November 15, 1984

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

FROM: R. W. Houston, Assistant Director
for Reactor Safety, DSI

SUBJECT: SAFETY EVALUATION REPORT FOR BEAVER VALLEY, UNIT 2

Plant Name: Beaver Valley Power Station, Unit 2
Docket No.: 50-412
Responsible Branch: LB #3, DL
Project Manager: M. Licitra, M. Ley
Review Branch: CSB: DSI
Review Status: Incomplete

Enclosed is the Safety Evaluation Report (SER) (Enclosure 1), for the Beaver Valley Power Station, Unit 2 (BVPS-2) as prepared by the Containment Systems Branch (CSB). This report is based on the staff's review of the applicant's Final Safety Analysis Report (FSAR) as amended, and the applicant's response to staff requests for additional information. We have noted that the FSAR contains blank tables with statements to the effect that information has been forwarded to the staff under separate cover. In many cases, this information has not been received; the applicant should be requested to provide a schedule for filing suitable amendments to complete the FSAR. In addition, the following unresolved items in the SER need to be addressed by the applicant.

1. The methodology used by the applicant to compute the mass and energy release rates from postulated reactor coolant pipe breaks for the containment analysis is currently under separate staff review. In this regard, the applicant's response to NRC Question 480.7 did not fully justify the use of the unapproved methodology.

2. The mass and energy release data for the postulated main steam line breaks have not been documented in the FSAR. Completion of our review of the applicant's main steam line break analysis is dependent on the receipt of this information.

3. The subcompartment design pressure differentials for the reactor cavity, steam generator and pressurizer compartments have not been documented in the FSAR. The applicant is required to complete Table 6.2-26 in the FSAR for completion of the staff's review.

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4. In the event of a LOCA, coolant flow velocities on the floor of the containment, at the time of recirculation spray system actuation, when the water level is low, are substantial and may transport debris to the sump; this could adversely affect pump performance. Therefore, the applicant should justify the acceptability of the 50 percent blockage assumption that was used to assess emergency sump performance, by assessing the susceptibility of insulation to become dislodged by virtue of its proximity to high energy line piping and be transported to the sump.

Please contact our staff if you have any questions regarding these items.

Enclosure 2 is the SALP input for this SER input in accordance with Office Letter No. 44.

Original Signed By
R. Wayne Houston

R. W. Houston, Assistant Director
for Reactor Safety, DSI

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CONTAINMENT SYSTEMS BRANCH
INPUT FOR SAFETY EVALUATION REPORT
BEAVER VALLEY POWER STATION - UNIT NO. 2
DOCKET NO.: 50-412

6.2 Containment Systems

The Containment Systems for Beaver Valley Power Station, Unit 2 include the containment structures and associated systems, such as the containment heat removal systems, containment isolation system, and containment hydrogen control system. These systems function to prevent or control the release of radioactive fission products which might be released into the containment atmosphere following onset of a postulated loss of coolant accident (LOCA), or fuel handling accident, mitigate the accumulation of combustible gases that can potentially be generated and mitigate the effects of secondary system pipe ruptures.

The staff has reviewed the information relating to the 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 10 CFR 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 reestablished within 60 minutes; this condition would be maintained for at least 30 days following onset of 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 funtional 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 values 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. The staff has not performed confirmatory analyses for the two MSLB cases due to a lack of information (see 6.2.1.4). Therefore, we are not in a position to conclude our evaluation at this time.

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 DBA) 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 of -0.08 psig. A barometric pressure of 14.36 psia was used in the analysis, which is based on climatological data for Pittsburgh (U.S. Dept. of Commerce, Weather Bureau, 1963-1964 Local Climatological Data, Pittsburgh, Pa, Greater Pittsburgh Airport), and adjusted to the plant grade.

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 methodologies used to calculate the mass and energy release rate data for the LOCA and MSLB accident have not been completely reviewed due to a lack of certain information (see Section 6.2.1.3 and 6.2.1.4 of the SER). Therefore, the staff can not complete its review of the applicant's analysis at this time. This will remain a confirmatory 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 barometric pressure at the plant site 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 anaysis 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 equal to the minimum RWST temperature). 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 Subcompartment Analyses

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, 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 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 compuer code.

The mass and energy release rates used in the subcompartment analyses were calculated using the SATAN-VI computer program (WCAP-8312A). The methodology described in WCAP-8312A was previously approved by the staff.

Separate discussions 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 is consistent with the recommendations concerning nodalization that are presented in NUREG/CR-1199, "Subcompartment Analysis Procedures Report." The staff has examined the applicant's nodal model and finds it to be in accordance with current NRC guideline as specified in NUREG/CR-1199 and, therefore, is acceptable.

Selection of the break size, location and use of restraints to limit the break area are discussed in Section 3.6. The assumed initial conditions were chosen to maximize the differential pressure response. The applicant calculated a peak differential pressure load on the reactor cavity wall of 115.9 psid, for the design basis 150-in^2 LDR.

The staff performed a confirmatory analysis using the COMPARE-MOD 1A computer code, which confirmed that the applicant's result is conservative. However, the design basis for the reactor cavity wall is not documented in the FSAR; therefore, the staff can not confirm that the reactor cavity wall design is acceptable. The applicant is requested to complete Table 6.2-26 in the FSAR.

The applicant also performed dynamic force and moment calculations on the reactor pressure vessel (RPV) from postulated pipe ruptures in the reactor cavity (see FSAR Section 5.4.14.3.1.1). The staff has reviewed the applicant's analytical approach, including methods and modeling for calculating asymmetric loads, and finds that it conforms withthe guidelines of NUREG-0609, "Asymmetrical Blowdown Loads on PWR Primary Systems". The staff's review of the structural aspects of the applicant's calculation of forces and moments on the RPV is discussed in Section 3.6.

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 707-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 for the steam generator wall is not documented in the FSAR.

Pressurizer Subcompartment Analyses

The applicant considered three breaks for the pressurizer cubicle, and the pressurizer relief tank cubicle; namely a spray line double-ended rupture (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 applicant's analysis of the spray line DER in the upper pressurizer cubicle gave a peak differential pressure of 18.07 psid across the pressurizer nodal boundary surface. However, the design basis for the pressurizer cubicle walls is not documented in the FSAR.

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. (See Table 6.2-4 in the FSAR). The applicant assumed minimum safeguards in determining the mass and energy releases, i.e., the loss of one emergency diesel resulting in minimum safety injection. 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 applicant's mass and energy release analysis is considered in four phases: blowdown, refill, reflood and post-reflood. The blowdown phase is the phase of the accident during which most of the energy contained in the reactor coolant system is released to the containment. The SATAN VI computer code was used by the applicant to determine the mass and energy addition rates to the

containnment during the blowdown phase of the accident. The model used for the blowdown transient is described in a Westinghouse letter (NS-TMA-2075) that is currently under staff review. At this time, we are not in a position to complete our review of the blowdown methodology. This will be a confirmatory item pending the completion of the staff's review.

The time delay due to lower plenum refill has been neglected by the applicant for containment analysis. Instead, the applicant has conservatively assumed that the bottom of the core is covered immediately after the end of blowdown.

The analysis of the reflood phase of the accident is important to pipe ruptures in the reactor coolant system cold legs since the steam and entrained liquid carried out of the core for these break locations, pass through the steam generators which constitute an additional energy source. The steam and entrained water leaving the core and passing through the steam generators will be evaporated and/or superheated to the temperature of the steam generator secondary fluid. The rate of enegy release to the containment during the reflood phase is proportional to the core flooding rate. The rupture of the cold leg at the pump suction results in the highest mass flow through the core, and thus through the steam generators.

During the core reflood phase of the accident, when the core is filling with water, mass and energy release rates were calculated by the applicant using the modified WREFLOOD code. This model is described in a Westinghouse letter that is also under staff review in conjunction with the review of the SATAN VI code. Staff acceptance of this model will be a confirmatory item pending the receipt of outstanding information and completion of the staff's review.

The applicant has included consideration of a possible additional energy release to the containment during the post-reflood phase. The post-reflood phase begins after the core has been recovered with water. During this phase, decay heat generation will produce boiling in the core and a two-phase mixture of steam and water will exist. In calculations performed by the FROTH code, the applicant assumed that this two-phase mixture of steam and water rises above the core and enters the steam generators. By this process the remainder

of the available steam generator energy is removed by boiling of the water entrained in the two-phase mixture and is carried into the containment as steam. In calculating the rate of energy removed from the steam generators, the applicant has used the maximum steam flow based on the hydraulic resistance of the system and the maximum steam generator heat transfer. A portion of the steam that flows through the unbroken loops through the ECCS injection points is assumed to be quenched before exiting to the containment. The mass and energy rates calculated by FROTH are used in the containment analysis to the time of containment depressurization.

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) which was previously approved by the staff. The MARVEL code describes the primary and secondary systems of a pressurized water reactor including the power excursion which may occur in the core following a MSLB. The code calculates heat flow from the core and intact steam generators into the primary system, and heat flow from the primary system into the affected steam generator. The primary system heat flow produces additional steam which is added to the containment. It is assumed that the flow from the break contains no entrained liquid so that the break flow is pure steam. This assumption maximizes the energy release to the containment. The analysis includes the blowdown of steam from the intact steam generators before closure of the isolation valves and from the unisolated steam lines and turbine plant piping. Feedwater flow is added to the affected steam generator based on the reduction in the discharge pressure calculated by the MARVEL code. No credit is taken for any feedwater flow reduction during the valve closure period. The unisolated feedwater mass is added to the steam generator inventory during the blowdown.

In the applicant's mass and energy release analysis, the unisolated feedwater line volume between the steam generator and the isolation valve was included as a source for additional high energy fluid to be discharged through the pipe break. Addition of auxiliary feedwater to the affected steam generator was

assumed to start at time zero and continue until manually stopped by the plant operator. The applicant has considered the long term blowdown of the water supplied by the auxiliary feedwater system. Auxiliary feedwater flow to the affected steam generator is limited to 43 lbm/sec by passive flow control devices installed in the line to each steam generator, and, for the analysis, is assumed to be manually terminated 30 minutes after the break. The blowdown rate is also limited by the rate at which water is added to the steam generator from the auxiliary feedwater system (if the main feedwater isolation valve in the broken loop fails to close, main feedwater will be terminated by the feedwater control valve). The staff has found the applicant's analysis has adequately considered the long term blowdown of water supplied by the auxiliary feedwater system, in accordance with the guidelines of IE Bulletin 80.04.

The mass and energy release data for the worst case MSLB's are not provided in the FSAR. The staff requests that this information be provided for review and to support the staff's confirmatory analysis. This matter will be a confirmatory item pending the receipt of the information.

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

Appendix K to 10 CFR 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 calculations, (see 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 cladding 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 Section 6.3.5 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 Summary and Conclusions

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. The method used by the applicant to compute the mass and energy release rates from postulated reactor coolant system pipe breaks for the containment analysis is currently under separate staff review. In this regard, the applicant's response to NRC Question 480.7 did not fully justify the use of the unapproved methodology.

2. 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.

3. For the subcompartment analyis, the design bases for the reactor cavity, steam generator and pressurizer compartments have not been documented in the FSAR.

6.2.2 Containment Heat Removal Systems

The function of the containment heat removal systems is to remove heat from the containment atmosphere 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 spray 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 subatmospheric 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, 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 checmical 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 maintain a subatmospheric 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 RWST reaches a predetermined low level, the flow from two of the RSS pumps is automatically diverted to the cold leg recirculation mode of 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 Classification," for engineered safety features. The applicant has provided information (FSAR Section 14.2, "Initial Test Program") in accordance with the guidelines of Regulatory Guide 1.68, "Initial Test Program for Water Cooled Nuclear Power Plants") which will ensure 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," provides design guidelines for containment sumps that are to serve as sources of water for the ECCS and 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 reviewed 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 separate 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.

The applicant has conducted sump model testing at the Alden Research Laboratory using a 1/3 scale hydraulic model of the sump. The objective of the testing was to evaluate the sump performance characteristics with a view towards eliminating conditions that may be conducive to the formation of air entrainment vortices and lowering the threshold containment water level at which vortex formation would be expected to be completely suppressed. The test program included two and four pump operation, up to 50 percent blockage of the sump screens or trash racks, and various farfield flow distributions and water levels. By lowering the pump suction line inlets several inches and installing horizontal gratings above the inlets, the containment water level for vortex free operation was reduced from EL 697 ft (with no blockage) to EL 693.8 ft (with up to 50 percent blockage of the screens or trash racks). (The maximum water level in the containment following a LOCA would be EL 708.5 ft, and the lowest sump elevation is EL 691 ft).

The applicant states that in the long term following onset of a LOCA, when the sump structure is fully submerged, the average flow velocity at the sump screen would be 0.31 fps (assuming maximum safeguards equipment conditions and 50 percent blockage of the screen area); Regulatory Guide 1.82 recommends a design velocity for the coolant at the inner screen of about 0.2 fps. Furthermore, initially, as the RWST water is being discharged to the containment via

the quench spray system and emergency core cooling system, flow velocities on the floor of the containment, as reported in the Aleden Research Laboratory test report submitted by the applicant, would be on the order of 1.7 fps. The applicant further states that the velocity distribution would be such that reflective metallic type insulation (used on most of the piping in the containment) would not be transported to the sump. However, the Alden Research Laboratory reported in NUREG/CR-3616, "Transportation and Screen Blockage Characteristics of Reflective Metallic Insulation Materials", that flow velocities well below 1.7 fps are capable of transporting the various component parts of this type of insulation to the sump structures. In light of this, the staff recommends that the applicant provide a debris generation and transport analysis, which describes the transient behavior of the sump as the water level in the containment is rising, to justify the acceptability of the 50 percent sump blockage assumption throughout the accident. The staff considers this to be a confirmatory item and will report on its resolution in a future supplement to this SER.

The staff has reviewed the applicant's net positive suction head (NPSH) calculation submitted by letter dated February 21, 1984, and the updated results reported in amendment 5 (February 1984) to the FSAR, which reflect the head loss across the modified sump structure. The analysis shows the NPSH available to the recirculation pumps during both the recirculation spray mode and the combined recirculation spray and low head safety injection mode is always greater than the required NPSH. At the beginning of the recirculation spray mode (when the containment water level is low, and conservatively calculated to be at EL 694 ft), the NPSH margin is calculated to be 0.9 ft (assuming minimum ESF operation to achieve a higher flow rate, and 50 percent blockage of the sump. The NPSH margin will continue to increase as the containment water level rises to its maximum level. 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 (see FSAR, Section 6.2.2.3.2) 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, e suring 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 additional 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 requirements of General Design Criteria 38, 39, and 40, the provisions of Regulatory Guide 1.1 on an acceptable alternative basis as defined above, and the provisions of Regulatory Guide 1.82, except as noted above.

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 barrier 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

Safety Injection Signal

a. High Containment Pressure (Hi-1)
b. Low Compensated Steam Line Pressure
c. Pressurizer Low Pressure
c. 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

Cotainment 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 two 42-in. butterfly valves, which are only open during plant cold shutdown and close automatically within 10 seconds upon receipt of a high radiation signal. During normal operation, the containment is not purged. The containment airborne radioactive contaminants are removed by the containment atmosphere filtration system.

(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 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 remotely-controlled 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 discharging 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 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 has also made

provision to assure 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. 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 Standards 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 and the provisions of BTP CSB 6-4. 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 within containment 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 hyrogen 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 onset of an accident. In addition to the two safety related hydrogen recombiners provided, a non-safety grade backup 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 functionally 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.

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, and therefore, is acceptable.

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 responses to NRC Questions, and finds it acceptable with the following exceptions. The applicant proposes to exclude certain valves from Type C testing (including the safety injection system penetrations and recirculation spray system penetrations). The applicant states that the justification for excluding these penetrations from Type C testing is based on the rationale presented in Technical Specification Amendment No. 65 to the operating license for BVPS, Unit 1. Excluding these valves from the Type c testing program was approved by a NRC letter dated March 22, 1983. The staff has examined the subject issue and the bases for approving Amendment No. 65. Since both plants are identical in design, the staff finds the applicant's proposal acceptable.

Based on above discussion the staff concludes that 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.

ENCLOSURE
SALP
prepared by the Containment Systems Branch
Regarding

Evaluation Criteria	Beaver Valley 2 (Docket No.: 50-412)
1. Management Involvement	N/A
2. Approach to Resolution of Technical Issues	Technically sound and thorough approach in most cases. Category 1
3. Responsiveness	Frequently requires extensions of time Category 3
4. Enforcement History	N/A
5. Reportable Events	N/A
6. Staffing	N/A
7. Training	N/A