment 3 to the FSAR, in response to a request for information, the applicant indicated that independence of controls between these locations is provided by transfer relays operated by transfer pushbuttons. The details for the electrical independence between power supplies to these controls will be pursued with the applicant and the results of the staff review will be reported in a supplement to this report.

8.3.3.6 Compliance With General Design Criterion (GDC) 18

The applicant has met the requirements of GDC 18, "Inspection and Testing of Electric Power Systems," with respect to the onsite ac and dc power system. The onsite power system is designed to be testable during operation of the nuclear power generating station as well as during those intervals when the station is shut down.

8.3.3.7 Compliance With General Design Criterion (GDC) 50

The applicant has met (except as noted) the requirements of GDC 50, "Containment Design Bases," with respect to electrical penetrations containing circuits of the safety and nonsafety onsite power systems. Criterion 50 requires, in part, that the reactor containment structures, including penetrations, be designed so that the containment structure and its internal compartments can accommodate without exceeding the design leakage rate, and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident.

The following items address the problem revealed during the staff review and resolution or status concerning them.

8.3.3.7.1 Description and Analysis of Compliance to GDC 50

In regard to electrical containment penetrations, a description as to how the Beaver Valley design meets the requirements of Criterion 50 of Appendix A to 10 CFR 50, with analysis demonstrating compliance, has not been provided in Section 8 of the FSAR.

By amendment 3 to the FSAR, the applicant, in response to a request for information, provided a description with results of test and analysis to show compliance to GDC 50. Based on this information, the staff considers this item resolved. Documentation of the description and analysis in Section 8.0 of the FSAR will be pursued with the applicant and the results of the staff review will be reported in a supplement to this report.

8.3.3.7.2 Compliance With RG 1.63

Section 8.3.1.2.1 of the FSAR indicates that primary and backup containment electrical penetration protection is provided only where the available fault-current exceeds the current-carrying capabilities of penetration conductors. This design for containment electrical penetration protection does not meet the guidelines of position 1 of Regulatory Guide 1.63. Positfon 1 requires primary and backup protection where maximum available fault-current exceeds the current-carrying capability of the penetration versus capability of the conductors.

By amendment 3 to the FSAR, the applicant indicated that the Beaver Valley design provides primary and backup protection as required by RG 1.63 and that the following additional information would be provided by March 1984:

a. fault-current versus time curve for each representative type cable conductor which penetrates primary containment

b. test report which verify the capability of penetration to withstand the total range of time versus fault current for worst case environmental conditions

Revision to the FSAR to indicate compliance to RG 1.63 without exception and review of the above additional information will be pursued with the applicant. The results of the staff review will be reported in a supplement to this report.

8.3.4 Evaluation Findings

The review of the onsite ac and dc power system for the Beaver Valley plant covered single line diagrams, station layout drawings, schematic diagrams, and descriptive information. The basis for acceptance of the onsite power systems in the staff's review was conformance of the design criteria and basis to the Commission's regulations as set forth in the General Design Criteria (GDC) of Appendix A to 10 CFR 50. The staff concludes that the plant design is acceptable, meets the requirements of GDC 2, 4, 17, 18 and 50, and conforms to applicable guidelines of regulatory guides, branch technical positions, and NUREG reports and is acceptable except as noted in preceding sections.

15 ACCIDENT ANALYSES

The accident analyses for Beaver Valley Unit 2 have been reviewed in accordance with Section 15 of the SRP (NUREG-0800). Conformance with the acceptance criteria, except as noted for each of the sections, formed the basis for concluding that the design of the facility for each of the areas reviewed is acceptable.

In accordance with SRP 15.1.1, Paragraph I, the applicant evaluated the ability of Beaver Valley Unit 2 to withstand anticipated operational occurrences and a broad spectrum of postulated accidents without undue hazard to the health and safety of the public. The results of these analyses are used to show conformance with GDC 10, 15, 27, and 31.

For each event analyzed, the worst operating conditions and the most limiting single failure were assumed, and credit was taken for minimum engineered safeguards response. In questions 440.73 and 440.74 the staff has asked the applicant to:

1. Supply listings of the single failures which were assumed for each event in the Chapter 15 analyses.

2. Supply the limiting single failure that results in the peak pressure or limiting performance for each event.

3. Show the effect of a loss of offsite power on all anticipated operational occurrences and postulated accidents.

When this information is received it will be incorporated into the evaluations of the individual events.

Parameters specific to individual events were conservatively selected. Two types of events were analyzed

(1) those incidents that might be expected to occur during the lifetime of the reactor

(2) those incidents not expected to occur that have the potential to result in significant radioactive material release (accidents)

The nuclear feedback coefficients were conservatively chosen to produce the most adverse core response. The reactivity insertion curve, used to represent the control rod insertion, accounts for a stuck rod; it is in accordance with GDC 26.

For transients and accidents, the applicant used a method that conservatively bounds the consequences of the event by accounting for fabrication and operating uncertainties directly in the calculations. DNBRs were calculated using the W-3 correlation with a modified spacer factor R, with a minimum DNBR of 1.3 used as the threshold for fuel failure.

The applicant accounts for variations in initial conditions by making the following assumptions as appropriate for the event being considered:

	3-Loop Operation	2-Loop Operation
Core Power (MWt)	2652 + 2%	1724 + 2%
Average Reactor Vessel Temperature (°F)	576.2 ± 4%	566.0 ± 4%
Pressure (psi) (at pressurizer)	2250 ± 30	2250 ± 30

The staff concludes the assumptions for initial conditions are acceptable because they are conservatively applied to produce the most adverse effects. These assumed values will form the basis for the technical specification limits. For transients and accidents used to verify the ESF design, the applicant used the safeguards power design value of 2780 MWt.

The applicant has also analyzed several events expected to occur one or more times in the life of the plant. A number of transients can be expected to occur with moderate frequency as a result of equipment malfunctions or operator errors in the course of refueling and power operation during the plant lifetime.

Specific events were reviewed to ensure conformance with the acceptance criteria provided in the SRP.

The acceptance criteria for transients of moderate frequency in the SRP include the following:

(1) Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design values (Section III of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code).

(2) Fuel clad integrity shall be maintained by ensuring that the minimum DNBR will remain above the 95/95 DNBR limit for PWRs. (The 95/95 criterion discussed in Section 4.4 of this SER provides a 95% probability, at a 95% confidence level, that no fuel rod in the core experiences a DNB.)

(3) An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.

(4) For transients of moderate frequency in combination with a single failure, no loss of function of any fission product barrier, other than fuel element cladding, shall occur. Core geometry is maintained in such a way that there is no loss of core cooling capability and control rod insertability is maintained.

Conformance with the SRP acceptance criteria for anticipated operational occurrences constitutes compliance with GDC 10, 15, and 26 of Appendix A to 10 CFR 50. See Section 6.8 of this SER for a discussion of auxiliary feedwater system conformance to TMI Action Plan Item II.E.1.1 and Sections 6.8 and 7.3.1.7 for a discussion of compliance with TMI Action Plan Item II.E.1.2.

The transients analyzed are protected by the following reactor trips:

(1) power range high neutron flux
(2) high pressure

(3) low pressure
(4) overpower ΔT
(5) overtemperature ΔT
(6) low coolant flow
(7) pump undervoltage/underfrequency

(8) low steam generator water level
(9) high steam generator water level

Time delays to trip, calculated for each trip signal, are included in the analyses. See Section 4.6 of this SER for a discussion of the staff review of reactivity control system functional design.

All of the events that are expected to occur with moderate frequency can be grouped according to the following plant process disturbances: changes in heat removal by the secondary system, changes in reactor coolant flow rate, changes in reactivity and power distribution, and changes in reactor coolant inventory. Design-basis accidents have been evaluated separately and are discussed at the end of this section of the SER.

15.1 <u>Increase in Heat Removal by the Secondary System</u>

The applicant's analysis of events that produced increased heat removal by the secondary system is addressed in the following paragraphs.

15.1.1 Decrease in Feedwater Temperature

The consequences of a decrease in feedwater temperature transient are bounded by those in Sections 15.1.2 and 15.1.4. The peak pressure is less than that in Section 15.1.2. The minimum DNBR is greater than that in Section 1.5.1.4.

15.1.2 Increase in Feedwater Flow

Increases in feedwater flow decrease the temperature of the reactor coolant water. Due to the negative moderator temperature coefficient this will insert positive reactivity and increase core power.

In Section 15.1.2.1 of the FSAR the applicant states that for these events the high neutron flux trip, overtemperature ΔT trip, and overpower ΔT trip prevent any power increase which could lead to a DNBR less than the limit value of 1.30. However, the only analytical results presented for these events are those where a steam generator hi-hi level trip closes all feedwater control and isolation valves, trips the main feedwater pumps, trips the turbine, and initiates a reactor trip. The applicant states that continuous addition of feedwater is prevented by the steam generator hi-hi level trip.

This analysis shows that the maximum reactivity insertion rate due to an increase in feedwater flow occurs at no-load conditions and is less than the maximum value calculated for an inadvertent control rod withdrawal, which is evaluated in Section 15.4 of this SER. However, this analysis also shows that an increased feedwater flow event can cause a peak RCS pressure of 2270 psia. This is below the design pressure of 2485 psig, but it is the highest RCS pressure the applicant calculated for any of this group of events.

15.1.3 Increase in Steam Flow

The consequences of an increase in steam flow transient are bounded by those in Sections 15.1.2 and 15.1.4. The peak pressure is less than that in Section 15.1.2. The minimum DNBR is greater than that in Section 15.1.4.

15.1.4 Inadvertent Opening of a Steam Generator Relief Valve or Safety Valve

The transient that is most limiting of this group of transients with respect to fuel performance is the inadvertent opening of the steam generator relief or safety valve. The suddenly increased steam demand causes a reactor power increase which results in a reactor trip due to high neutron flux, overtemperature, or overpower signals. The continued steam flow through the open valve will cause additional cooldown which will, because of the negative moderator temperature coefficient, result in positive reactivity. The safety injection system (SIS) will inject highly concentrated boric acid from the boron injection tank into the primary coolant system on either two out of three pressurizer low pressure signals, or two out of three low steamline pressure signals in any one loop. This ensures the reactor will be shut down during any subsequent cooldown. The normal steam generator feedwater would be isolated automatically upon SIS initiation, and then the plant would be gradually cooled down with only safety-grade equipment. DNB does not occur during this transient.

The applicant has provided results of its study for a transient of this group in combination with its limiting single failure. No credible single failure has been identified that could result in a more limiting peak reactor coolant system pressure or DNBR than that from the events themselves.

The applicant's analyses show that for transient events leading to an increase in heat removal by the secondary system (with or without single failure), the minimum DNBR is 1.3. Thus no fuel failure is predicted to occur, core geometry and control rod insertability are maintained with no loss of core cooling capability, and the maximum reactor coolant system pressure remains below 110% of design pressure. The staff finds the results of these analyses in conformance with the acceptance criteria of SRP 15.1.1 through 15.1.4, and, therefore, acceptable.

15.1.5 Steamline Rupture ~~Accident~~

The applicant has submitted analyses of postulated steamline breaks that show no fuel failures attributed to the accident. These results are similar to those obtained for previously reviewed Westinghouse three-loop plants.

A postulated double-ended rupture at hot standby power with no decay heat was analyzed as the worst case. Since the steam generators have integral flow restrictors with a 1.4 ft^2 throat area, any rupture with a break area greater than 1.4 ft^2, regardless of location, will have the same effect on the system as a 1.4 ft^2 break; so this was assumed in the analysis. The doubled-ended rupture would cause the reactor to increase in power due to the decrease in reactor coolant temperature. The reactor would be tripped by either reactor overpower ΔT or by the actuation of the SIS. The SIS will be actuated by any of the following: two out of three low pressurizer pressure signals; two out of three HI-1 containment pressure signals; or two out of three low steamline pressure signals in any one loop. The transient is terminated using only safety-grade equipment. The injection of highly borated water ensures the reactor is returned to and then maintained in a shutdown condition.

The staff concludes that the consequences of postulated steamline breaks meet the relevant criteria in GDC 27, 18, 31, and 35 regarding control rod insertability and core coolability and TMI Action Plan Items. This conclusion is based upon the following:

(1) The applicant has met the criteria of GDC 27 and 28 by demonstrating that fuel damage, if any, is such that control rod insertability will be maintained, and there will be no loss of core cooling capability. The minimum DNBR experienced by any fuel rod was ≥1.30, resulting in none of the fuel elements being predicted to experience cladding perforation.

(2) The applicant has met the criteria of GDC 31 with respect to demonstrating the integrity of the primary system boundary to withstand the postulated accident.

(3) The applicant has met the criteria of GDC 35 with respect to demonstrating the adequacy of the emergency cooling systems to provide abundant core cooling and reactivity control (via boron injection).

(4) A mathematical model, which accounts for incomplete coolant mixing in the reactor vessel, has been reviewed and found acceptable by the staff. This model was used to analyze the effects of steamline breaks inside and outside of containment, during various modes of operation, with and without offsite power.

(5) The parameters used as input to this model were reviewed and found to be suitably conservative.

15.2 Decrease in Heat Removal by the Secondary System

The applicant's analyses of events that result in a decrease in heat removal by the secondary system are presented below.

15.2.1 Steam Pressure Regulator Malfunction or Failure that Results in Decreasing Steam Flow

In Section 15.2.1 of the FSAR the applicant states that any steam flow decrease caused by a malfunction or failure of any steam pressure regulator is conservatively bounded by the turbine trip event and analyzed in Section 15.2.3.

15.2.2 Loss of External Load

In Section 15.2.2 of the FSAR the applicant states that the results of the turbine trip event analysis are more severe than those expected for the loss of external load. The reason given is that a turbine trip actuates the turbine stop valve whereas a loss of external load actuates only the turbine control valves. Since the stop valve can more suddenly cut off the steam flow to the turbine this is a more severe "decreased heat removal" transient.

15.2.3 Turbine Trip

Assuming offsite power is available to run the reactor coolant pumps, the applicant analyzed the turbine trip event for a complete loss of steam load from full power without a direct reactor trip and with only the pressurizer and steam generator safety valves assumed for pressure relief. These assumptions result in the highest peak RCS pressure for any "decreased heat removal" event. The calculated peak value is 2560 psia, which is well below the ASME limit of 110% of the design pressure. For these assumptions the minimum DNBR is 1.75, which is well above the minimum limiting value of 1.30.

The applicant's analyses show that if instead of relying on just the safety valves, the pressurizer spray and PORV's are used to limit the pressure during this turbine trip event, the minimum DNBR can go down to 1.60. If a stuck open

PORV were to be assumed as the single failure during this course of action, it appears that the DNBR could go lower. The applicant has not discussed the possibility of a stuck open PORV or atmospheric steam dump valve being the worst single failure during this course of action.

The consequences of a turbine trip without offsite power available are discussed in Section 15.2.6.

15.2.4 Inadvertent Closure of Main Steam Isolation Valves

Consequences are the same as those discussed in Sections 15.2.3 and 15.2.6.

15.2.5 Loss of Condenser Vacuum

Consequences are the same as those discussed in Sections 15.2.3 and 15.2.6.

15.2.6 Loss of Nonemergency ~~AC~~ Power to the ~~Plant~~ Station Auxiliaries

A loss of nonemergency ac power event is more limiting than the turbine-trip-initiated decrease in secondary heat removal without loss of ac power because the reactor coolant pumps are lost and the subsequent flow coastdown further reduces the amount of heat the primary coolant can remove from the core. In this transient, the loss of offsite power is closely followed by a turbine trip and reactor trip. The reactor trip is assumed to come from low-low steam generator level which is the second safety-grade trip. The emergency feedwater system is automatically started and one electric-motor-driven pump is assumed to be feeding all three steam generators.

The applicant's LOFTRAN analysis shows that the natural circulation flow available adequately transfers the decay heat from the core to steam generators, which are being fed with emergency feedwater flow. The steam which is generated is assumed to be relieved through the steam generator safety valves. The primary system relief valves are assumed not to function.

The emergency feedwater comes from the primary plant demineralizer water storage tank (PPDWST) which, the applicant states in FSAR Section 10.4.9.1, contains sufficient water to reduce the hot leg temperatures to 350°F for this transient. At 350°F the RHRS can be started to take away the decay heat.

The DNBR remains above 1.30 throughout this transient, and the peak RCS pressure remains below 110% of the design pressure.

15.2.7 Loss of Normal Feedwater Flow

The consequences of this anticipated operational occurrence are more severe if a concurrent loss of offsite power is assumed. However, if a loss of offsite power is assumed the consequences will be the same as the loss of nonemergency ac power event discussed in Section 15.2.6.

15.2.8 Feedwater System Pipe Breaks

The applicant has provided a feedwater line break analysis for Beaver Valley Unit 2 using assumptions that minimize secondary system heat removal capability, maximize heat addition to the primary system coolant, and maximize the calculated primary system pressure. A double-ended rupture of the largest feedwater line was assumed, as well as failure of the turbine-driven auxiliary feedwater pump to start and supply emergency feedwater to the steam generator.

The applicant used the NRC approved LOFTRAN code to do this analysis. The analysis assumed that with a single failure of the auxiliary feedwater system, emergency feedwater flow is supplied to two intact steam generators by only one electric-motor-driven auxiliary feedpump. This is sufficient feedwater flow to adequately remove the residual heat after reactor shutdown. The use of only safety-grade equipment will mitigate this accident. No fuel damage was calculated to occur, and the peak calculated pressurizer pressure was approximately 2500 psia. As required for all other events a list of the single failures that were considered and the most limiting single failure must be provided.

15.3 Decreases in Reactor Coolant Flow Rate

15.3.1/15.3.2 Loss of Forced Reactor Coolant Flow, Including Trip of Pump and Flow Controller Malfunctions

The applicant has analyzed the total loss of forced reactor coolant flow event that bounds partial loss of forced reactor coolant flow. This event was reviewed with the procedures and acceptance criteria set forth in SRP 15.3.1 -15.3.2.

The loss of offsite power and resulting loss of all forced coolant flow through the reactor core causes an increase in the average coolant temperature and a decrease in the margin to DNB. The reactor is tripped from an undervoltage trip monitoring the reactor coolant pump (RCP) power supply, and a minimum DNBR of 1.47 is reached 3.2 seconds into the transient. The maximum calculated RCS pressure is 2310 psia during the transient.

15.3.3/15.3.4 Reactor Coolant Pump Rotor Seizure and Shaft Break ~~Accident~~

The applicant has analyzed the reactor coolant pump (RCP) rotor seizure and shaft break events with the LOFTRAN and FACTRAN computer codes. Since the initial rate of reduction of coolant flow is greater after an RCP rotor seizure, this is the limiting event. For the analyses the applicant assumed that the fuel cooling goes into the nucleate boiling regime (i.e., DNB) immediately at the beginning of the transient. The maximum RCS pressure will occur in the event of an RCP rotor seizure while only two of the three loops are operating. This maximum pressure is calculated to be 2647 psia with only the opening of the pressurizer and steam generator safety valves. The applicant states that 2647 psia is below the faulted condition stress limit of the RCS.

In response to a question on a loss of offsite power (LOOP) during these events, the applicant states that a LOOP will have only a negligible effect on the critical parameters of RCS pressure and clad temperature and that it would have no effect whatsoever on the conclusions. The staff finds that a quantitative analysis of the worst case, which would have only two loops in operation, with a concurrent loss of offsite power is needed for the evaluation of this issue.

The staff's evaluation and finding on fuel damage and consequent control rod insertability and core cooling considerations during this event are included in SER Section 4.2. The LOFTRAN computer code has been approved by the NRC. The remaining staff findings are

(1) The parameters used as input to the mathematical model are suitably conservative.

(2) The use of "Service Limit C" of the ASME Code is acceptable for conforming to GDC 31 and demonstrating the integrity of the RCS during this accident; the maximum pressure is below this limit.

15.4 ~~Changes in~~ Reactivity and Power Distribution Anomalies

15.4.4/15.4.5 Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature

In FSAR Section 15.4.4, the applicant provides the results of an analysis for startup of an inactive reactor coolant pump event. This event was reviewed with the procedures and acceptance criteria set forth in SRP 15.4.4.

During the first part of the transient, the increase in core flow with cold water results in an increase in nuclear power and a decrease in core average temperature. Reactivity addition for the inactive loop startup event is the result of the decrease in core inlet water temperature. This transient was evaluated by the applicant using a mathematical model that has been reviewed and found acceptable to the staff. The maximum calculated RCS pressure is 2310 psia and the minimum DNBR is above 1.3 throughout the transient.

15.4.6 Inadvertent Boron Dilution

Various chemical and volume control system (CVCS) malfunctions which could lead to an unplanned boron dilution incident have been reviewed. The malfunctions that allow the operator the shortest time for corrective action have been analyzed starting from plant conditions of startup, power operation (automatic and manual), hot standby, and cold shutdown. The applicant used acceptably conservative assumptions in these analyses. The results show that the operator has at least 15 minutes between the time when an alarm announces an unplanned moderator dilution and the time of loss of shutdown margin, i.e., criticality.

The maximum reactivity insertion rate by boron dilution was found to be 1.5×10^{-5} $\Delta k/k$ (1.5 pcm) per second. In the event the operator does not stop the dilution, the DNBR will still remain above 1.49, and the RCS and main steam pressures will remain below 110% of design.

In response to a question on protection from inadvertent boron dilution during refueling, the applicant stated that during refueling the RCS is isolated from the potential source of unborated water. This isolation is accomplished by having the operators place danger tags on the primary grade water header isolation valves, or by locking these valves closed whenever the RCS water is below the normal level. The operator performing these tasks is required to sign off on each step of a procedural checklist. This long term use of administrative controls to prevent an inadvertent boron dilution during refueling has not been accepted by the staff on other plants, and will be evaluated. The staff is not at this point, convinced that a design basis event can be eliminated from detailed evaluation based on administrative means alone. We will report the resolution of this issue in a subsequent safety evaluation.

With the exception of the refueling mode the staff concludes that the analysis for the decrease in reactor coolant boron concentration event is acceptable and conforms to General Design Criterion 10, 15, and 26. This conclusion is based on the following:

1. The applicant has met the criteria of GDC 10 with respect to demonstrating that the specified acceptable fuel design limits are not exceeded for this event. This criterion has been met since the results of the analysis showed that the thermal margin limits are satisfied as indicated by SER Section 4.4.

2. The applicant has met the criteria of GDC 15 with respect to demonstrating that the reactor coolant pressure boundary limits have not been exceeded for this event. This criterion has been met since the analysis showed that the maximum pressure in the reactor coolant and main steam systems did not exceed 110% of the design pressure.

3. The applicant has met the criteria of GDC 26 with respect to demonstrating that the control rod system has the capability of overcoming the effects of boron dilution events during reactor operation. The applicant has demonstrated conformance with these criteria by showing that under the postulated accident conditions, and with appropriate margins for stuck rods, the specified acceptable fuel design limits are not exceeded.

15.5 Increases in Reactor Coolant System Inventory

15.5.1 Inadvertent Operation of the Emergency Core Cooling System During Power Operation

ECCS operation could be initiated by a spurious signal or an operator error. Two cases were examined, one in which reactor trip occurs simultaneously as a result of the safety injection signal, and the other in which the reactor trips later in the transient because of low reactor coolant system (RCS) pressure. The reactor pressure decreases during the initial phase of the transient and then increases to a peak pressure of 2350 psia at 200 seconds into the transient. The DNBR never drops below its initial value for either case. All of these transients are terminated by use of only safety-grade systems. If the operator fails to turn off the HHSI/charging pumps the safety valves will open. Continued operation of these pumps would overfill the Pressure Relief Tank. However, as stated in Table 6.3-1 of the FSAR the cutoff head of the HHSI/charging pumps is 6000 ft (2600 psig); so they cannot create 110% of the reactor vessel design pressure (2733 psig) and thus cannot fail the vessel.

15.5.2 CVCS Malfunction That Increases Reactor Coolant Inventory

Evaluation of consequences is included in Section 15.4.6.

15.6 Decrease in Reactor Coolant Inventory

15.6.1 Inadvertent Opening of a Pressurizer Safety or Relief Valve

In FSAR Section 15.6.1, the applicant provides the results of an analysis for inadvertent opening of a pressurizer safety valve. During this event, nuclear power is maintained at the initial value until reactor trip occurs on low pressurizer pressure. The DNBR decreases initially, but increases rapidly following the trip. The minimum DNBR of 1.50 occurred at 31 seconds into the transient. The RCS pressure decreases throughout the transient.

15.6.3 Steam Generator Tube Rupture

In response to the staff's concern that 30 minutes is not sufficient to diagnose and isolate a stream generator tube rupture, the applicant has provided additional data regarding the systems response and radiological consequences after a steam generator tube rupture accident. This information, however, did not support the isolation time of the affected steam generator at 30 minutes.

Upon receipt of additional information, the staff will complete the review of the consequences of this accident and provide our evaluation.

15.6.5 LOCAs

In FSAR Section 15.6.5, the applicant has analyzed the double-ended cold leg guillotine (DECLG) as the most limiting large-break LOCA. The analysis was done for three different flow coefficients. The results of these show that the DECLG with a Moody break discharge coefficient of 0.4 is the worst case. In this analysis, the peak clad temperature reached is 2179°F. For the small-break LOCA the applicant has determined that a cold leg rupture of less than 10-in. diameter is the most limiting. The analysis was performed for 3-in., 4-in. and 6-in.-diameter breaks. The results show that the 3-in.-diameter break is the worst case, and it results in a peak clad temperature of 1985°F. Both of these accidents are terminated by SIS and ECCS operations. Only safety-grade equipment is used to mitigate the accident.

The applicant has performed analyses of the performance of the ECCS in accordance with the Commission's regulations (10 CFR 50.46 and Appendix K to 10 CFR 50).

The analyses considered a spectrum of postulated break sizes and locations. As shown in NUREG-0390, these analyses were performed with an evaluation model that had been previously reviewed and approved by the staff. The results show that the ECCS satisfy the following criteria:

(1) The calculated maximum fuel rod cladding temperature does not exceed 2200°F.

(2) The calculated maximum local oxidation of the cladding does not exceed 17% of the total cladding thickness before oxidation.

(3) The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam does not exceed 1% of the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

(4) Calculated changes in core geometry are such that the core remains amenable to cooling.

(5) After any calculated successful initial operation of the ECCS, the calculated core temperature is maintained at an acceptably low value and decay heat is removed for the extended period of time required by the long-lived radioactivity.

The staff concludes that the calculated performance of the ECCS following postulated LOCA accidents conform to the Commission's regulations and to applicable regulatory guides and staff technical positions, and the ECCS performance is considered acceptable for the postulated accidents.

15.9 TMI Action Plan Requirements

15.9.1/15.9.2

II.K.1.5 Review ESF Valve Positions, Controls, and Related Test and Maintenance Procedures To Assure Proper ESF Functioning

II.K.1.10 Review and Modify Procedures for Removing ESF From Service To Assure Operability Status Is Known

The applicant states that the intent of these two items will be met when the Operating and Maintenance Procedures are written. They are scheduled to be completed in June, 1985. The acceptability of the measures taken to satisfy these items will be evaluated when these procedures are submitted.

15.9.3 II.K.2.13 Thermal Mechanical Report: Effect of High-Pressure Injection on Vessel Integrity for Small-Break LOCA With No Auxiliary Feedwater

Staff review of this item will be covered in NRC unresolved safety issue A-49, "Pressurized Thermal Shock."

15.9.4 II.K.2.17 Potential for Voiding in the Reactor Coolant System During Transients

Westinghouse has performed a study that addresses the potential for void formation in Westinghouse-designed NSSS during natural circulation cooldown/depressurization transients. This study has been submitted to the NRC by the Westinghouse Owners Group. As stated in R. Wayne Houston's December 6, 1983 memorandum to Gus C. Lainas entitled, "Multiplant Action Item F-33, Voiding in the Reactor Coolant System During Anticipated Transients," the results of this study have been accepted.

15.9.5 II.K.2.19 Sequential Auxiliary Flow Analysis

Sequential auxiliary feedwater flow criteria are only of concern to once-through steam generator designs. Since Westinghouse has inverted U-tube steam generator designs, the analysis requested by Item II.K.2.19 is not needed for Beaver Valley Unit 2.

15.9.6 II.K.3.2 Report on Overall Safety Effect of Power-Operated Relief Valve Isolation System

As a response to Item II.K.3.2, the applicant referenced a generic Westinghouse Owners Group submittal. Should staff generic review of this material conclude otherwise, NRC will request further consideration of modification of Beaver Valley Unit 2.

15.9.7 II.K.3.3 Reporting SV and PORV Challenges and Failures

The applicant states in FSAR Table 1.10-1 that it will be responsible for ensuring that any failure of PORVs or safety valves to close will be reported promptly to the NRC and that all challenges to PORVs and safety valves will be documented in the annual report. The staff concludes that the Beaver Valley Unit 2 procedures meet the criteria of this item and are acceptable.

15.9.8 II.K.3.5 Automatic Trip of RCPs During LOCA

In response to this criterion, the applicant stated that Westinghouse performed an analysis of delayed RCP trip during LOCA. This analysis is documented and is the basis for the Westinghouse position on RCP trip (i.e., automatic RCP trip is not necessary because sufficient time is available for manual tripping of the RCPs).

Westinghouse has submitted a generic report which is under review. The applicant should state whether or not it intends to endorse this report and comply with the criteria proposed in it assuming the NRC finds it acceptable.

15.9.9 II.K.3.10 Proposed Anticipatory Trip Modification

The applicant has not proposed any modification to its standard anticipatory trip. Therefore, no TMI action plan requirements are imposed.

15.9.10 II.K.3.17 Report on Outages of ECCS

The applicant states in Table 1.10-1 and in Section 13.5.2.1 of the FSAR that it will meet the intent of this item when the Operating and Maintenance Procedures are written. They are scheduled to be completed in June 1985. The acceptability of the measures taken to satisfy this item will be evaluated when these procedures are submitted.

15.9.11 II.K.3.25 Effect of Loss of AC Power on RCP Seals

In response to this criterion, the applicant stated that in the event of loss of offsite power, the RCP motor is de-energized, the diesel generators are automatically started, and both seal injection flow and component cooling water flow are automatically restored within seconds.

The staff concludes that the applicant's design meets the criteria of this item and is acceptable.

15.9.12 II.K.3.30 Revised Small-Break LOCA Methods To Show Compliance With 10 CFR 50; Appendix K

In response to this criterion, the applicant stated that Westinghouse has submitted a new small-break evaluation model to NRC. The staff is currently reviewing this submittal.

15.9.13 II.K.3.31 Plant-Specific Calculations To Show Compliance with 10 CFR 50.46

The applicant states that the present (i.e., July, 1983) Westinghouse small-break, loss-of-coolant evaluation model was used for the analyses which are discussed in FSAR Section 15.6.5. However, this does not constitute a review that shows Beaver Valley Unit 2 is in full compliance with 10 CFR 50.46. After the staff's review of this evaluation model is completed a specific submittal on this issue will be required.