Duquesne Light Company

Beaver Valley Power Station
P.O. Box 4
Shippingport, PA 15077-0004
(412) 393-5206
(412) 643-8069 FAX

GEORGE S. THOMAS
Division Vice President
Nuclear Services
Nuclear Power Division

March 10, 1995

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555-0001

Subject: Beaver Valley Power Station, Unit No. 1
Docket No. 50-334, License No. DPR-66
Generic Letter 88-20 (TAC No. M747378)

Reference: NRC Letter to Duquesne Light Company (DLC), Beaver Valley Unit No. 1 - Request for Additional Information Concerning Generic Letter 88-20 Individual Plant Examination for Severe Accident Vulnerabilities (TAC No. M74378)

Duquesne Light Company's responses to the NRC's referenced Request for Additional Information (RAI) are attached.

Should you have any questions regarding this submittal, please contact N. R. Tonet at (412) 393-5210.

Sincerely,

George S. Thomas

George S. Thomas

Attachment

cc: Mr. L. W. Rossbach, Sr. Resident Inspector
Mr. T. T. Martin, NRC Region I Administrator
Mr. D. S. Brinkman, Sr. Project Manager
Mr. R. Maiers, Pennsylvania Department of Environmental Resources

140000

A011
1/1

BV1 IPE FRONT-END QUESTIONS

1. Reactor Coolant Pump (RCP) seal Loss Of Coolant Accidents (LOCAs) are a significant contributor to the overall Core Damage Frequency (CDF); however, the submittal does not address the RCP seal LOCA model used in the IPE. Please identify and discuss the application of the RCP seal LOCA model used in the IPE.

Response:

The seal LOCA model is described in Appendix B, Section B.2 of the IPE analysis, which gives the background for the electric power recovery model used in the IPE. This Appendix was not part of the IPE summary report, therefore, a brief synopsis of the RCP seal LOCA model and how it was applied in the IPE is given below.

The model for the pump seal leak rates was based on the four-loop RCP seal LOCA study of Reference B.2-4 for the Westinghouse RCPs with the old style O-rings that existed in the Beaver Valley Unit 1 RCPs at the time of the study, and scaled by the number of loops at Unit 1 to reflect the leak rates per pump. These data were then used to develop the probability leak rate model for the electric power recovery analysis. The specific seal LOCA leak rates, used as a function of time after the loss of seal cooling, are provided in Table B.2-1, copy attached. The flow rates listed in gpm define the effective flow area, assuming an RCS pressure of 2250 psig. The time to core uncovery for a given leak rate, which varies with time, was computed accounting for the decrease in RCS pressure as the accident progresses and includes the effects of the operator action to depressurize the Steam Generators. Reference B.2-4 is as follows:

> NUREG-11560, Report Reactor Coolant Seal LOCA, "Results of Expert Opinions Elicitation on Internal Event Front-End Issues for NUREG-1150: Expert Panel", NUREG/CR-5116, Volume 1, Sandia 88-0642, April 1988.

PLG developed an engineering code, SEALOC (Reference B.2-5), to calculate the time of core uncovery due to a pump seal LOCA during an SBO with the turbine-driven or dedicated Auxiliary Feedwater pump available. SEALOC was used to calculate the time of core uncovery for the various probabilities of seal leak rates shown in Table B.2-1 and the impact of RCS depressurization. A constant leak rate (initiated immediately when all onsite AC power fails and RCP seal cooling is lost) of 21 gpm per pump was used in the electric power recovery analysis as the leak rate for the first hour prior to the severe seal damage.

For the worst case, after the first hour, a 480 gpm leak rate per RCP is expected, given the assumption that the low pressure seal leakoff piping would rupture after failure of the #1 RCP seal (the assumed rupture of the seal leakoff piping is only made to conservatively bound the maximum flow rate through the seals). On the basis of the analyses performed with the SEALOC code, this leak rate would result in core damage (1,200°F) approximately 7 hours after the initiation of the station blackout, if operators took action to depressurize the Steam Generators in 2 hours or less. If the operators depressurize after 2 hours, or completely fail to depressurize at all, core damage would occur approximately 2.9 hours after the station blackout. The results of this analysis was then used in conjunction with other parameters to determine a nonrecovery factor used in the electric power recovery event tree. Reference B.2-5 is as follows:

> Maneke, J. A., D. R. Buttemer, and R. K. Deremer, "Reactor Coolant Pump Seal LOCA Analysis during Station Blackout Events at Seabrook Station", prepared for New Hampshire Yankee, Pickard, Lowe and Garrick, Inc., PLG-0724, January 1990.

Table B.2-1 Seal LOCA Flow Rates (GPM) per Pump with and without Primary Depressurization							
Probability	**Cumulative Probability**	**Time after Station Blackout (hours)**					
		0-1.0 (gpm)	**1.0-1.5 (gpm)**	**1.5-2.5 (gpm)**	**2.5-3.5 (gpm)**	**4.5-5.5 (gpm)**	**5.5 + (gpm)**
0.2712	.2712	21	21	21	21	21	21
0.0151	.2863	21	21	21	61	61	61
0.0161	.3024	21	21	61	61	61	61
0.0181	.3205	21	61	61	61	61	61
0.0120	.3325	21	61	108	108	108	108
0.0059	.3384	21	61	108	108	120	175
0.1120	.4504	21	61	250	250	250	250
0.0136	.4640	21	120	250	250	250	250
0.5302	.9942	21	250	250	250	250	250
0.0016	.9958	21	308	308	308	308	308
0.0042	1.0000	21	480	480	480	480	480

2. It is not clear from the submittal how spray induced failures of equipment were addressed in the internal flooding analysis. Please provide a discussion of the treatment of spray induced failures and, if they were screened out, provide the basis for the screening.

Response:

The IPE did focus on modeling submergence-induced failures of equipment as a result of internal floods. To assess the effects of spray, design basis event analyses for moderate and high energy line breaks were reviewed prior to the plant walkdown for internal flooding. Spray-induced failure modes were then considered during the plant walkdown to identify key scenarios. As a result of the walkdown, spray-induced effects were judged to be localized, so emphasis was placed on effects beyond design basis; namely, larger floods.

Subsequent consideration of spray effects was then limited to inadvertent actuations, and leaks and breaks in the fire suppression systems; i.e., the Fire Water System. In general, inadvertent actuations of the Fire Water System were not considered significant flooding sources because of the relatively low capacity of the sprinklers and because alarms would alert the operators. One possible exception to this that was considered, was the actuation of the sprinklers above the CCR pumps. However, the frequency derived for inadvertent actuation of this portion of the Fire Water System was more than an order of magnitude less than the frequency of losing CCR from all other causes and, therefore, was judged insignificant.

The screening approach to the analysis of internal flooding used in the Beaver Valley Unit 1 IPE is conservative. Spray effects that are localized to a particular compartment are accounted for by the screening assumption that all susceptible items within a location are initially failed. Scenarios that survive the screening are then examined on a case-by-case basis. The resulting flood scenario frequencies that survive the screening have frequencies expected to be as high as spray events and with greater plant impacts.

For example at Beaver Valley Unit 1, the highest frequency flood scenarios retained after screening for quantification occur in the Intake Structure (9.8×10^{-3} per year) and the Turbine Building (7.7×10^{-3} per year). The Intake Structure flood was modeled as failing all normal river water and a raw water pump. The Turbine Building flood was modeled as impacting main feedwater, turbine plant component cooling water, and station instrument air. It is difficult to see how spray-related failures could result in more severe scenarios than these. Experience at this and other plants (e.g., Seabrook) suggests that even when spray effects are systematically reviewed, they are not significant when compared to submergence effects.

3. Please address the following item with respect to the need for containment cooling to support core cooling. The success criteria indicates that containment cooling is required if energy is released into containment; however, the discussion of the event trees implies that containment cooling was modeled only to establish conditions for the back-end analysis. How did the IPE model the loss of containment cooling as impacting the ability to cool the core?

<u>**Response**</u>:

The inside and outside Recirculation Spray pumps provide the containment heat removal function (via the Recirculation Spray coolers) and part of the containment spray function in conjunction with the Quench Spray System. In addition to these functions, the outside Recirculation Spray pumps can also provide vessel injection by diverting partial flow from the Recirculation Spray header to the suction of the HHSI pumps, when the LHSI pumps are unavailable for the SI recirculation mode. The success of the containment spray and heat removal functions are only required for the containment analysis and have no impact on the calculation of the core damage frequency.

For non-LOCA cases, core cooling is provided by the Auxiliary Feedwater, Main Feedwater, or dedicated Feedwater Systems. Core cooling for LOCA cases is provided by the HHSI and LHSI systems which transfer RWST inventory into the core during the SI injection phase and containment sump water into the core during the SI recirculation phase. The IPE modeled the common cause failure of all four Recirculation Spray coolers, due to their River Water supply check valves failing to open, as failing the containment sump (Top Event SM), since this could impact the NPSH requirements for the LHSI and RS pumps. The independent failure of all four Recirculation Spray trains was not modeled as impacting the NPSH to these pumps in the IPE submittal. It was expected that the occurrence of this would be extremely small or that such sequences would have already progressed to core melt. To be certain, however, the PRA was requantified with this dependency modeled. As expected, there was no increase in the CDF. The IPE model also takes credit for a continued SI injection mode if the sump is unavailable, or the sump suction flow path fails during the recirculation phase, by making up to the RWST (modeled in Top Events MU and WM) to prevent core melt. It is assumed that as long as the flow provided to the core, via the HHSI or LHSI systems, matches the boil-off flow rate, core cooling will be provided.

If vessel melt-through occurs, a sustained makeup of 140 gpm would be adequate to cool the core debris once the decay heat level reaches approximately 0.75% of full power, assuming a coolable geometry. Injection of the RWST into the containment provides a source of water for debris cooling if vessel injection during the recirculation mode is available, or if the debris is dispersed from the keyway.

4. The contribution to CDF from Anticipated Transient Without Scram (ATWS) for Unit 1 (20%) is significantly different from Unit 2 (4%). The submittal indicates that top events RT (Reactor Trip) and PA (Primary System Pressure Relief) are significant contributors to CDF from ATWS; however, additional information received by the staff from Duquesne Light Company indicates that the importance of these items is significantly less than previously reported. Please provide a discussion regarding your assessment of this event and its contribution, and

the contributors (including the PORV block valves) to CDF, based on your current understanding.

Response:

As part of an effort to determine the impact of a proposed Technical Specification (which will require two operable PORVs) on plant operations, a review of the ATWS and associated pressure relief models in the IPE has been performed. For the IPE, some assumptions had to be made about the ATWS analysis, which were not documented in WCAP 11993. During this recent review, however, additional information about the WCAP analysis was obtained that indicated that the earlier assumptions for PORV availability were inconsistent with the reference analysis in WCAP 11993, and that these inconsistencies were causing the core damage frequencies for ATWS events to be overly conservative. Therefore, a reevaluation of the ATWS model was recently performed to show what effects this new insight had on core damage frequency. A discussion of this reevaluation is provided in the following paragraphs.

WCAP 11993 was prepared to demonstrate that Westinghouse plants comply with the ATWS target goal listed in SECY-83-293. The results of this analysis are in the form of probabilistic frequencies of ATWS events which lead to overpressurization of the RCS. The analysis uses plant parameters selected to bound all Westinghouse plants for ATWS considerations. The event tree includes system top events for those systems which play an important role in mitigation of an ATWS event. This includes overpressure mitigation capability through the use of PORVs and Pressurizer Safety Valves. The overpressure protection capability is characterized with respect to the variability of the moderator temperature coefficient over the fuel cycle. The number of relief valves required to prevent overpressurization of the RCS is calculated as a function of the time in core life and conversely, the time period during a cycle for which overpressure protection is not adequate for a given number of operable valves is tabulated. The latter is defined as unfavorable exposure time (UET). The UET is then adjusted based on the frequency weighting for transients during a cycle, averaged over the cycle and normalized to a standard period for use in the risk assessment.

From WCAP 11993, for the reference plant, the following UETs were calculated for various PORV availabilities and assumptions for Manual Rod Insertion (MRI) and Auxiliary Feedwater flows (values are in days for an 18 month cycle):

Condition	2 PORVs	1 PORV	0 PORV
With MRI, all Aux Feed	0.0	0.0	76.3
With MRI, half Aux Feed	0.0	18.9	82.6
No MRI, all Aux Feed	81.7	138.9	192.9
No MRI, half Aux Feed	110.7	154.8	209.1

All auxiliary feedwater for the reference plant is assumed to include flow from all pumps, i.e., 2 motor-driven pumps and 1 turbine-driven pump. Half auxiliary feedwater is the capacity of 1 turbine-driven pump or 2 motor-driven pumps. In the WCAP, the UETs listed were further adjusted to account for the frequency of transients as a function of cycle life and normalized to a one year period. This was done to account for the fact that historical data showed that a higher frequency of feedwater-related transients occurred during the early part of a cycle.

In reviewing how this information was applied to the Beaver Valley IPEs, the following observations were made:

Only the UETs for half Aux Feed were used for both half and full Aux Feed cases for each condition of rod insertion. This has the effect of not taking credit for the slight improvement in UET for cases in which all Aux Feed is available. In the current PRA model, success of the Auxiliary Feedwater top event only requires that either the turbine-driven pump or both of the motor-driven pumps be operable. Therefore, no additional mitigating credit is taken for the successful operability of all three Auxiliary Feedwater pumps; i.e., it is treated the same as a failure of the turbine-driven pump or a failure of one or two motor-driven pumps. This is a conservative approach employed primarily for modeling efficiency considerations.

In the IPE, the adjusted UET values from Table B-3 of WCAP 11993 were used directly without normalization or consideration of the effects of transient weighting with respect to initiating frequencies as used in the IPE. In the WCAP, the frequency of overpressure events due to ATWS for an assumed PORV availability condition was calculated by using the adjusted UET values and the failure frequencies for the PORV and safety valves. For any period in which the assumed number of PORVs and 3 safety valves was insufficient, the failure probability for the valves is equal to 1, and the total failure probability is equal to the event frequency for this period. The total failure rate for the cycle is determined by adding the failure rate for the different periods (based on pressure relief requirements) and normalizing to a one year period to arrive at a mean failure rate. In the IPE, the ATWS event tree is integrated into the overall model and is entered upon failure of automatic or manual reactor trip. Initiating event frequencies are annualized and, therefore, no weighting of transients during the cycle is required since all transients will be included for the period. In order for the UET as presented in the WCAP to be consistent with this approach, adjustment of the UET to a one year period is necessary, but transient weighting should not be included. In the current application, the transient weighting was included and no adjustment to a one year period (normalization) was performed. The result is that the model becomes very conservative with respect to the frequency of overpressure failures resulting from ATWS initiators.

A shift of the UET values from the WCAP was performed based on the number of PORVs at Beaver Valley versus the reference plant assumed in the WCAP. The assumption made here was that the total capacity of all PORVs was equivalent regardless of the number of PORVs installed. Therefore, one PORV at Beaver Valley was assumed to have only two-thirds of the relief capacity of one PORV at the reference plant. Based on this assumption, the UET values as presented in the WCAP were shifted such that where one PORV was assumed operable for the reference plant, it was assumed that Beaver Valley required two PORVs. Likewise, the UET values listed for the no PORVs operable case in the WCAP were used to represent the condition where one PORV was operable at Unit 1. Discussions with Westinghouse personnel, and a review of the reference documents from WCAP 11993, have indicated that the individual PORV capacities at Beaver Valley are equivalent to those assumed for the reference plant. Therefore, the adjustment made for valve capacities is not required and represents another conservatism in the analysis.

Direct application of the UET values from WCAP 11993 based on 1 PORV available, half Auxiliary Feedwater, without weighting for transient time, and adjusted to a one year period results in the following:

With manual rod insertion:	5.2% of the time 1 PORV + 3 SRV are insufficient
	17.4% of the time 1 PORV + 3 SRV are required
	77.4% of the time 3 SRV are required
Without manual rod insertion:	42.4% of the time 1 PORV + 3 SRV are insufficient
	14.9% of the time 1 PORV + 3 SRV are required
	42.7% of the time 3 SRV are required

Using these values in the Beaver Valley Unit 1 IPE model results in a reduction of the contribution to the total core damage frequency from ATWS sequences in general, and specifically those associated with PORV block valve alignment. The previously reported contribution for ATWS events for Beaver Valley Unit 1 was 20.1% of the total CDF. Based on the revised fractions as listed above, the contribution is reduced to 6% of the prior total CDF. The contribution previously reported for ATWS events in which failure resulted from inadequate overpressure protection capacity (PORV Block Valve Alignment) was 15.6%. Using the fractions listed above, the contribution is reduced to 2.87% of the previous total CDF.

Previous discussions on resolution of this vulnerability had indicated that proposed amendments to the Technical Specifications, which will require two operable PORV vent paths, would reduce this vulnerability by increasing the number of PORVs available. While increased availability of the PORVs may provide some benefit in terms of ATWS mitigation, this is only true if the block valves remain open. However, the existing Technical Specifications and the proposed changes do not require that a block valve remain open in order for a PORV to be considered operable. Therefore, it has no direct impact on the PRA ATWS model because the model takes no credit for the PORVs unless the operable PORVs' block valves are also open. The revised model will utilize actual plant experience as the basis for PORV availability. The reevaluation which assumes one PORV available with the block valve open demonstrates that with modeling conservatisms reduced, as stated above, the vulnerability is reduced to an acceptable level consistent with the ATWS licensing basis.

5. NUREG-1335 requests human reliability data, the time available for operator recovery and other generic or plant-specific data for important equipment or events. The submittal does not identify the data used to model recovery of offsite power and recovery of failed Diesel Generators (DGs) which are important recovery actions. Please provide this data and the source of the data.

Response:

The electric power recovery model used in the IPE is described in Appendix B, Section B.2., and as stated in response to Front-End Question 1, was not part of the IPE summary submittal. The loss of offsite power data at U.S. nuclear plants reported for all years through 1988 (Reference B.2-1) were the basis for developing the mean probability of nonrecovery of offsite power for PLG's generic database and for use in the Beaver Valley Unit 1 FRA model. Specific line data from the Seabrook site (Reference B.2-2) and South Texas Project site (Reference B.2-3) were used along with data from NUREG/CR-5032 to develop the distributions from the mean for the 10th and 90th percentile cases. These distributions for the probability of nonrecovery used in the electric power recovery model for Beaver Valley are shown in Figure B.2-1, attached. References B.2-1, B.2-2, and B.2-3, respectively, are as follows:

Nuclear Safety Analysis Center, Electric Power Research Institute, Inc., "Losses of Offsite Power at U.S. Nuclear Power Plants All Years through 1988", NSAC-144, April 1989.

Pickard, Lowe, and Garrick, Inc., "STADIC4 Model for Frequency of Nonrecovery of Electric Power at Seabrook Station for Plant at Power and Shutdown", prepared for New Hampshire Yankee, PLG-0507, May 1988.

Pickard, Lowe, and Garrick, Inc., "South Texas Project Probabilistic Safety Assessment", prepared for Houston Lighting & Power Company, PLG-0675, May 1989.

The analysis considers recovery factors for cases in which 0, 1 or 2 Emergency Diesel Generators are available for recovery. The discussions which follow first describes the case when only one Diesel Generator is recoverable, and then describes the case for when two Diesel Generators are recoverable.

The model that was used for a single Diesel Generator recovery (i.e., only one Diesel Generator has DC power available) is:

$$\Phi_1(t + \tau) = \int[\mathit{\Phi}_{OR}(t)][\Phi_{DH}(\tau)]dt \qquad \text{(Equation B.2.4)}$$

where;

$\Phi_1(t + \tau)$ = cumulative frequency of power recovery from a single Diesel Generator when only one diesel is available for recovery

$\mathit{\Phi}_{OR}(t)dt$ = frequency of auxiliary operator response to Diesel Generator Building between t and t + dt after the failure of Diesel Generator power

$\Phi_{DH}(\tau)$ = cumulative frequency of Diesel Generator hardware recovery within time (τ) after operator response

This analysis is performed for conditions when only one diesel is available for recovery; i.e., the other diesel has failed at t = 0 and cannot be recovered within 24 hours. In this analysis, approximately 20% of the single diesel unavailability is assumed to be attributed to pre-existing maintenance scenarios. The 5th percentile model for the single diesel recovery reduces the cumulative frequency of recovery for one diesel by 20%. For the 95th percentile, however, a more optimistic view is taken, and it is assumed that this fraction of unavailability is recoverable. This would include, for example, restoring the diesel to service after minor maintenance or testing. For the 50th percentile, it is assumed that the fraction of unavailability due to maintenance is recoverable after 2 hours.

The 5th percentile of the single diesel recovery model represents a pessimistic model for the operator response and delays the auxiliary operator's time by 30 minutes. The 50th percentile of the model represents a delay of the operator's arrival by 10 minutes. The 95th percentile bound represents a more optimistic model for operator response, and no delay in the auxiliary operator's arrival is included. Figure B.2-2 (attached) presents the complementary cumulative distribution for the Diesel Generator nonrecovery that is derived for these bounding models.

For the case where power can be recovered from either Diesel Generator (i.e., two Diesel Generators are recoverable), successful recovery has been defined for this analysis as the restoration of power from at least one of the two Diesel Generators. This nonrecovery model for one out of two Diesel Generator recovery is characterized by the expression:

$$\Phi_{1/2}(t + \tau) = \Phi_1(t + \tau) + [1 - \Phi_1(t + \tau)][\Phi_1(t + \tau - 0.5)] \qquad \text{(Equation B.2.5)}$$

This model allows recovery of the first of the two Diesel Generators to begin when an auxiliary operator arrives at the Diesel Generator Building. Recovery of the second diesel begins 30 minutes after the auxiliary operator arrives, and the repairs of both diesels are modeled as continuing in parallel thereafter.

Two bounding scenarios are applied as the 5th and 95th percentiles for the Diesel Generator recovery model. For the 5th percentile bound, the single Diesel Generator recovery model (above Equation B.2.4) is used. This model represents a pessimistic model for operator response, and it allows recovery of power from only one Diesel Generator. Parallel repairs of the second Diesel Generator are not considered. This bound accounts for possible unidentified dependencies in the recovery efforts for both Diesel Generators, which could couple the repair time distributions; e.g., limited spare parts availability, limited support personnel availability, etc. For the 95th percentile bound, the dual Diesel Generator recovery model (Equation B.2.5) is used. The recovery of the second diesel begins 30 minutes after the operator arrives, and the repairs of both diesels are modeled as continuing in parallel thereafter. The 95th percentile bound thus represents a more optimistic assessment of operator response, and it includes a more realistic model for single and parallel Diesel Generator repairs. The 50th percentile is estimated from the 5th and 95th percentile curves. Figure B.2-3 (attached) presents the cumulative distributions for the Diesel Generator nonrecovery derived from these bounding models.

It should be mentioned that the three curves for each quantity (i.e., the 5th, 50th and 95th) are weighted in the calculation as 0.1, 0.8 and 0.1, respectively. The above terms for recovery of offsite power and for the recoverable Diesel Generators are used in the electric power recovery Equation 3.3.3.4 in Section 3.3.3.4.1 of the IPE submittal.

NOTES:

1. 50th PERCENTILE FROM NSAC-144 DATA.
2. 10th AND 90th PERCENTILE FROM ONE OR THREE LINE DATA FROM SOUTH TEXAS, SEABROOK, AND NUREG/CR-5032.

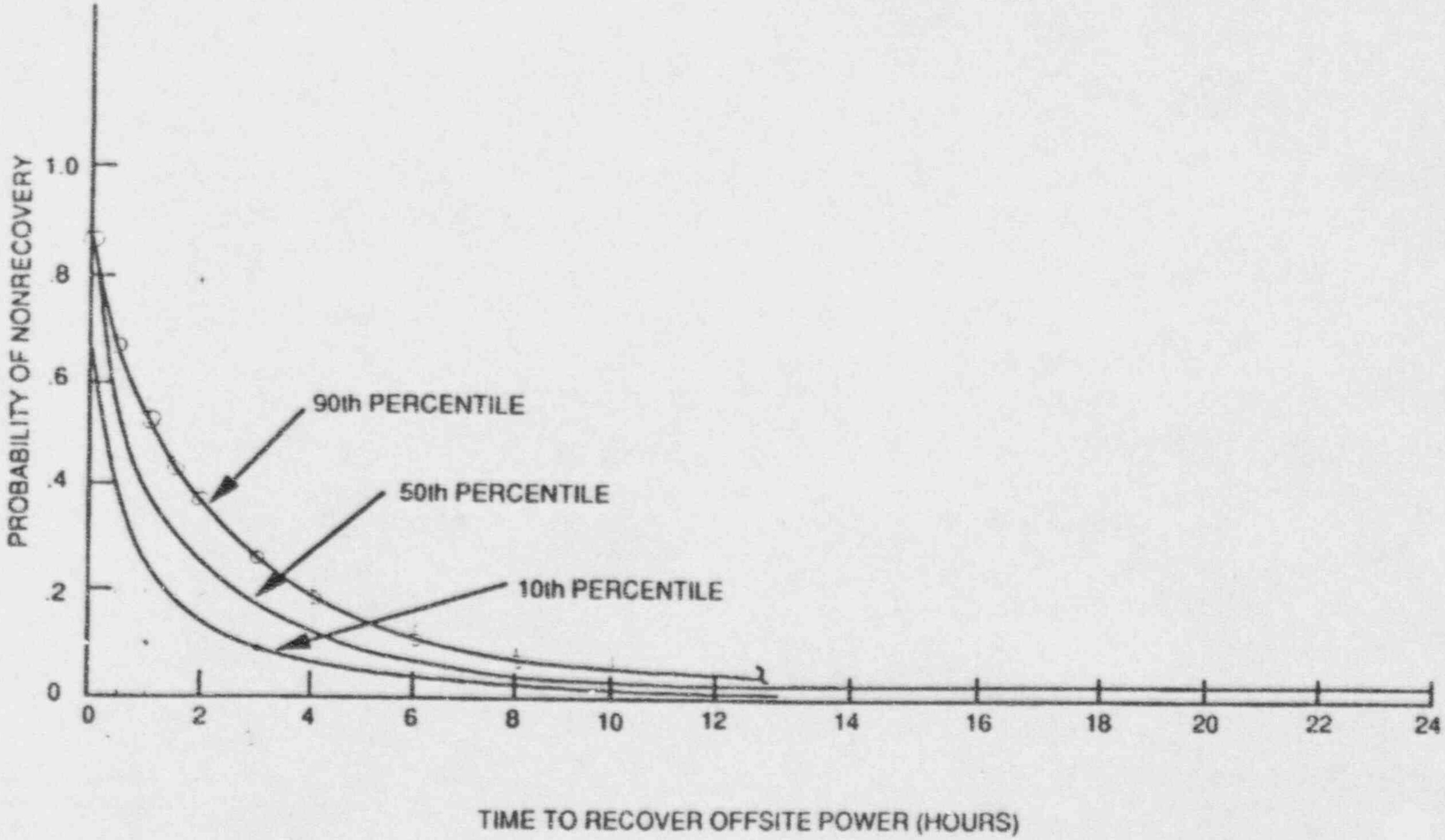


Figure B.2-1. Probability of Nonrecovery of Offsite Power

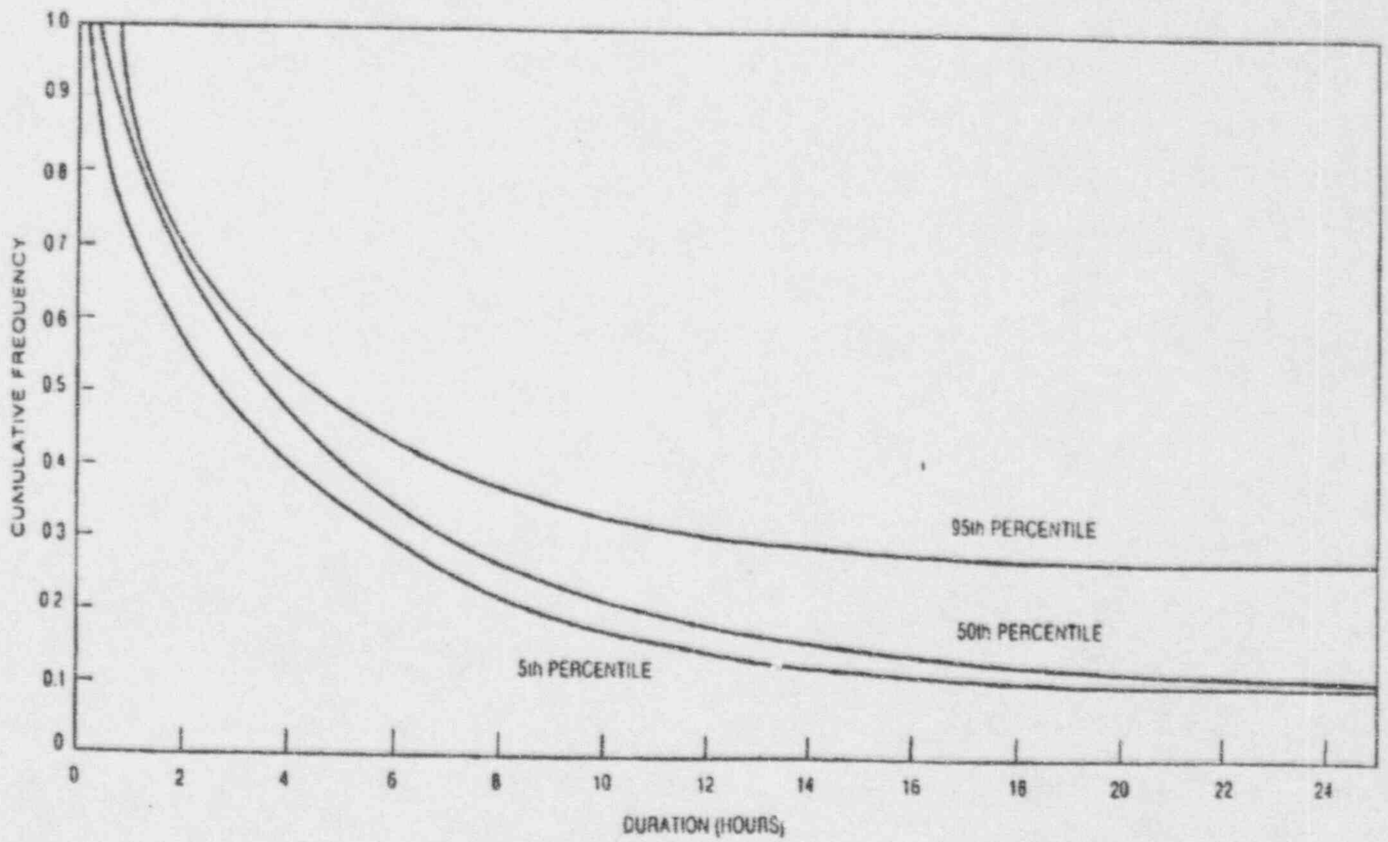


Figure B.2-2. Nonrecovery of Single Diesel Generator

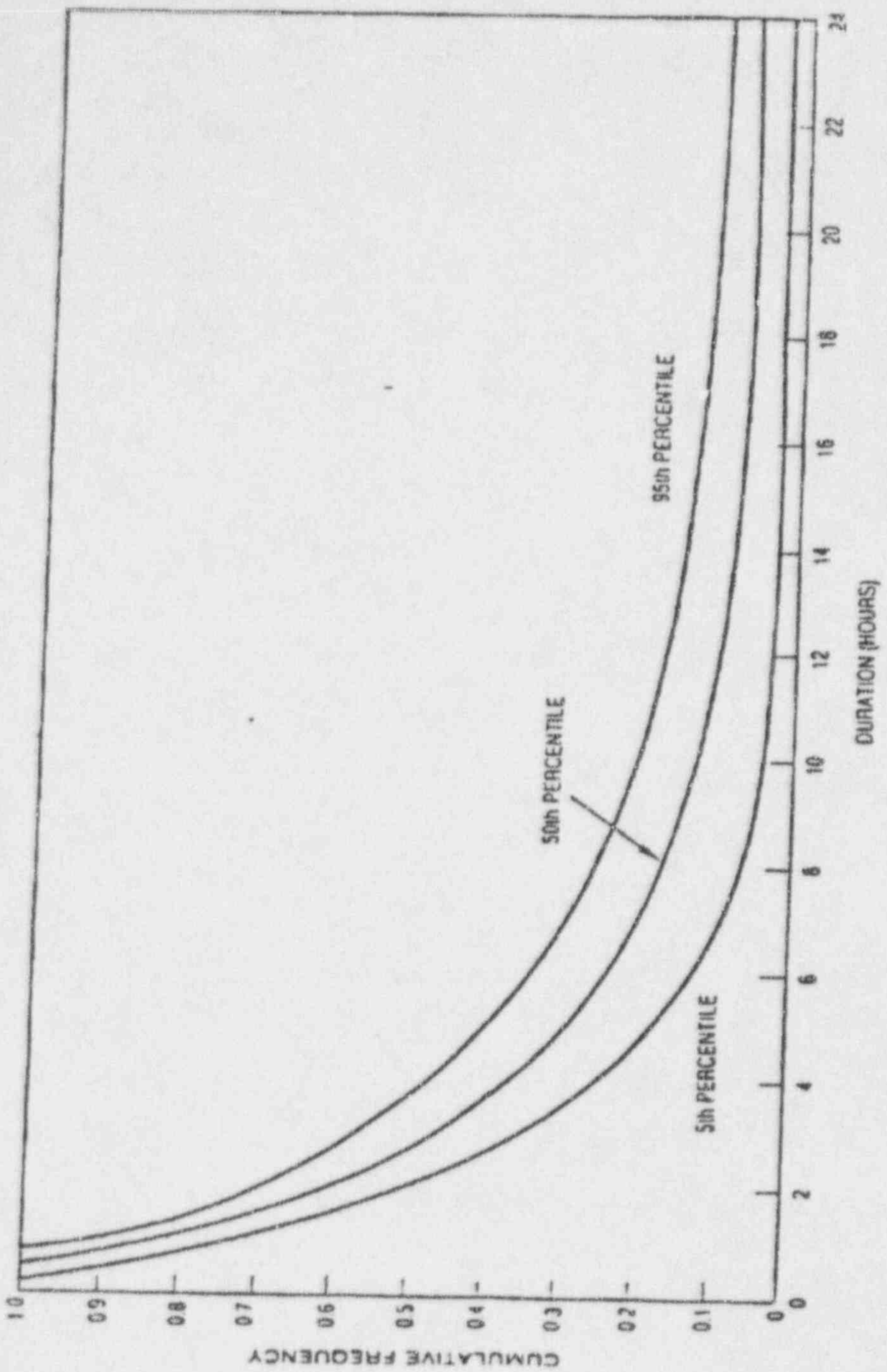


Figure B.2-3. Nonrecovery of One Out of Two Diesel Generators

14 of 69

6. Duquesne Light Company has identified a number of vulnerabilities for BV-1 in the submittal, and their importance to CDF. However, it is not clear to what extent the potential enhancements have been incorporated in the plant, nor what reduction in CDF is expected for implementation of the enhancements. As requested by NUREG-1335, please identify the status of the implementation of the proposed enhancements and the reduction in CDF for each enhancement, or the total reduction in CDF, for those enhancements expected to be implemented. In addition, please provide a discussion for those identified potential enhancements not being considered for implementation, and the basis for the decision.

Response:

The status of the implementation of the proposed enhancements for the vulnerabilities identified in the Beaver Valley Unit 1 IPE submittal is addressed in the following paragraphs. The reduction in CDF from the implementation of these enhancements was not calculated for each item; rather a total reduction in CDF was calculated for all of the changes made including implementation of vulnerability enhancements, slight changes to the top event models to reflect plant modifications performed through 1993, and plant-specific data updates of component failures and maintenance through June 1993. Model changes associated with the vulnerability enhancements only reflect installation of the 4160V station crosstie and revision of the primary pressure relief top event for ATWS events as stated in the response to Front-End Question 4. The new total CDF as a result of all changes to the PRA model is 1.20×10^{-4} per reactor year (using the same initiating event frequencies as reported in the IPE). This represents an approximate 44% reduction in core damage frequency.

Loss of Emergency Switchgear Room HVAC

This specific sequence results from a loss of both normal and emergency cooling to the emergency switchgear area which could lead to equipment damage in these areas and subsequent loss of power to emergency equipment. Although credit was taken for restoration, since operators are aware of the potential results of losing both trains of cooling, the previous alarm response procedures did not provide specific guidance for mitigating the consequences of this event through the use of portable ventilation. More specific response procedures have been developed to provide temporary ventilation for the emergency switchgear areas through the use of portable fans. However, the human reliability analysis for ventilation restoration was not revised to account for the procedure enhancements and, therefore, the CDF was not affected.

Fast 4160V Bus Transfer Failure

The specific sequence results in failure of the 4160V fast bus transfer and failures of the diesel generators which would lead to a station blackout condition. Recovery of electrical power through repair of the fast bus transfer breaker was identified as one method of mitigating the consequences of this event. A review of the existing procedures indicates that ECA 0.0 provides direction to the operator to transfer power manually from offsite sources in the event of a failure and procedure 1/2.36.4A provides specific direction for racking breakers in and out. It was also noted that the significance of this vulnerability is lessened somewhat by the installation of the 4160V station crosstie capability (see AC Power Generation Capability below). This procedure enhancement was identified as having been implemented at the time of IPE submittal and, therefore, no changes to the model were required.

Battery Capacity for Steam Generator Level Instruments

This item is considered resolved as a result of installation of the 4160V station crosstie since one train of the emergency battery chargers will be powered from this source. Therefore, the enhanced procedures on shedding battery loads is not necessary. The reduction in CDF associated with this vulnerability is realized by the installation of the station crosstie.

Reactor Trip Breaker Failure

The specific sequences for this event are those that lead to an ATWS followed by opening of PORVs. Procedures direct the operator to manually drive in the control rods and to manually disconnect the Control Rod M-G sets following failure of the reactor trip breakers, however, this must be done locally and there may be inadequate time to prevent overpressurization of the RCS. With the changes made to the PORV ATWS model (see response to Front-End Question 4, above), the contribution to core damage frequency due to ATWS events is significantly reduced. Based on this, the installation of capability to remove power from the control rods from the control room is not considered warranted.

AC Power Generation Capability

For station blackout sequences, both onsite and offsite recovery actions to reestablish power to 4160V emergency AC electrical buses are important. Installation of the station crosstie connecting the 4 kV normal buses of Beaver Valley Unit 1 and Beaver Valley Unit 2 is now complete. The PRA model was revised to reflect this modification, which now takes credit for the Unit 2 emergency diesel generators, if both are available, given the failure of both Unit 1 emergency diesel generators and the loss of offsite power.

RCP Seal Cooling for Station Blackout

No model changes to specifically address reductions in Seal LOCAs were included; however, the CDF associated with Seal LOCAs has been greatly reduced by installation of the 4160V station crosstie. Seal injection can be provided within 1 hour of a station blackout using the 4160V crosstie. Additionally, new RCP seal materials will be installed on a replacement basis as stock of current spares is expended. It is expected that operating with the new seals will greatly extend the time available for recovery during station blackouts. Duquesne Light will evaluate any modifications required by any future NRC rulemaking with regard to RCP seal cooling.

Pressurizer PORV Sticking Open After Loss of Offsite Power

Most Westinghouse plants do not experience a PORV challenge following a loss of offsite power or a loss of load. The IPE submittal assumed Beaver Valley would experience a PORV challenge because of its 100% load rejection capability, which was assumed to eventually be unsuccessful. This assumption was based on RETRAN analyses for the Diablo Canyon plant, which also has a 100% load rejection capability. Therefore, the loss of offsite power sequences consist of an unsuccessful 100% load rejection which results in a delayed reactor trip and a challenge to a PORV. If the PORV sticks open and a loss of onsite power occurs, a small LOCA will result which greatly shortens the time available for electric power recovery. This is also true for the loss of load sequences if a fast bus transfer failure occurs. An evaluation of the events and specific failures which result in PORV challenges with a loss of isolation capability was performed. Two initiating events were identified as affecting the 100% load rejection capability, a loss of both 345 kV and 138 kV lines (initiating event LOSP), and a loss of only the 345 kV line (initiating event TT).

Table 5.3-2 "Beaver Valley Unit 1 Potential Enhancements" lists the pressurizer PORV sticking open after a loss of offsite power as a vulnerability with a 2.0% contribution to core damage. By taking credit for the station electric power crosstie, further reductions in this vulnerability's contribution to the core damage frequency were gained. This is due to an AC electric power train being available to enable the operator to close the stuck open PORVs' block valve. With the revised PRA model, these sequences now account for approximately 0.4% of the new CDF and, therefore, are no longer considered a vulnerability.

A pressurizer PORV sticking open after a loss of load (i.e., turbine trip) initiating event contributes less than 0.1% to the core damage frequency, and is not considered a plant vulnerability. Therefore, the design enhancement to eliminate a PORV challenge by defeating the 100% load rejection capability is not necessary.

Additionally, it was identified that Table 3.1.1-2, on page 3.1-10 of the IPE submittal has some formatting errors associated with the "Offsite Grid" subsystems. The following table reflects the correct formatting for the Offsite Grid subsystems.

System/Subsystem	Impact on Safety System(s) or Key Plant Equipment	Initiating Event Category/Code Designator	Comment
Offsite Grid 345-kV Line	Turbine Trip Reactor Coolant Pumps (RCP) Main Feedwater (MFW) Condensate Turbine Plant Component Cooling Water Reactor Trip	9/TT	Results in turbine/generator trip but equipment listed is repowered when fast transfer to 138-kV line is complete.
138-kV Line	None	--	Does not cause a plant trip.
Both 345 and 138-kV Lines	RCPs MFW Condensate Turbine Plant Component Cooling Water	16/LOSP	Results in plant trip. Equipment listed is unavailable. Equipment normally operating and powered from emer. buses must restart.

Pressurizer PORV Block Valve Alignment

See the response to Front-End Question 4, above.

Containment

The containment building issues were discussed and concluded as being potential actions to be included as part of a severe accident management program. These will be investigated further as the Duquesne Light Company severe accident management program progresses. The revised PRA model did not include any changes to the Level 2 analysis.

Operator Actions for SGTR Events

It was originally identified that during a SGTR in which all HHSI fails, the procedures directed the operators to depressurize the RCS but only after the sequence has progressed to extreme conditions in which partial core uncovery has occurred. However, earlier depressurization under these conditions may have prevented core damage and terminated releases earlier, and therefore would be beneficial. Upon further review of the Emergency Operating Procedures (EOPs) and discussions with procedure writers, the following was concluded:

> Operator responses to a SGTR are governed by EOPs E-0 and E-3. These procedures do not provide specific guidance for conditions where a failure of all HHSI has occurred but rely on the status trees for core cooling to provide criteria for exit into the function restoration procedures. The criteria in the status tree rely on core exit thermocouple temperatures and RVLIS indications

to identify conditions which are indicative of core cooling deficiencies. During a SGTR it is not anticipated that these conditions would occur prior to accomplishment of depressurization as required by E-3. Therefore, as long as the existing procedures were followed in parallel with efforts to restore HHSI, if it were recognized to be unavailable, the desired actions to depressurize the RCS would be accomplished. Therefore, no procedure changes are required; however, operators should be trained to continue following E-3 even if HHSI completely fails.

Also for SGTRs it was identified that under certain circumstances a stuck open safety relief valve on a steam generator may be locally gagged to isolate the ruptured steam generator. It was determined that this type of action, if needed, would have to be evaluated based on existing conditions which would likely be directed from the Technical Support Center of the Emergency Response Organization. Therefore, guidance on this type of action will be included in the severe accident management guidelines.

No change to the human error rate assigned to these actions was made in the revision to the PRA model.

7. Flooding at the Intake Structure is identified as 36% of the CDF due to flooding at Unit 1. Beaver Valley Units 1 and 2 share a common Intake Structure and, therefore, may be subject to a dual unit initiating event. Please discuss possible flooding events at the Intake Structure which may impact both units, and the ability to cross-tie or share equipment which would normally be credited in the analysis.

Response:

The Beaver Valley Intake Structure houses the Unit 1 River Water pumps and raw water pumps, the Unit 2 Service Water pumps, the fire water pumps that supply both units, and some emergency Motor Control Centers (MCCs). The pumps and the MCCs are contained in cubicles with normally closed security doors that open into the cubicles and have no gaskets on the bottom. The following table summarizes the content of the intake structure pump cubicles:

CUBICLE A	**CUBICLE C**
Unit 1 River Water Pump (WR-P-1A)	Unit 1 River Water Pump (WR-P-1C)
Unit 1 Raw Water Pump (WR-P-6A)	Unit 2 Service Water Pump (2SWS-P21B)
Motor Driven Fire Water Pump (FW-P-1)	Unit 2 Emergency MCC (MCC-2-E02)
Unit 1 Emergency MCC (MCC-1-E1)	**CUBICLE D**
CUBICLE B	Unit 2 Service Water Pump (2SWS-P21A)
Unit 1 River Water Pump (WR-P-1B)	Unit 1 Raw Water Pump (WR-P-6B)
Unit 2 Service Water Pump (2SWS-P21C)	Diesel Driven Fire Water Pump (FW-P-2)
Unit 1 Emergency MCC (MCC-1-E2)	Unit 2 Emergency MCC (MCC-2-E01)

Each cubicle contains a sump with a level switch and sump pump. A fire door connects Cubicle A with Cubicle B, and another fire door connects Cubicle C with Cubicle D. These doors do not have any gaskets, so water is assumed to propagate only between Cubicles A and B, or between Cubicles C and D. No credit is taken for propagation outside a cubicle, although the main open floor area has sufficient grating that would allow drainage back to the river. Additionally, no credit for operator actions to isolate the break prior to pump failures is taken.

The Unit 1 IPE Intake Structure flood modeled a large flood from the river water, raw water, fire water, or service water piping in Cubicle A or B as failing the normal river water pumps and associated valves, and the "A" train of the Raw Water System (the Fire Water System is not modeled in either unit's IPE). No credit was taken for the third River Water pump located in Cubicle C, since it is assumed to be in maintenance. Moreover, the discharge valves associated with WR-P-1C are powered from the MCCs located in Cubicles A and C, which are assumed to be lost due to the flood. Failure of the normal River Water System challenges the Auxiliary River Water pumps located in the alternate Intake Structure. An 18 inch crosstie to the Unit 2 Service Water piping exists on the River Water "A" header, and a 12 inch crosstie to the raw water piping exists on the River Water "B" header, however, no credit was taken for operators to locally open these manual crosstie valves. Likewise, the Unit 2 IPE Intake Structure flood modeled a large flood from the service water, river water, raw water, or fire water piping, but only this time in Cubicle C or D, as failing the normal Service Water pumps and associated valves, thus challenging the alternate Service Water pumps. Once again, no credit was taken for operators locally opening the river water crosstie manual valve, nor for the third service water pump (assumed to be in maintenance).

Therefore, even though the Intake Structure is a common structure shared by both Beaver Valley units, the impact of a flood in Cubicle A or B is not significant to Unit 2, and the impact of a flood in Cubicle C or D is not significant to Unit 1. So while the same flood could impact both units, the degree of impact is quite different. Also, for both units to proceed to core damage, independent and redundant train failures of auxiliary river water and alternate service water must occur. These alternate systems are in a separate intake structure from where the flood is postulated to originate, therefore, the possibility of a significant dual unit initiating event is very remote.

BV1 IPE BACK-END QUESTIONS

1. In the discussion of Top Event 18, Page 4.6-17, you state that the large majority of hydrogen-burn failures are due to detonations as compared to deflagrations. It is also clear from Page 4.2-6 that large amounts of hydrogen can enter the Containment atmosphere. Thus, it is not clear why you state that understanding the uncertainties associated with hydrogen mixing, transport and detonations are "...beyond the scope of this submittal." (Page 4.6-17) Please explain.

Response:

The statement on Page 4.6-17 states that, "the uncertainties associated with predicting hydrogen mixing and transport and the magnitude of dynamic loads associated with detonations are beyond the scope of this submittal." Because the Beaver Valley and Surry plants are essentially "sister" plants, the BV1 Backend analysis relied heavily on the insights obtained from the analyses performed for Surry for NUREG-1150 (Ref. 1). Accordingly, detailed evaluations of hydrogen mixing and transport in the containment were not performed, nor was any pressure load analysis (static and dynamic) performed for BV1. The BV1 analysis did consider the possibility of locally high concentrations of hydrogen and the possibility for Deflagration to Detonation Transition (DDT). Based on the simplified model, a large containment failure was predicted to occur when the underlying conditions for DDT were predicted to occur. The approach used for DDT is discussed in the paragraphs which follow.

As noted in the discussion for Top Event 11 - Containment Failure Prior to Vessel Breach on Page 4.6-14 of the BV1 IPE submittal, the potential for containment failure prior to vessel breach, due to hydrogen burns, was discounted for Surry (Ref. 1) on the bases of the containment strength and the amounts of hydrogen that are produced prior to vessel breach. This contention was accepted for BV1 except for the case when a large amount of hydrogen is generated in-vessel, and the hydrogen is suddenly released (i.e., a thermally induced hot leg failure) into a non-inerted containment (i.e., containment sprays operating prior to piping failure). However, to minimize the number of CET top events related to burns, containment failures related to induced piping failures were addressed at Top Events HE, CE, and LE rather than as a containment failure prior to vessel breach.

There is no specific statement in the discussion for Top Event 18 that states, "that the large majority of hydrogen-burn failures are due to detonations as compared to deflagrations." It is stated that for the case of interest (sprays operating, large quantities of hydrogen generated in-vessel, no burns prior to or at vessel breach), the mean probability of containment failure of 0.4 (given that a burn occurs) is dominated by the <u>assumption</u> that any burn that occurs at a hydrogen concentration greater than 12% propagates to a detonation. It should be noted that a value of 0.38 (see conditional split fraction CEF) was used in the Beaver Valley Unit 1 IPE study. This value corresponds to the probability that the quantity of hydrogen generated in-vessel exceeds that required to achieve a global hydrogen concentration of 12% in the containment based on dry air.

MAAP analyses performed for Beaver Valley indicated that for most severe accident scenarios, burns would either be precluded by steam inertion or would occur when the hydrogen concentrations reached global flammability levels as determined by the MAAP algorithm. An exception to this observation is when a large amount of hydrogen is suddenly released into a non-inerted containment. As noted on Page 4.6-17 of the BV1 IPE submittal, only thermally induced hot leg failures were assumed in the evaluation. As noted in the discussion for conditional split fraction CEC in Table 4.6-4, hydrogen is expected to burn at "global" concentrations below 12% (i.e., burning in the cavity and/or local burning as the flammable gas leaves the cavity) during pour type vessel failures at RCS pressures greater than 200 psia and sprays in operation.

The burn pressures calculated by MAAP were significantly less than those which would result in a significant probability of containment failure. Hand calculations for adiabatic burns (deflagrations) up to the concentration limits for detonation indicated that the pressure rises associated with "real" (non-adiabatic) deflagrations were not likely to result in containment failure.

The dynamic loads associated with detonations are difficult to calculate and containment strength criteria for these types of loads were not available. Accordingly, the BV1 IPE adopted what was believed to be a conservative treatment for detonations and consequent containment failure. It should be noted that NUREG/CR-4551 did not address detonations for Surry.

As noted on Page 4.2-3 of the IPE submittal, the analysis of hydrogen combustion for the Surry plant for NUREG-1150 (Ref. 1) assumed that if electrical power were available during the period of hydrogen generation, "the sprays will keep the steam concentration low and sparks from electrical equipment will cause ignition near the lower deflagrable limit", preventing significant concentrations of hydrogen. Based on the extent of mixing promoted by spray operation and the relatively low ignition energy levels required for ignition, this argument was assumed to be valid for BV1 as well, except for the sudden release of hydrogen into the containment (e.g., vessel blowdown at high pressure after severe core degradation).

The peak pressures associated with detonations are well above the quasi-static pressures associated with deflagrations. However, the energies required for detonation are many orders of magnitude above those required for deflagration. As noted in Reference 2, detonation initiation within a range of hydrogen concentration from 18 to 59 volume percent (the approximate range of hydrogen detonability) requires an energetic ignition source, severe confinement, and/or a sufficiently large volume of gas mixture. Reference 2 concluded that, "the energy levels required to directly initiate detonation are orders of magnitude greater than those necessary to initiate burning at the same hydrogen concentration", and that "a de facto transition to detonation is highly unlikely in reactor containment buildings particularly when there are high steam concentrations or hydrogen concentrations below about 18 volume percent". Minimum ignition energies of 4100 joules have been reported (Ref. 2) for hydrogen-air mixtures. According to Reference 2, this energy level is several orders of magnitude higher than would be produced from an electrical spark caused by contact arcing or by electrostatic discharge, and approximately eight orders of magnitude higher than the minimum ignition energy required to initiate deflagration.

If electrical power is not available, the containment sprays will not operate, and the containment is likely to be inerted by high concentrations of steam. When steam inertion prevents combustion, the recovery of electrical power and containment sprays becomes a concern, since operation of the sprays will condense the steam and drive the gas mixture towards the flammability range. The Surry analyses performed for NUREG 1150 assumed that hydrogen would be ignited and burned as soon as the gas mixture entered the flammability range, guaranteeing that the burn would occur at low hydrogen concentration. Operation of the containment spray would guarantee substantial mixing.

The recovery of AC power during or after core degradation was not addressed in the IPE submittal. Because of potential deleterious effects (such as containment deinerting) the strategy for recovery of mitigating systems such as containment sprays will be carefully examined and fully evaluated in the context of an accident management program.

As noted earlier, for scenarios in which the containment sprays are operating, it is likely that hydrogen burns will occur at low concentrations when hydrogen is "slowly" released into the containment. Only when the hydrogen is suddenly released into the containment (e.g., due to an induced failure of the hot leg or at vessel breach), will the hydrogen concentrations achieve significant values. When vessel breach is accompanied by a High Pressure Melt Ejection (HPME), the containment loads discussed for Top Event C2 include the contribution of hydrogen burns. However, for "pour" type vessel breaches at high pressure, there could be a sudden release of hydrogen into the reactor cavity and then into the containment. Pour type failures are unlikely for high pressure. Nevertheless, such events were addressed at Top Events HE, CE, and LE. However, as noted earlier, containment failure due to burns following pour type melts was deemed to be unlikely.

For those scenarios in which there was a sudden release of a large quantity of hydrogen into a non-steam inerted containment atmosphere following an induced hot leg piping failure, it was assumed that if the global concentration exceeded 12%, a burn would occur which would, in turn, fail the containment. The logic implicit in this assumption is as follows:

1. A deflagration at a 12% hydrogen concentration is not likely to fail the BV-1 containment (based on peak containment pressures determined using the adiabatic burn assumption).

2. Although MAAP simulations showed that the containment was relatively well mixed when sprays were in operation, it was assumed that local concentrations could be 20% higher than the global concentration.

3. Although the BV-1 containment configuration is not necessarily amenable to a deflagration to detonation transition (DDT), it was assumed that a DDT would occur if local concentrations exceeded a value of 15% (minimum value reported in Ref. 3).

4. It was assumed that a DDT would result in a large containment failure.

Figure 4.2-1 of the BV1 IPE submittal (based on the in-vessel hydrogen generation distributions reported in Volume 2 of Ref. 1), was used to determine the probability that the amount of hydrogen generated in-vessel would exceed a level necessary to produce a global concentration of 12%. This probability was estimated to be 0.38. and was used as the split fraction value for Top Events C2 and CE when vessel blowdown occurred at high pressure in the absence of HPME.

The references used in this response are as follows:

1. Breeding, R.J., et al, "Evaluation of Severe Accident Risks: Surry Unit 1", NUREG/CR-4551 (SAND89-1309), Volume 3, Revision 1, Parts 1 and 2, October 1990

2. IDCOR, "Hydrogen Combustion in Reactor Containment Buildings", Technical Report 12.3, September 1983

3. Sherman, M.P., et al, "FLAME Facility...The Effect of Obstacles and Transverse Venting on Flame Acceleration and Transition to Detonation for Hydrogen-Air Mixtures at Large Scale", NUREG/CR-5275, April 1989

2. For split fraction CEC (Page 4.6-38), the failure fraction assigned is 0.0. Why is Containment integrity ensured for all "pour type" failures when such large amounts of hydrogen are released into the Containment?

Response:

In the analysis of severe accidents performed for Surry (NUREG/CR-4551, Vol. 3, Rev. 1, Part 2, Page A.1.1-61) by Sandia National Laboratories, it is concluded for sequences in which debris pours out of the reactor vessel at breach, "there may be some local burns in the cavity but a general deflagration is not expected". MAAP analyses for Beaver Valley indicated that burns would either occur in the cavity or as the hydrogen entered the lower compartment (i.e., "jet" burning). In either case, the associated pressure rise within 4 hours of vessel breach (the break point for early failures) is insufficient to challenge the containment. The Sandia conclusion and the MAAP results provided the basis for the Beaver Valley assumptions.

As a project objective, it was decided to take full advantage of the physical similarities between Beaver Valley Unit 1 and Surry, and deviate from the NUREG/CR-4551 analyses only when warranted because of significant physical differences. This approach was believed to be entirely consistent with the IPE requirements identified in Generic Letter 88-20 and the guidance provided in NUREG-1335.

3. Regarding the type of concrete in the basemat, you state that it will behave like the "siliceous" concrete of Surry, i.e., produce little CO and H_2 during Core-Concrete Interactions (CCI) (Page 4.5-9), even though the concrete has five (5) times the amount of CO_2 and two (2) times the amount of H_2O that typical "basaltic" concrete contains. How would the results be impacted because of higher gas formation from CCI?

Response:

There is no statement on page 4.5-9 or anywhere else in the Beaver Valley Unit 1 IPE submittal that suggests that the quantities of CO and H_2 produced during core-concrete interactions are small. In fact, in the discussion for Top Event 21 - Late Burn of Combustible Gases (H3) which begins on page 4.6-18 of the submittal, it is stated that "if the debris remains in the reactor cavity, large amounts of hydrogen and carbon monoxide can be generated." It is also not clear why the reviewer makes the comparison to basaltic concrete, since no use was made of any data for that type of concrete.

In the discussion for Top Event 21, it is noted that MAAP calculations for a fast station blackout (loss of all ac power and feedwater) sequence with a large RCP seal LOCA (which minimizes the amount of water in the vessel at the time of vessel breach and hence the amount ex-vessel debris cooling) indicate that large amounts of hydrogen are produced by the CCI and the hydrogen burned as it left the cavity in the form of an annular jet and entered the lower compartment of the containment. This combustion was observed in the MAAP results to continue until the oxygen in the containment became sufficiently depleted and to deposit large quantities of energy to the containment atmosphere, increasing its temperature and pressure dramatically. At the time of the evaluation, this burning process appeared to be independent of whether or not the receiver compartment (i.e., the lower compartment) was inerted. There was uncertainty at the time the analyses were performed relative to the ability of MAAP to model this phenomena. The Surry analysis assumed that if containment sprays were available, the hydrogen emanating from the cavity would burn as soon as flammable concentrations were achieved and many small burns would occur but these would not challenge the containment. On the other hand, the Surry analysis assumed that if sprays were not available, the containment atmosphere would be inerted, precluding burns. However, as noted on page 4.6-19 of the Beaver Valley Unit 1 IPE submittal, <u>it was assumed</u> that if the debris is not being cooled, there is a 50% chance that there will be a hydrogen burn at this time (at Top Event 21), regardless of whether or not the lower compartment was inerted. Furthermore, it was assumed that containment failure was guaranteed, given this burn, if containment heat removal was not available.

In the analysis of long term containment behavior during severe accidents which was performed for Surry (NUREG/CR-4551, Vol. 3, Rev. 1, Part 2, Page A.1.1-85) by Sandia National Laboratories, it was noted that "*eventual OP* (overpressure) *of the Surry containment due only to the noncondensible gases generated by CCI* (core-concrete interaction) *is not credible.*" This reference also states that "*the concrete forming the Surry containment is siliceous .... so the amount of noncondensible gases produced by CCI will be much smaller than if it were composed of limestone concrete.*" Furthermore, "*the amount of concrete that would have to be decomposed to overpressure the Surry containment is such that the CCI would penetrate the basemat before enough gas was generated.*" As noted in page 4.5-9 of the IPE submittal, based on an evaluation of the petrographic reports for the plant, the Beaver Valley Unit 1 concrete was also classified as siliceous and it was "concluded that the observations for Surry regarding the minimal contributions of noncondensible gases to containment overpressure are valid for Beaver Valley Unit 1 as well."

As noted on page 4.6-20 of the IPE submittal, there is somewhat of a race between long term overpressure failure modes and the basemat melt-through mode. Sequences involved in either type of failure were binned to Release Category Group III (Late Containment Failures) which, keeping in mind that no credit for system recovery was taken after the inception of core damage, amounted to 43% of the CDF. In the Beaver Valley Unit 1 IPE, it was assumed that basemat melt-through would eventually occur (CET Top Event 26 [BI]) if the debris was not cooled ex-vessel. It was also assumed that in the absence of long-term containment heat removal, containment overpressure failure was assumed to occur first (CET Top Event 24 [C4]), independent of basemat attack. Thus, the only Release Category Group III sequences which are in question are those in which long term containment heat removal is available but concrete attack could not be prevented and eventual basemat melt-through occurs. Since containment heat removal is available, the only mechanism for overpressurizing the containment is the production and accumulation of noncondensible gases. As noted in the Surry evaluation, overpressurization solely by this mechanism alone is not credible.

4. In describing the high-level simplification of the cavity wet/dry issue in defining release categories, the following is noted on Page 4.7-1 of the submittal:

"The last case is conditional on the loss of high head ECCS injection, or recirculation with vessel pressure remaining high, or loss of high and low head ECCS injection with the operators failing to go into "early" ECCS recirculation (i.e., with water still in the RWST): in other words, successful Quench and Recirculation Spray operation and lineup for vessel injection, but with ECCS failure."

Based on your last phrase, "successful Quench and Recirculation Spray operation and lineup for vessel injection, but with ECCS failure", it is not clear what system failures and operators actions are involved in this case. Please explain this accident scenario in more detail as it relates to the source term calculation.

Response:

Table 4.7-1 notes that a "wet" cavity provides a means of cooling (the debris) or at least attenuating the fission product release from the core debris. It should be noted, however, that for the BV1 source term analysis, the reference to cavity wet or dry was dropped and all source term cases were conservatively treated as if the cavity were dry (see p.4.7-2). Therefore, no credit was given for the scenarios in question.

As noted in Section 4.3.3.1, because of the BV1 reactor cavity/instrument tunnel (keyway) configuration relative to the containment, it is impossible for water in the containment to 'spill' into the keyway. As indicated in Section 4.3.3.1, during operation of the QS, a small amount of water (140 gpm if both QS pumps are operating) is diverted to the reactor cavity. If the QS pumps are the only pumps taking suction from the RWST, the maximum quantity of water directed to the reactor cavity prior to emptying the RWST is approximately 12,000 gal. The spray pattern of the RS does not result in any accumulation of water in the reactor cavity.

The discussion on page 4.7-1 attempts to define the possible methods in which a continuous supply of water can be provided to the reactor cavity via the reactor vessel after vessel breach and explain how such sources can be available in the long term when they were apparently unsuccessful in preventing core damage. If low pressure systems are available but high pressure injection and/or recirculation were unavailable and the RCS remained at high pressure prior to vessel breach, core damage could not be prevented and the low pressure systems would not operate until the RCS pressure dropped below their pump shutoff heads at the time of vessel breach. The accident scenarios of interest address the situation where normal ECCS is unavailable but the QS and RS are successful and the outside RS pumps are manually aligned for vessel injection, thereby providing a pathway for water to the reactor cavity via the RCS and the hole in the bottom of the vessel.

5. Please address the following items related to containment isolation failure:

(a) With respect to the analysis of containment isolation failure probability, NUREG-1335 (Section 2.2.2.5, Page 2-11) states that "the analyses should address the five (5) areas identified in the Generic Letter, i.e., (1) the pathways that could significantly contribute to containment isolation failure, (2) the signals required to automatically isolate the penetration, (3) the potential for generating the signals for all initiating events, (4) the examination of the testing and maintenance procedures, and (5) the quantification of each containment isolation failure mode (including common-mode failure)". Please discuss your findings related to the above five (5) areas.

Response:

Containment isolation failures for the Level 2 analysis are governed by the Level 1 Top Event CI model discussed in Section 3.2.1.16 of the IPE submittal. Pre-existing leakage paths via penetrations thought to be closed but due to unforeseen circumstances left open were not modeled because of the subatmospheric containment design; i.e., that any such paths would be detected due to the demands on the system maintaining the vacuum. The first two areas identified in the Generic Letter; i.e., (1) the pathways that could significantly contribute to containment isolation failure, and (2) the signals required to automatically isolate the penetration, are addressed in Table 3.2.1.16-1, which is explained in Section 3.2.1.16.2. In response to the examination of the testing and maintenance procedures (item 4), Section 3.2.1.16.7 on page 3.2-106, notes that all penetrations which were not screened out from Table 3.2.1.16-1 require a Type C leak test and quarterly operability verification per OST 1.47.3A. In addition, maintenance is performed on an as-needed basis and operability checks are performed after each maintenance event. The potential for generating the isolation signals (item 3), listed in Table 3.2.1.16-1, for each initiating event analyzed in the IPE is listed below.

CIA: SLOCI - isolable small LOCA
SLOCN - nonisolable small LOCA
MLOCA - medium LOCA
LLOCA - large LOCA
ELOCA - excessive LOCA
VSX - interfacing systems LOCA
SGTR - steam generator tube rupture
SLB1 - steam line break in one steam generator
SLBC - steam line break in common RHR valve line
SLBD -steam line break down stream of the main steam isolation valves
AMSIV - closure of all main steam isolation valves
IMSIV - closure of one main steam isolation valve
MSV - main steam relief or safety valve opening
ISI - inadvertent safety injection signal
CRFL - control room HVAC equipment area internal flood

CIB: SLOCI - isolable small LOCA
SLOCN - nonisolable small LOCA
MLOCA - medium LOCA
LLOCA - large LOCA
ELOCA - excessive LOCA
SLB1 - steam line break in one steam generator
CRFL - control room HVAC equipment area internal flood

The quantification of each containment isolation failure mode including common cause (Generic Letter item 5) are addressed in Tables 5-1 and 5-2, which are reports generated from the PRA containment isolation (Top Event CI) model. A brief description for each of these reports is discussed on the following pages.

Table 5-1. This table consists of the containment isolation common cause failure modes, which were developed using the Multiple Greek Letter (MGL) m[illegible]odology. Incorporated into this table are the common cause group identifiers [illegible] basic events that are affected in the group, the order of the common cause [illegible]re mode modeled, the failure mode, and the database variables that were used to quantify the MGL equations.

MODEL Name: BV1
CCF Model Report for Top Event CI

13:22:46 30 JAN 1995
Page 1

Group ID : A

Basic Events	Description
AVFCTV1DA100A	TV-DA-100A FAILS TO CLOSE
AVFCTV1DA100B	TV-DA-100B FAILS TO CLOSE

Algebraic Method: MGL
Order = 1 out of 2

Failure Mode ID : CLOSE
Total Failure Rate = ZTVAOD
Beta = ZBVAOD

Group ID : B

Basic Events	Description
AVFCTV1DG108B	TV-DG-108B FAILS TO CLOSE
AVFCTV1DG108A	TV-DG-108A FAILS TO CLOSE

Algebraic Method: MGL
Order = 1 out of 2

Failure Mode ID : CLOSE
Total Failure Rate = ZTVAOC
Beta = ZBVAOD

Group ID : C

Basic Events	Description
AVFCTV1DG109A2	TV-DG-109A2 FAILS TO CLOSE
AVFCTV1DG109A1	TV-DG-109A1 FAILS TO CLOSE

Algebraic Method: MGL
Order = 1 out of 2

Failure Mode ID : CLOSE
Total Failure Rate = ZTVAOD
Beta = ZBVAOD

MODEL Name: BV1
CCF Model Report for Top Event CI

13:22:51 30 JAN 1995
Page 2

Group ID : D

Basic Events	Description
MVFCMOV1CH381	MOV-CH-381 FAILS TO CLOSE
MVFCMOV1CH378	INSIDE CNMT ISOL VALVE MOV-CH-378 FAILS TO CLOSE

Algebraic Method: MGL
Order = 1 out of 2

Failure Mode ID : CLOSE
Total Failure Rate = ZTVMOD
Beta = ZBVMOD

Group ID : E

Basic Events	Description
FCFCLCV1CH460A	LCV-CH-460A FAILS TO CLOSE
FCFCLCV1CH460B	LCV-CH-460B FAILS TO CLOSE
AVFCTV1CH204	TV-CH-204 FAILS TO CLOSE
AVFCTV1CH200A	TV-CH-200A FAILS TO CLOSE
AVFCTV1CH200C	TV-CH-200C FAILS TO CLOSE
AVFCTV1CH200B	TV-CH-200B FAILS TO CLOSE

Algebraic Method: MGL
Order = 3 out of 6

Failure Mode ID : CLOSE
Total Failure Rate = ZTVAOD
Beta = ZBVAOD
Gamma = ZGVAOD
Delta = ZDVAOD

Group ID : F

Basic Events	Description
AVFCTV1CV150A	TV-CV-150A FAILS TO CLOSE
AVFCTV1CV150B	TV-CV-150B FAILS TO CLOSE

Algebraic Method: MGL
Order = 1 out of 2

Failure Mode ID : CLOSE
Total Failure Rate = ZTVAOD
Beta = ZBVAOD

MODEL Name: BV1
CCF Model Report for Top Event CI

13:22:55 30 JAN 1995
Page 3

Group ID : G

Basic Events	Description
AVFCTV1CV150D	TV-CV-150D FAILS TO CLOSE
AVFCTV1CV150C	TV-CV-150C FAILS TO CLOSE

Algebraic Method: MGL
Order = 1 out of 2

Failure Mode ID : CLOSE
Total Failure Rate = ZTVAOD
Beta = ZBVAOD

Group ID : H

Basic Events	Description
AVFCTV1CV101A	TV-CV-101A FAILS TO CLOSE
AVFCTV1CV101B	TV-CV-101B FAILS TO CLOSE

Algebraic Method: MGL
Order = 1 out of 2

Failure Mode ID : CLOSE
Total Failure Rate = ZTVAOD
Beta = ZBVAOD

Group ID : I

Basic Events	Description
AVFCTV1LM100A1	TV-LM-100A1 FAILS TO CLOSE
AVFCTV1LM100A2	TV-LM-100A2 FAILS TO CLOSE

Algebraic Method: MGL
Order = 1 out of 2

Failure Mode ID : CLOSE
Total Failure Rate = ZTVAOD
Beta = ZBVAOD

Group ID : J

Basic Events	Description
AVFCTV1CV102	TV-CV-102 FAILS TO CLOSE
AVFCTV1CV1021	TV-CV-102-1 FAILS TO CLOSE

Algebraic Method: MGL
Order = 1 out of 2

Failure Mode ID : CLOSE
Total Failure Rate = ZTVAOD
Beta = ZBVAOD

Table 5-2. This table provides the cause table for each of the containment isolation split fractions that were quantified by using the fault tree for Top Event CI. The cause tables consist of the quantified minimal cutsets for each particular split fraction. These cutsets are ranked in descending order according to their quantified values. Additionally, this table shows the % Importance, or the percentage that each cutset contributes to the Monte Carlo mean split fraction value, and the % Cumulative, which is the cumulative summation of the % Importance. The cause table reports were generated by using a 99.9% cumulative cutoff for each of the split fractions. The alignment of the system when the cutset was quantified is also provided. These are all shown as being in normal alignment, since no maintenance or tests are performed on an unisolated component during plant operation. It should be noted that singleton cutsets, whose basic event identifiers are separated by a comma and enclosed in brackets [ ], are common cause failures of components. Independent failures of common cause components are shown as a single basic event enclosed in brackets.

MODEL Name: BV1LVL1
Cause Table for Top Event CI
Split Fraction CI1 - ALL SUPPORT

PE Value of CI1 = 6.8151E-03 Date : 25 JUN 1992 19:05
MC/LH Value of CI1 = 6.8701E-03 Date : 03 JUL 1992 02:51

13:21:25 30 JAN 1995
Page 1

No...	Cutsets...........	Value.....	% Importance	% Cumulative	Alignment...
1	[AVFCTVCV102,AVFCT VCV1021]	6.776E-04	9.8630	9.8630	NORMAL
2	[AVFCTVLM100A1,AVF CTVLM100A2]	6.776E-04	9.8630	19.7259	NORMAL
3	[AVFCTVCV150A,AVFC TVCV150B]	6.776E-04	9.8630	29.5889	NORMAL
4	[AVFCTVCV101A,AVFC TVCV101B]	6.776E-04	9.8630	39.4519	NORMAL
5	[AVFCTVCV150D,AVFC TVCV150C]	6.776E-04	9.8630	49.3149	NORMAL
6	[AVFCTVDA100A,AVFC TVDA100B]	6.776E-04	9.8630	59.1778	NORMAL
7	[AVFCTVDG109A2,AVF CTVDG109A1]	6.776E-04	9.8630	69.0408	NORMAL
8	[AVFCTVDG108B,AVFC TVDG108A]	6.776E-04	9.8630	78.9038	NORMAL
9	OPRCI2	6.500E-04	9.4612	88.3650	NORMAL
10	[MVFCMOVCH381,MVFC MOVCH378]	1.101E-04	1.6026	89.9676	NORMAL
11	[AVFCTVDG109A2] * [AVFCTVDG109A1]	8.296E-05	1.2075	91.1751	NORMAL
12	[AVFCTVCV102] * [AVFCTVCV1021]	8.296E-05	1.2075	92.3827	NORMAL
13	[AVFCTVLM100A1] * [AVFCTVLM100A2]	8.296E-05	1.2075	93.5902	NORMAL
14	[AVFCTVCV101A] * [AVFCTVCV101B]	8.296E-05	1.2075	94.7978	NORMAL
15	[AVFCTVCV150D] * [AVFCTVCV150C]	8.296E-05	1.2075	96.0052	NORMAL
16	[AVFCTVCV150A] * [AVFCTVCV150B]	8.296E-05	1.2075	97.2129	NORMAL
17	[AVFCTVDG108B] * [AVFCTVDG108A]	8.296E-05	1.2075	98.4205	NORMAL
18	[AVFCTVDA100A] * [AVFCTVDA100B]	8.296E-05	1.2075	99.6281	NORMAL

MODEL Name: BV1LVL1
Cause Table for Top Event CI
Split Fraction CI2 - LOSS OF AC ORANGE

PE Value of CI2 = 8.52[illegible]E-03 Date : 25 JUN 1992 19:05
MC/LH Value of CI2 = 8.4332E-03 Date : 03 JUL 1992 02:51

13:21:40 30 JAN 1995
Page 1

No...	Cutsets...........	Value.....	% Importance	% Cumulative	Alignment...
1	[MVFCMOVCH381]	1.715E-03	20.3364	20.3364	NORMAL
2	[AVFCTVDG108B,AVFC TVDG108A]	6.641E-04	7.8749	28.2113	NORMAL
3	[AVFCTVCV102,AVFCT VCV1021]	6.641E-04	7.8749	36.0861	NORMAL
4	[AVFCTVLM100A1,AVF CTVLM100A2]	6.641E-04	7.8749	43.9610	NORMAL
5	[AVFCTVCV150A,AVFC TVCV150B]	6.641E-04	7.8749	51.8359	NORMAL
6	[AVFCTVCV101A,AVFC TVCV101B]	6.641E-04	7.8749	59.7107	NORMAL
7	[AVFCTVCV150D,AVFC TVCV150C]	6.641E-04	7.8749	67.5856	NORMAL
8	[AVFCTVDA100A,AVFC TVDA100B]	6.641E-04	7.8749	75.4605	NORMAL
9	[AVFCTVDG109A2,AVF CTVDG109A1]	6.641E-04	7.8749	83.3353	NORMAL
10	OPRCI2	5.965E-04	7.0733	90.4086	NORMAL
11	[MVFCMOVCH381,MVFC MOVCH378]	1.099E-04	1.3032	91.7118	NORMAL
12	[AVFCTVDG109A2] * [AVFCTVDG109A1]	8.493E-05	1.0071	92.7189	NORMAL
13	[AVFCTVCV102] * [AVFCTVCV1021]	8.493E-05	1.0071	93.7260	NORMAL
14	[AVFCTVLM100A1] * [AVFCTVLM100A2]	8.493E-05	1.0071	94.7331	NORMAL
15	[AVFCTVCV101A] * [AVFCTVCV101B]	8.493E-05	1.0071	95.7402	NORMAL
16	[AVFCTVCV150D] * [AVFCTVCV150C]	8.493E-05	1.0071	96.7473	NORMAL
17	[AVFCTVCV150A] * [AVFCTVCV150B]	8.493E-05	1.0071	97.7544	NORMAL
18	[AVFCTVDG108B] * [AVFCTVDG108A]	8.493E-05	1.0071	98.7615	NORMAL
19	[AVFCTVDA100A] * [AVFCTVDA100B]	8.493E-05	1.0071	99.7686	NORMAL

MODEL Name: BV1LVL1
Cause Table for Top Event CI
Split Fraction CI3 - LOSS OF AC PURPLE

PE Value of CI3 = 8.9030E-03 Date : 25 JUN 1992 19:05
MC/LH Value of CI3 = 8.8773E-03 Date : 03 JUL 1992 02:51

13:21:55 30 JAN 1995
Page 1

No...	Cutsets...........	Value.....	% Importance	% Cumulative	Alignment...
1	[MVFCMOVCH378]	1.701E-03	19.1612	19.1612	NORMAL
2	[AVFCTVDG108B,AVFC TVDG108A]	6.731E-04	7.5823	26.7435	NORMAL
3	[AVFCTVCV102,AVFCT VCV1021]	6.731E-04	7.5823	34.3257	NORMAL
4	[AVFCTVLM100A1,AVF CTVLM100A2]	6.731E-04	7.5823	41.9080	NORMAL
5	[AVFCTVCV150A,AVFC TVCV150B]	6.731E-04	7.5823	49.4902	NORMAL
6	[AVFCTVCV101A,AVFC TVCV101B]	6.731E-04	7.5823	57.0725	NORMAL
7	[AVFCTVCV150D,AVFC TVCV150C]	6.731E-04	7.5823	64.6547	NORMAL
8	[AVFCTVDA100A,AVFC TVDA100B]	6.731E-04	7.5823	72.2370	NORMAL
9	[AVFCTVDG109A2,AVF CTVDG109A1]	6.731E-04	7.5823	79.8192	NORMAL
10	OPRCI2	6.118E-04	6.8917	86.7109	NORMAL
11	CVFCCH369	3.642E-04	4.1026	90.8135	NORMAL
12	[MVFCMOVCH381,MVFC MOVCH378]	1.093E-04	1.2312	92.0448	NORMAL
13	[AVFCTVDG109A2] * [AVFCTVDG109A1]	8.569E-05	.9653	93.0100	NORMAL
14	[AVFCTVCV102] * [AVFCTVCV1021]	8.569E-05	.9653	93.9753	NORMAL
15	[AVFCTVLM100A1] * [AVFCTVLM100A2]	8.569E-05	.9653	94.9406	NORMAL
16	[AVFCTVCV101A] * [AVFCTVCV101B]	8.569E-05	.9653	95.9058	NORMAL
17	[AVFCTVCV150D] * [AVFCTVCV150C]	8.569E-05	.9653	96.8712	NORMAL
18	[AVFCT 150A] * [AVFCTVCV150B]	8.569E-05	.9653	97.8364	NORMAL
19	[AVFCTVDG108B] * [AVFCTVDG108A]	8.569E-05	.9653	98.8017	NORMAL
20	[AVFCTVDA100A] * [AVFCTVDA100B]	8.569E-05	.9653	99.7670	NORMAL

MODEL Name: BV1LVL1
Cause Table for Top Event CI
Split Fraction CI4 - LOSS OF SSPS TRAIN A

PE Value of CI4 = 8.0151E-02 Date : 25 JUN 1992 19:05
MC/LH Value of CI4 = 7.8372E-02 Date : 03 JUL 1992 02:51

13:22:11 30 JAN 1995
Page 1

No...	Cutsets...........	Value.....	% Importance	% Cumulative	Alignment...
1	[AVFCTVDG108B]	8.829E-03	11.2655	11.2655	NORMAL
2	[AVFCTVDG109A2]	8.829E-03	11.2655	22.5310	NORMAL
3	[AVFCTVCV1021]	8.829E-03	11.2655	33.7966	NORMAL
4	[AVFCTVLM100A2]	8.829E-03	11.2655	45.0621	NORMAL
5	[AVFCTVCV101B]	8.829E-03	11.2655	56.3276	NORMAL
6	[AVFCTVCV150D]	8.829E-03	11.2655	67.5931	NORMAL
7	[AVFCTVCV150A]	8.829E-03	11.2655	78.8586	NORMAL
8	[AVFCTVDA100B]	8.829E-03	11.2655	90.1242	NORMAL
9	[MVFCMOVCH381]	1.722E-03	2.1972	92.3214	NORMAL
10	OPRCI2	6.600E-04	.8421	93.1635	NORMAL
11	[AVFCTVLM100A1,AVF CTVLM100A2]	6.520E-04	.8319	93.9954	NORMAL
12	[AVFCTVCV150A,AVFC TVCV150B]	6.520E-04	.8319	94.8274	NORMAL
13	[AVFCTVCV101A,AVFC TVCV101B]	6.520E-04	.8319	95.6593	NORMAL
14	[AVFCTVCV150D,AVFC TVCV150C]	6.520E-04	.8319	96.4912	NORMAL
15	[AVFCTVDA100A,AVFC TVDA100B]	6.520E-04	.8319	97.3232	NORMAL
16	[AVFCTVDG109A2,AVF CTVDG109A1]	6.520E-04	.8319	98.1551	NORMAL
17	[AVFCTVDG108B,AVFC TVDG108A]	6.520E-04	.8319	98.9870	NORMAL
18	[AVFCTVCV102,AVFCT VCV1021]	6.520E-04	.8319	99.8189	NORMAL

MODEL Name: BV1LVL1
Cause Table for Top Event CI
Split Fraction CI5 - LOSS OF SSPS TRAIN B

PE Value of CI5 = 8.0603E-02 Date : 25 JUN 1992 19:05
MC/LH Value of CI5 = 8.0354E-02 Date : 03 JUL 1992 02:51

13:22:21 30 JAN 1995
Page 1

No...	Cutsets...........	Value.....	% Importance	% Cumulative	Alignment...
1	[AVFCTVDA100A]	8.966E-03	11.1581	11.1581	NORMAL
2	[AVFCTVDG109A1]	8.966E-03	11.1581	22.3161	NORMAL
3	[AVFCTVCV102]	8.966E-03	11.1581	33.4742	NORMAL
4	[AVFCTVLM100A1]	8.966E-03	11.1581	44.6322	NORMAL
5	[AVFCTVCV101A]	8.966E-03	11.1581	55.7903	NORMAL
6	[AVFCTVCV150C]	8.966E-03	11.1581	66.9484	NORMAL
7	[AVFCTVCV150B]	8.966E-03	11.1581	78.1064	NORMAL
8	[AVFCTVDG108A]	8.966E-03	11.1581	89.2645	NORMAL
9	[MVFCMOVCH378]	1.696E-03	2.1106	91.3751	NORMAL
10	[AVFCTVCV102,AVFCT VCV1021]	7.240E-04	.9010	92.2761	NORMAL
11	[AVFCTVLM100A1,AVF CTVLM100A2]	7.240E-04	.9010	93.1771	NORMAL
12	[AVFCTVCV150A,AVFC TVCV150B]	7.240E-04	.9010	94.0781	NORMAL
13	[AVFCTVCV101A,AVFC TVCV101B]	7.240E-04	.9010	94.9791	NORMAL
14	[AVFCTVCV150D,AVFC TVCV150C]	7.240E-04	.9010	95.8802	NORMAL
15	[AVFCTVDG109A2,AVF CTVDG109A1]	7.240E-04	.9010	96.7812	NORMAL
16	[AVFCTVDG108B,AVFC TVDG108A]	7.240E-04	.9010	97.6822	NORMAL
17	[AVFCTVDA100A,AVFC TVDA100B]	7.240E-04	.9010	98.5832	NORMAL
18	OPRCI2	5.675E-04	.7062	99.2894	NORMAL
19	CVFCCH369	2.056E-04	.2559	99.5453	NORMAL
20	[MVFCMOVCH381,MVFC MOVCH378]	6.151E-05	.0765	99.6218	NORMAL

MODEL Name: BV1LVL1
Cause Table for Top Event CI
Split Fraction CI6 - LOSS OF ALL AC, SA & SB AVAILABLE

PE Value of CI6 = 8.9247E-03 Date : 25 JUN 1992 19:05
MC/LH Value of CI6 = 8.9281E-03 Date : 03 JUL 1992 02:51

13:22:36 30 JAN 1995
Page 1

No...	Cutsets...........	Value.....	% Importance	% Cumulative	Alignment...
1	OPRCI1	2.289E-03	25.6381	25.6381	NORMAL
2	[AVFCTVDG108B,AVFC TVDG108A]	6.702E-04	7.5066	33.1448	NORMAL
3	[AVFCTVCV102,AVFCT VCV1021]	6.702E-04	7.5066	40.6514	NORMAL
4	[AVFCTVLM100A1,AVF CTVLM100A2]	6.702E-04	7.5066	48.1580	NORMAL
5	[AVFCTVCV150A,AVFC TVCV150B]	6.702E-04	7.5066	55.6646	NORMAL
6	[AVFCTVCV101A,AVFC TVCV101B]	6.702E-04	7.5066	63.1713	NORMAL
7	[AVFCTVCV150D,AVFC TVCV150C]	6.702E-04	7.5066	70.6779	NORMAL
8	[AVFCTVDA100A,AVFC TVDA100B]	6.702E-04	7.5066	78.1845	NORMAL
9	[AVFCTVDG109A2,AVF CTVDG109A1]	6.702E-04	7.5066	85.6912	NORMAL
10	OPRCI2	5.810E-04	6.5075	92.1987	NORMAL
11	[AVFCTVDG109A2] * [AVFCTVDG109A1]	8.441E-05	.9454	93.1441	NORMAL
12	[AVFCTVCV102] * [AVFCTVCV1021]	8.441E-05	.9454	94.0896	NORMAL
13	[AVFCTVLM100A1] * [AVFCTVLM100A2]	8.441E-05	.9454	95.0350	NORMAL
14	[AVFCTVCV101A] * [AVFCTVCV101B]	8.441E-05	.9454	95.9805	NORMAL
15	[AVFCTVCV150D] * [AVFCTVCV150C]	8.441E-05	.9454	96.9259	NORMAL
16	[AVFCTVCV150A] * [AVFCTVCV150B]	8.441E-05	.9454	97.8713	NORMAL
17	[AVFCTVDG108B] * [AVFCTVDG108A]	8.441E-05	.9454	98.8167	NORMAL
18	[AVFCTVDA100A] * [AVFCTVDA100B]	8.441E-05	.9454	99.7621	NORMAL

(b) Section 7, Page 7-1 of the submittal, notes the following:

The operation of the containment at subatmospheric conditions and the continual monitoring of in-leakage make the likelihood of a pre-existing failure of containment isolation at the time of a severe accident negligible. However, Table 1-3, Page 1.4-7, indicates that BV-1 has a containment isolation failure frequency of 3.48E-5 per reactor year, which is associated with 16.3% of the total CDF. Please explain this discrepancy. Why do the containment isolation failures contribute so high (16.3%) to the CDF?

Response:

Section 7, page 7-1 of the IPE submittal should read, "The operation of the containment at subatmospheric conditions and the continual monitoring of in-leakage make the likelihood of a **large** pre-existing failure of containment isolation at the time of a severe accident negligible." The reasoning behind this assumption is that pre-existing containment isolation failures greater than 3 inches in diameter would be obvious to the operator since he would be unable to maintain subatmospheric containment pressure. This is noted on page 4.3-6 "Containment Isolation and Bypass Status" and on page 4.6-14 "Top Event 12 - Large Containment Failure Prior to Vessel Breach (L1)". The containment isolation failure frequency of 3.48×10^{-5} per reactor year (16.3% of the total CDF) reported in Table 1-3, page 1.4-7 only addresses small containment isolation failures, i.e., smaller than 3 inches in diameter. The reason that the small containment isolation failure plant damage states contribute so much to the CDF is because the majority of the failures (96.2%, based on the saved sequence database) are due to the emergency switchgear ventilation failing (15.5% of the total CDF), which results in the guaranteed failure of all emergency power and consequently, containment isolation (Top Event CI). The normally open RCP seal return line requires AC power to close. Failure to isolate the RCP seal return line was modeled as a failure of containment isolation.

(c) Since the BV1 plant has a non-negligibly high containment isolation failure, please explain and give the magnitudes of the contributors to the isolation failures at BV-1.

Response:

As stated in the response to Back-End Question 5.(b) above, guaranteed failures of containment isolation due to emergency switchgear ventilation failures, contribute to 96.2% of the total frequency for small containment isolation failure plant damage states. No credit was given for manual isolation of the RCP seal return line for sequences involving loss of emergency switchgear ventilation. The loss of this ventilation was assumed to also result in failure of all vital instrumentation which would complicate the action to affect manual isolation. Another 3.4% of the total frequency is due to guaranteed failures of containment isolation that are related to failures of both SSPS trains or the failure of one SSPS train and the opposite train of AC electrical power. These guaranteed failures of containment isolation account for 99.6% of the total small containment isolation failure plant damage states. Hence, only 0.4% of the total frequency actually come from probabilistic failures of the valves which must close to effect containment isolation.

(d) As shown in Table 4.8-3, Page 4.8-60 of the submittal, containment isolation failures are involved in 10% of the large, early release group (RCG I). However, isolation failures are excluded from the table on Page 4.8-1, which lists the major contributors to RCG I, including rocket mode failures that contribute less than 0.01% to the group frequency. Please explain.

Response:

As noted in Section 4.6 (page 4.6-14), "because the Beaver Valley Unit 1 containment normally operates at subatmospheric conditions, the existence of large preexisting leaks is believed to be negligible." This statement is rooted in the basic understanding of the manner in which subatmospheric containments are operated and is consistent with the severe accident analyses performed for Surry (NUREG/CR-4551, Vol. 3, Rev. 1, Part 1, Section 2.1.2, Page 2.2) by Sandia National Laboratories. This reference states that since containment pressure is normally maintained at approximately 5 psia below atmospheric pressure, ... "makes the probability of pre-existing leaks negligible." The vacuum pumps that normally maintain containment pressure below atmospheric are of very limited capacity, and can not maintain such conditions for any significantly sized leak. Deviations from the allowable range of differential pressure between the containment and ambient would be alarmed in the control room. Because of Technical Specification requirements, the size of any hole that would "go unnoticed is so small that it could be ignored."

The containment isolation failure contribution to large, early releases shown in Table 4.8-3 merely reflects that the fact that **small** isolation failures were prevalent in 10% of the large, early failure frequency. Such failures alone do not result in large releases. Small breaches in the containment boundary will not necessarily preclude the possibility of large, early failures, which in fact are predicted to occur. The table shown on page 4.8-1 indicates the relative importance of the various phenomena which cause a **large**, early containment breach.

(e) The containment isolation failure size for BV-1 was assumed to be less than 3" in diameter. What was the lower limit of the opening size below which containment was considered isolated?

Response:

Pre-existing containment isolation failures greater than 3 inches in diameter were ruled out in the Level 2 analysis based on the reasoning stated in the response to Back-End Questions 5.(b) and 5.(d), above. Additionally, containment penetrations greater than 2 inches in diameter that connect to the containment environment and are normally isolated during plant operation were screened out in the Level 1 analysis, as described in Section 3.2.1.16, on page 3.2-103 of the IPE submittal. The Level 1 containment isolation analysis did not have a lower limit of the opening size in which the containment was considered isolated. It reviewed all containment penetrations, from 42 inches down to 1/8 inch diameter in size, as shown on Table 3.2.1.16-1, and did not screen out any penetration based on size alone. As can be seen on Table 3.2.1.16-1 and summarized on Table 3.2.1.16-2 , containment penetrations of 2", 1" and 3/8" in diameter were used in the containment isolation top event model, however, 3/8 inch lines and smaller were screened out in the Bypass LOCA analysis (see Table 3.1.3-8) as being negligible.

6. You state (on Page 4.6-9) that the MAAP results indicate that the hot leg will fail first "are somewhat questionable". Please describe those MAAP results and why they do not affect your conclusions regarding the hot leg failing before the Steam Generator tubes.

Response

The statement on page 4.6-9 states that "the MAAP results are somewhat questionable" and not "*that the MAAP results indicate that the hot leg will fail first are somewhat questionable*" as stated in the question. MAAP analyses were performed for Beaver Valley at a time when there were significant changes in the results when the same sequence was modeled with different versions of the program. This was especially true for hydrogen generation , RCS gas temperatures, and natural circulation flows. In addition, confirmatory runs indicated that there were even computer to computer differences for the same run and same version of the program. Thus, the analyses of induced steam generator tube and hot leg/surge line failures which was reported in Section 4.6.2 was tainted by questionable MAAP results. A probabilistic analysis based on the available MAAP results indicated that the mean value of the conditional (conditioned on RCS pressure at the system setpoint during core degradation) probability of induced steam generator tube rupture was less than 0.001 and the conditional probability of a hot leg/surge line failure occurring before vessel breach was 0.9.

As noted on page 4.6-12, the values for the conditional split fractions actually used in the Beaver Valley IPE Study for induced steam generator tube failure (0.018) and hot leg/surge line failure (0.72) were based on the evaluations performed for Surry by Sandia National Laboratories (NUREG/CR-4551, Vol. 3, Rev. 1, Part 2, APET Question Nos. 19 and 20). Relative to the predictions based on MAAP, the Sandia values are somewhat conservative. Thus, our concerns with the MAAP results had no impact on our conclusions regarding the induced failure of the hot leg prior to failure of the steam generator tubes.

7. Please provide the following:

   (a) Frequencies of the most significant release categories

   Response:

   See included Table 7-1.

Table 7-1

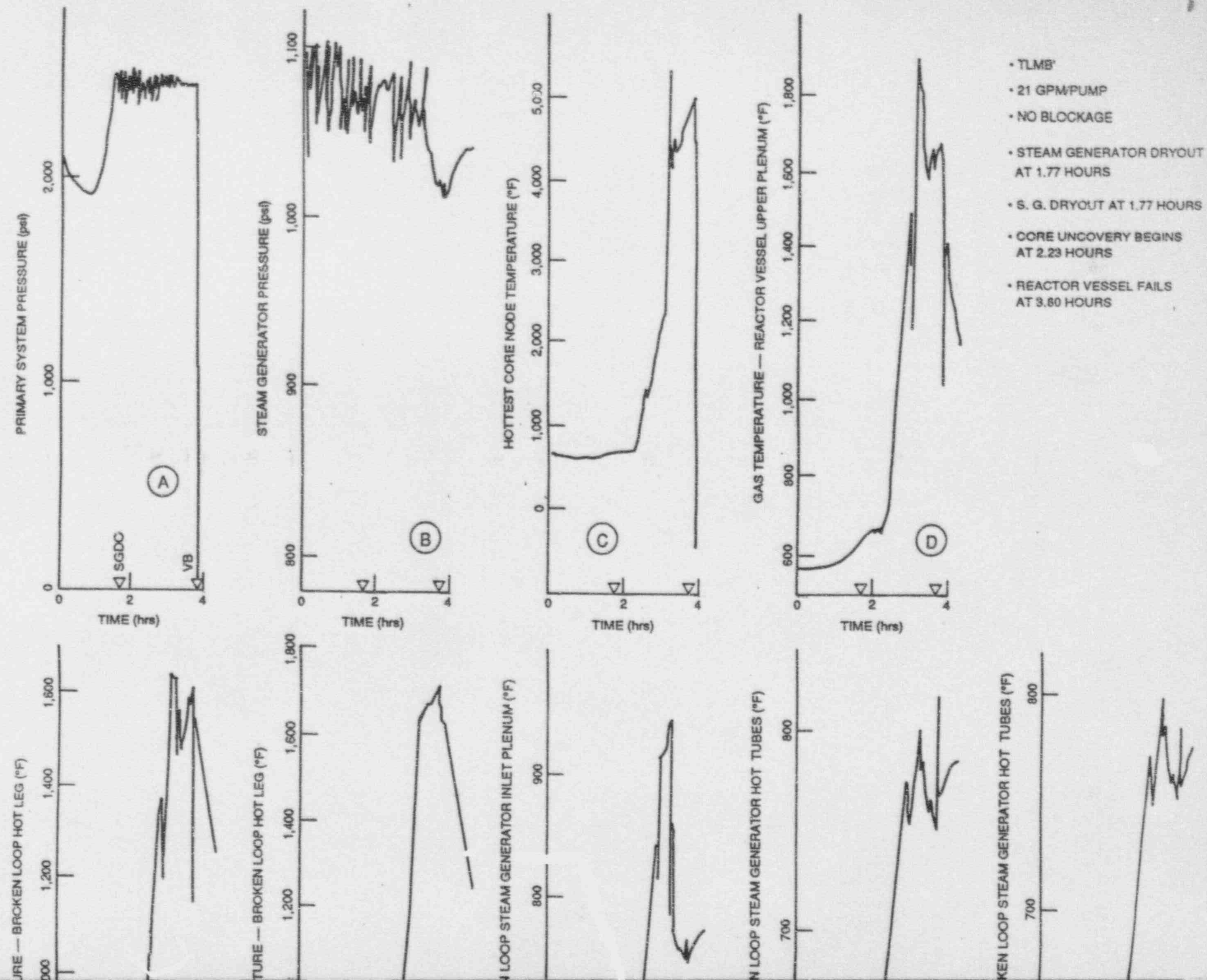
MODEL Name: BV1LVL2
Release Category End State Totals

Release Category	Frequency	Release Cat. Group
BV01	2.2225E-06	I
BV02	3.8136E-06	I
BV03	2.1281E-06	I
BV04	9.7823E-07	I
BV05	1.8143E-05	II
BV06	0.0000E+00	II
BV07	2.0676E-05	II
BV08	0.0000E+00	II
BV09	0.0000E+00	III
BV10	5.0985E-08	III
BV11	0.0000E+00	III
BV12	5.1041E-07	III
BV13	7.7928E-05	III
BV14	0.0000E+00	III
BV15	4.9134E-06	III
BV16	0.0000E+00	III
BV17	8.1426E-06	III
BV18	3.2414E-07	I
BV19	1.0897E-06	I
BV20	8.3235E-06	II
BV21	6.1746E-05	IV

(b) MAAP output curves that are readable. Page 4.6-50

Response:

See included Figure 4.6-7



PRIMARY SYSTEM PRESSURE (psi)
0
1,000
2,000
TIME (hrs)
SGDC
VB
A
STEAM GENERATOR PRESSURE (psi)
800
900
1,000
1,100
B
HOTTEST CORE NODE TEMPERATURE (°F)
0
1,000
2,000
3,000
4,000
5,000
C
GAS TEMPERATURE — REACTOR VESSEL UPPER PLENUM (°F)
600
800
1,000
1,200
1,400
1,600
1,800
D
• TLMB'
• 21 GPM/PUMP
• NO BLOCKAGE
• STEAM GENERATOR DRYOUT AT 1.77 HOURS
• S. G. DRYOUT AT 1.77 HOURS
• CORE UNCOVERY BEGINS AT 2.23 HOURS
• REACTOR VESSEL FAILS AT 3.60 HOURS
URE — BROKEN LOOP HOT LEG (°F)
1,200
1,400
1,600
TURE — BROKEN LOOP HOT LEG (°F)
1,200
1,400
1,600
1,800
N LOOP STEAM GENERATOR INLET PLENUM (°F)
800
900
N LOOP STEAM GENERATOR HOT TUBES (°F)
700
800
KEN LOOP STEAM GENERATOR HOT TUBES (°F)
700
800

9503130375-01

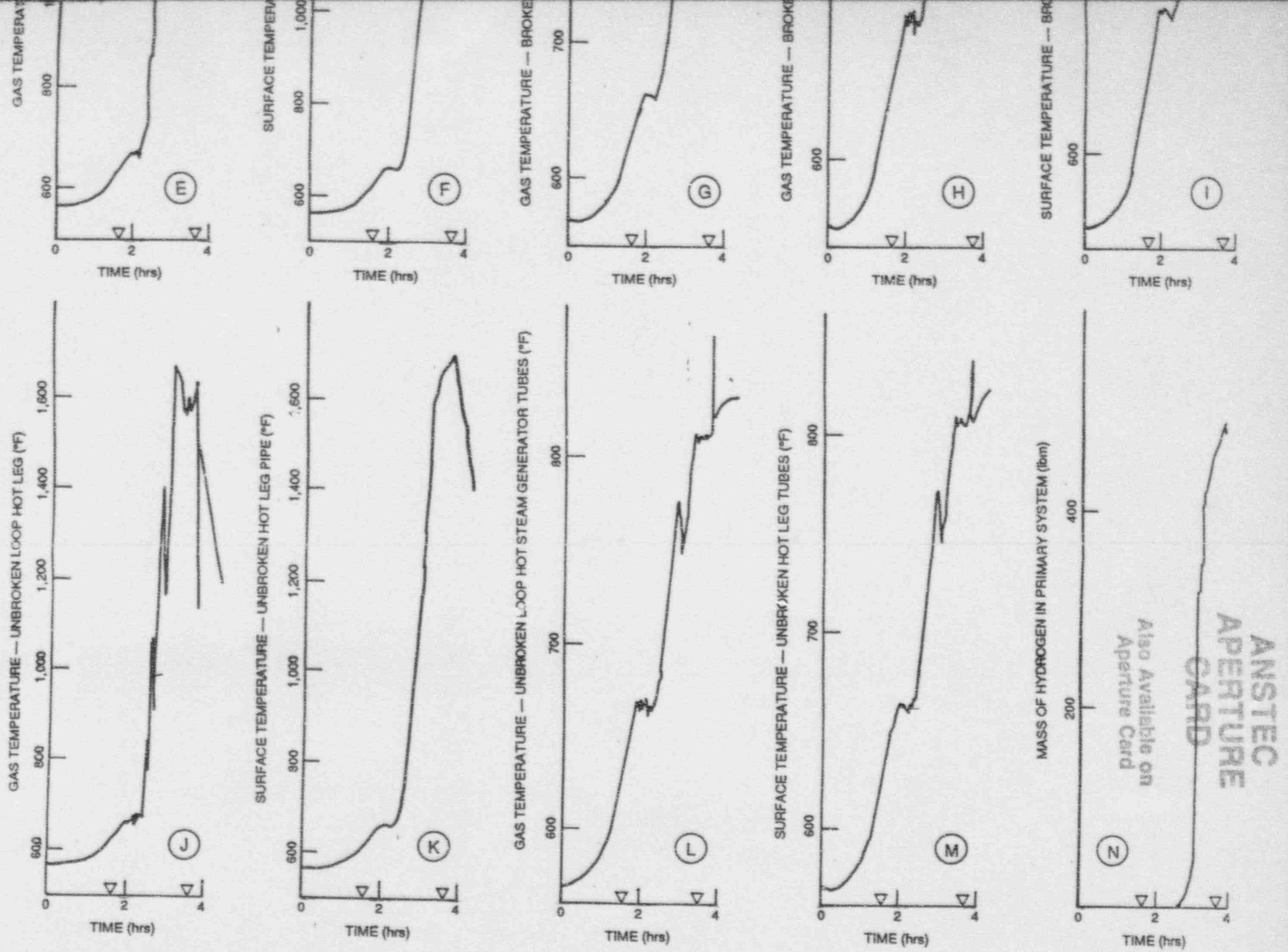
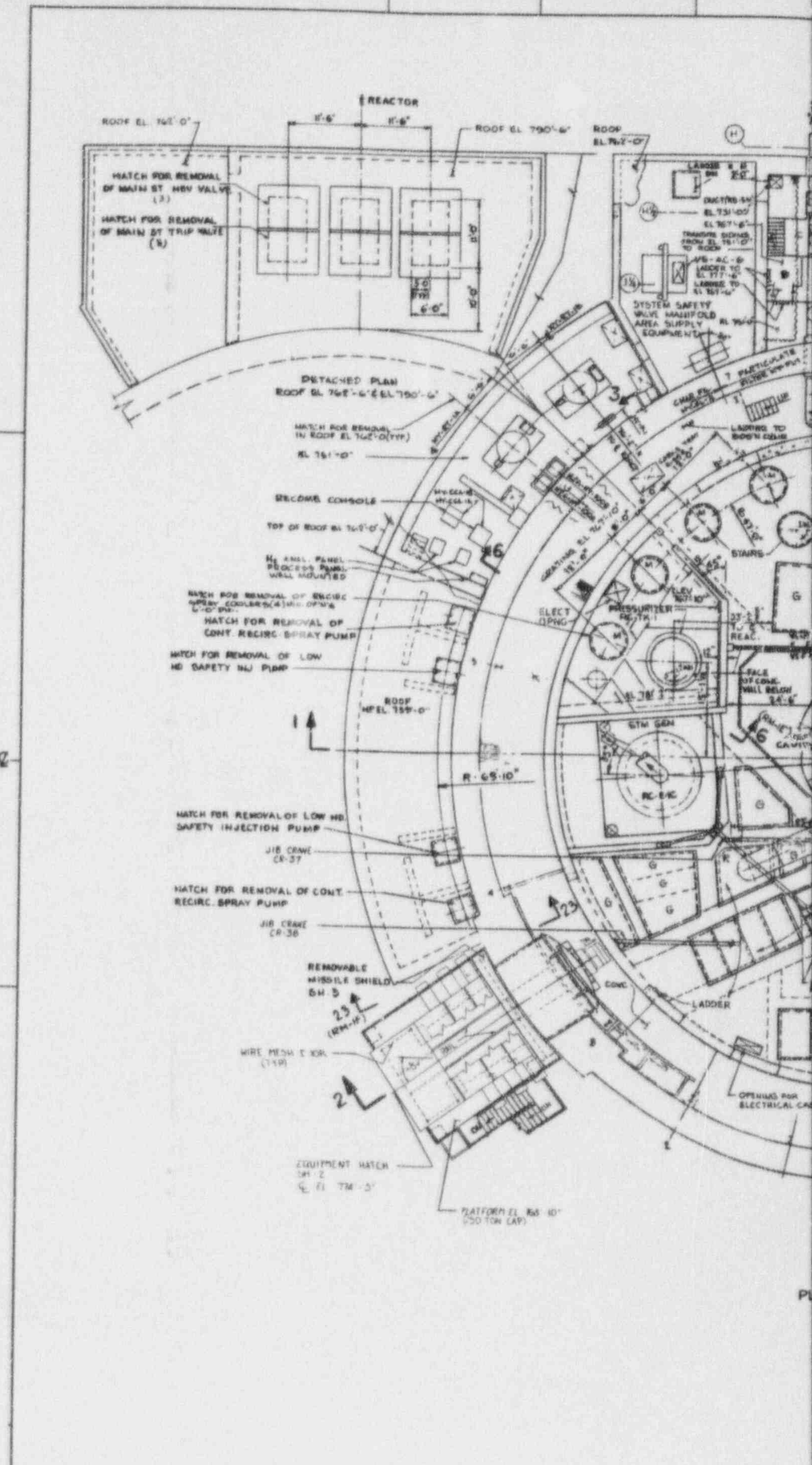


Figure 4.6-7. MAAP Output Curves

(c) Drawing of the vessel and reactor cavity showing the shield tanks, the lead shield discussed in the IPE submittal.

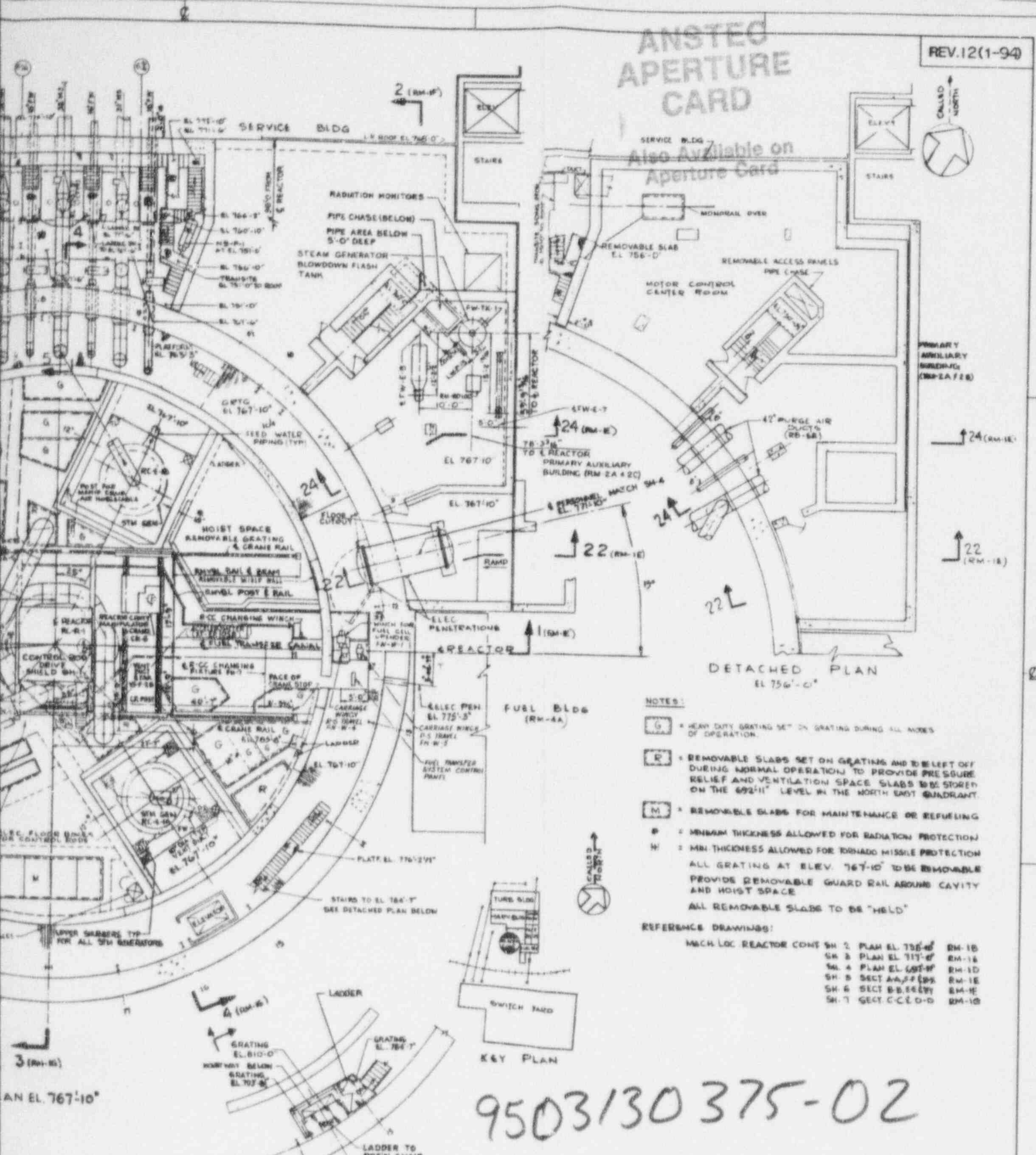
Response:

See included UFSAR Figures 5.1-1 through 5.1-7.



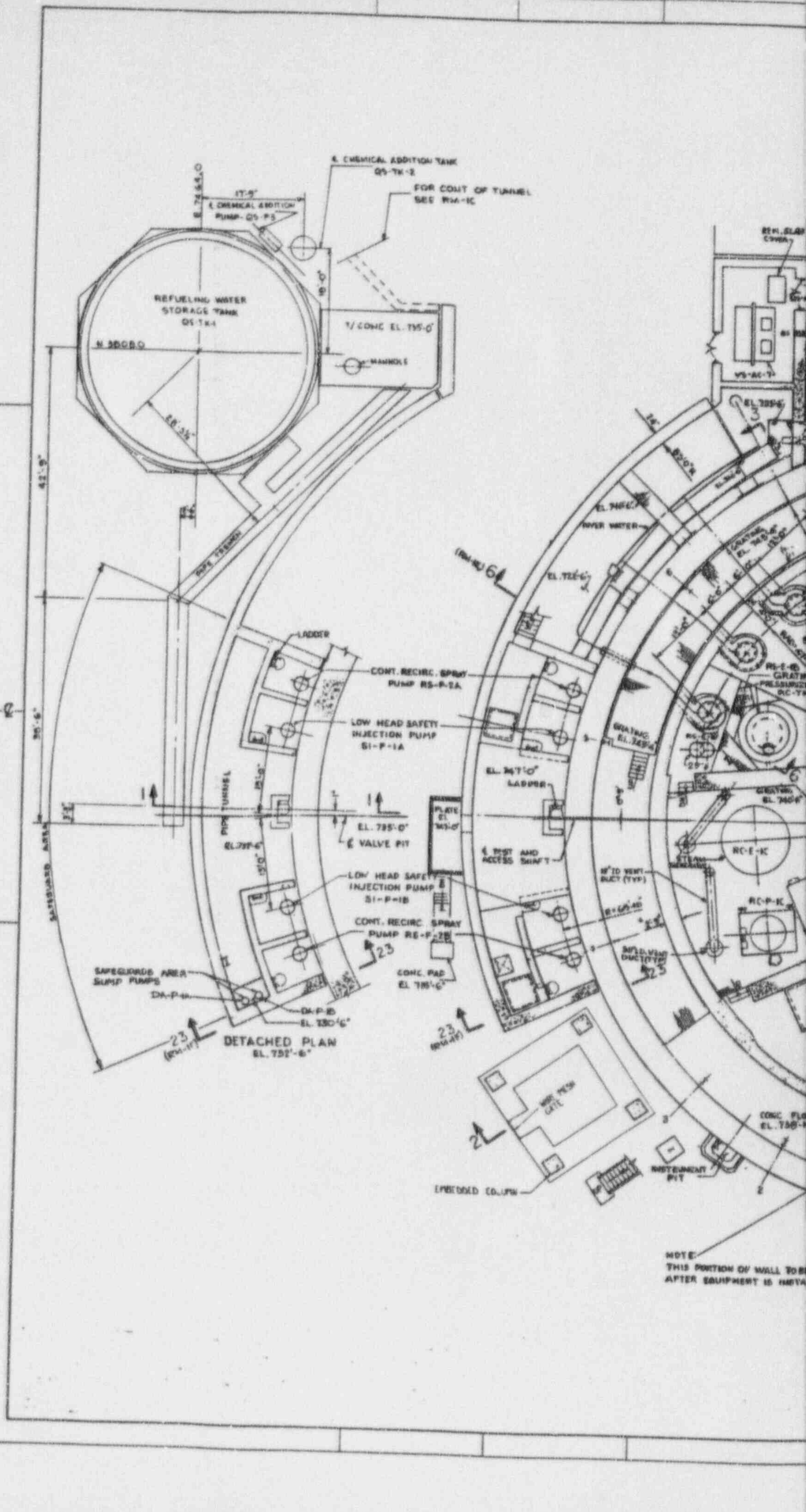
REACTOR
ROOF EL 762'-0"
ROOF EL 790'-6"
HATCH FOR REMOVAL OF MAIN ST. HDV VALVE (3)
HATCH FOR REMOVAL OF MAIN ST TRIP VALVE (3)
DETACHED PLAN ROOF EL 762'-6" & EL 790'-6"
HATCH FOR REMOVAL IN ROOF EL 762'-0"(TYP)
EL 761'-0"
RECOMB. CONSOLE
H2 ANAL. PANEL PROCESS PANEL WALL MOUNTED
HATCH FOR REMOVAL OF CONT. RECIRC. SPRAY PUMP
HATCH FOR REMOVAL OF LOW HD SAFETY INJ PUMP
R. 65'-10"
HATCH FOR REMOVAL OF LOW HD SAFETY INJECTION PUMP
JIB CRANE CR-37
HATCH FOR REMOVAL OF CONT. RECIRC. SPRAY PUMP
JIB CRANE CR-38
REMOVABLE MISSILE SHIELD SH-3
WIRE MESH DOOR
EQUIPMENT HATCH
SYSTEM SAFETY VALVE MANIFOLD AREA SUPPLY EQUIPMENT
STAIRS
ELECT OPNG
PRESSURIZER
STM GEN
LADDER
OPENING FOR ELECTRICAL CA

REV.12(1-94)

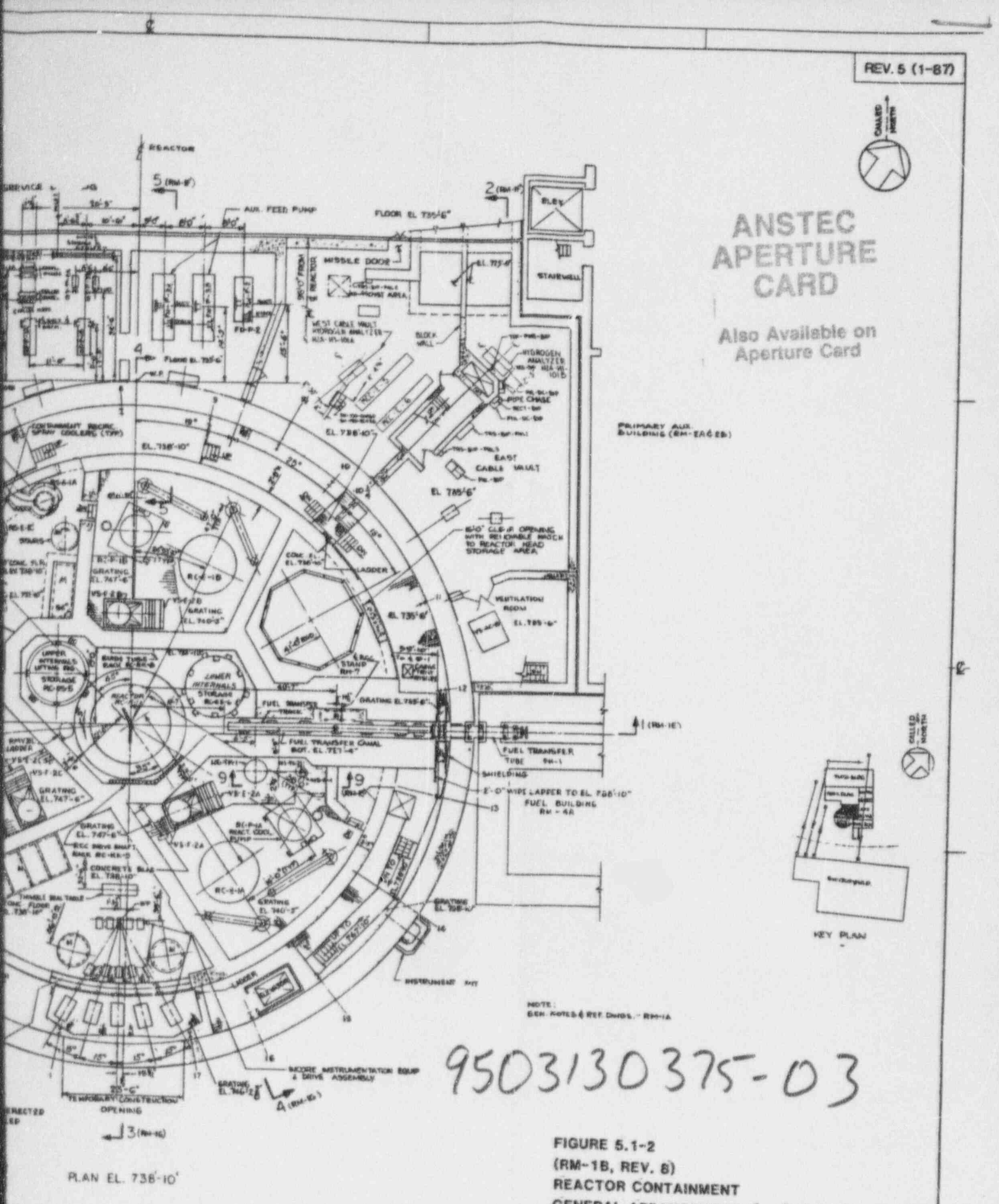


9503130375-02

FIGURE 5.1-1
(RM-1A)
REACTOR CONTAINMENT GENERAL ARRANGEMENT
EL. 767' - 10"
BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT

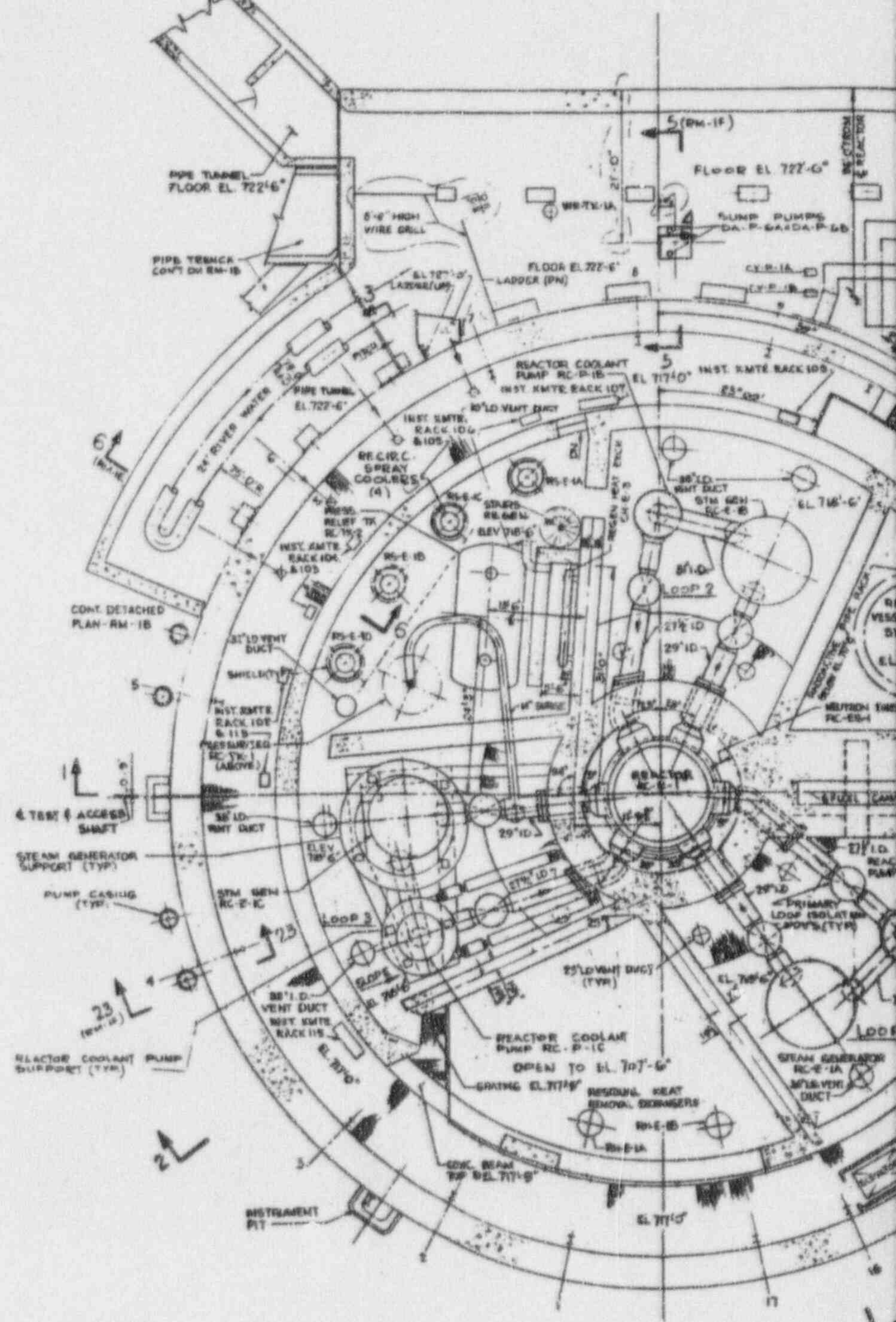

REFUELING WATER STORAGE TANK QS-TK-1
℄ CHEMICAL ADDITION TANK QS-TK-2
℄ CHEMICAL ADDITION PUMP-QS-P-3
FOR CONT OF TUNNEL SEE RM-1C
T/CONC EL. 735'-0"
MANHOLE
PIPE TRENCH
PIPE TUNNEL
SAFEGUARD AREA
LADDER
CONT. RECIRC. SPRAY PUMP RS-P-2A
LOW HEAD SAFETY INJECTION PUMP SI-P-1A
EL. 735'-0"
℄ VALVE PIT
LOW HEAD SAFETY INJECTION PUMP SI-P-1B
CONT. RECIRC. SPRAY PUMP RS-P-2B
SAFEGUARDS AREA SUMP PUMPS
DA-P-1A
DA-P-1B
EL. 730'-6"
CONC. PAD EL. 715'-6"
DETACHED PLAN EL. 732'-6"
RIVER WATER
GRATING
LADDER
℄ HOIST AND ACCESS SHAFT
STEAM
RC-E-1C
RC-P-1C
INSTRUMENT PIT
EMBEDDED COLUMN
NOTE: THIS PORTION OF WALL TO BE AFTER EQUIPMENT IS INSTA

REV. 5 (1-87)



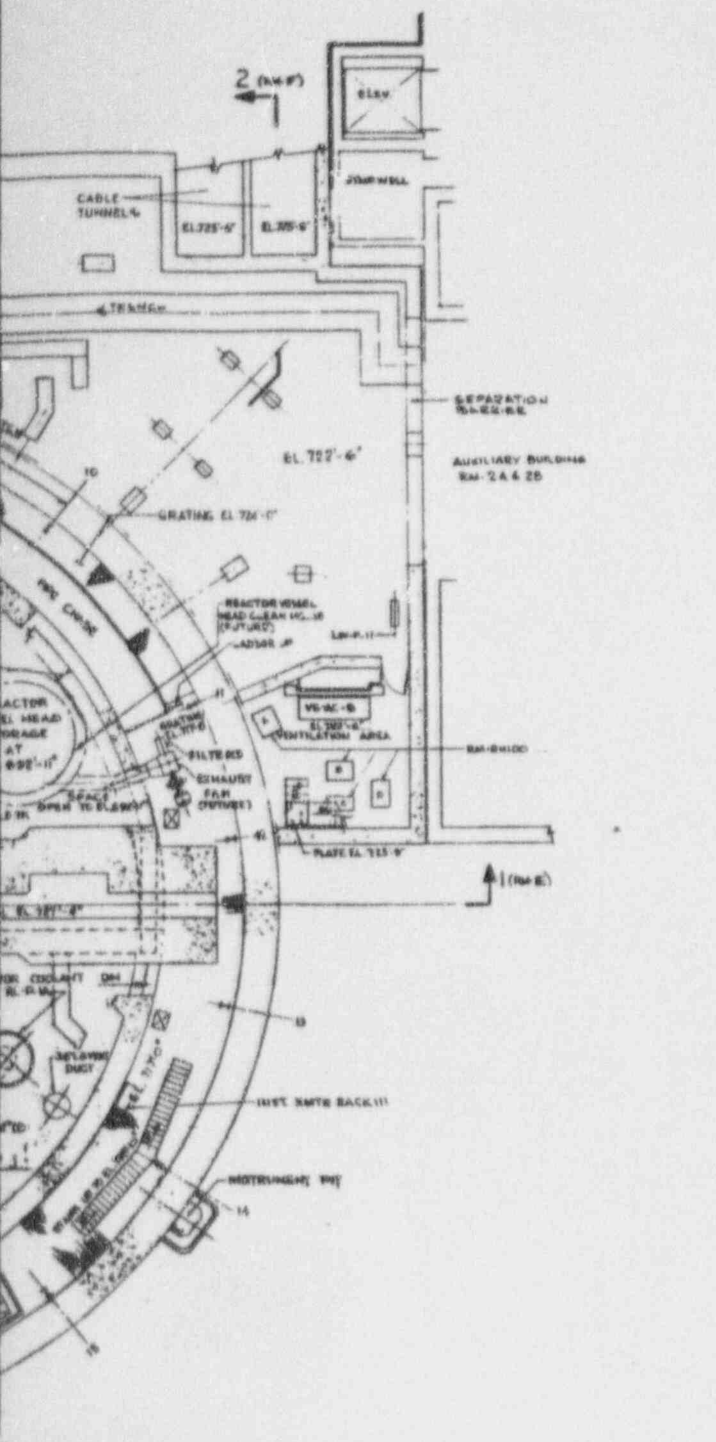
9503130375-03

FIGURE 5.1-2
(RM-1B, REV. 8)
REACTOR CONTAINMENT
GENERAL ARRANGEMENT EL. 738'-10"
BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT


PIPE TUNNEL FLOOR EL. 722'-6"
PIPE TRENCH CONT ON RM-1B
FLOOR EL. 722'-6"
SUMP PUMPS
REACTOR COOLANT PUMP RC-P-1B
INST. XMTR RACK 109
RECIRC. SPRAY COOLERS (4)
PIPE TUNNEL EL. 722'-6"
LOOP 2
CONT. DETACHED PLAN-RM-1B
PRESSURIZER RC-TK-1 (ABOVE)
STEAM GENERATOR SUPPORT (TYP)
PUMP CASING (TYP)
LOOP 3
REACTOR COOLANT PUMP SUPPORT (TYP)
REACTOR COOLANT PUMP RC-P-1C
OPEN TO EL. 707'-6"
RESIDUAL HEAT REMOVAL EXCHANGERS
INSTRUMENT PIT
STEAM GENERATOR RC-E-1A
PRIMARY LOOP ISOLATION MOV'S (TYP)
3 (RM-1G)

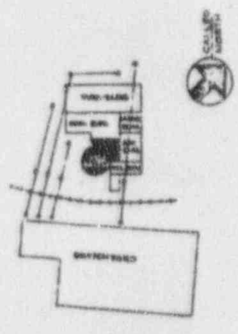
PLAN EL 718'-6"

REV. 0 (1/82)



ANSTEC APERTURE CARD

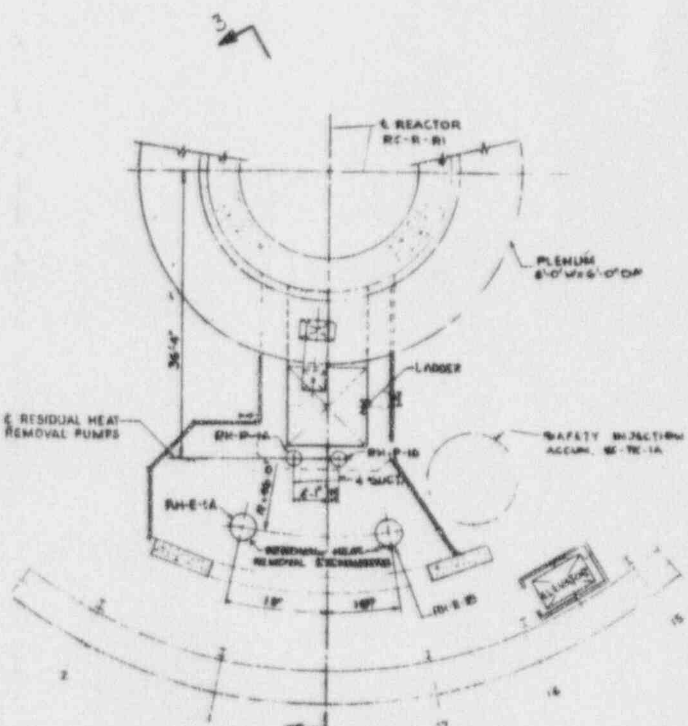
Also Available on Aperture Card



KEY PLAN

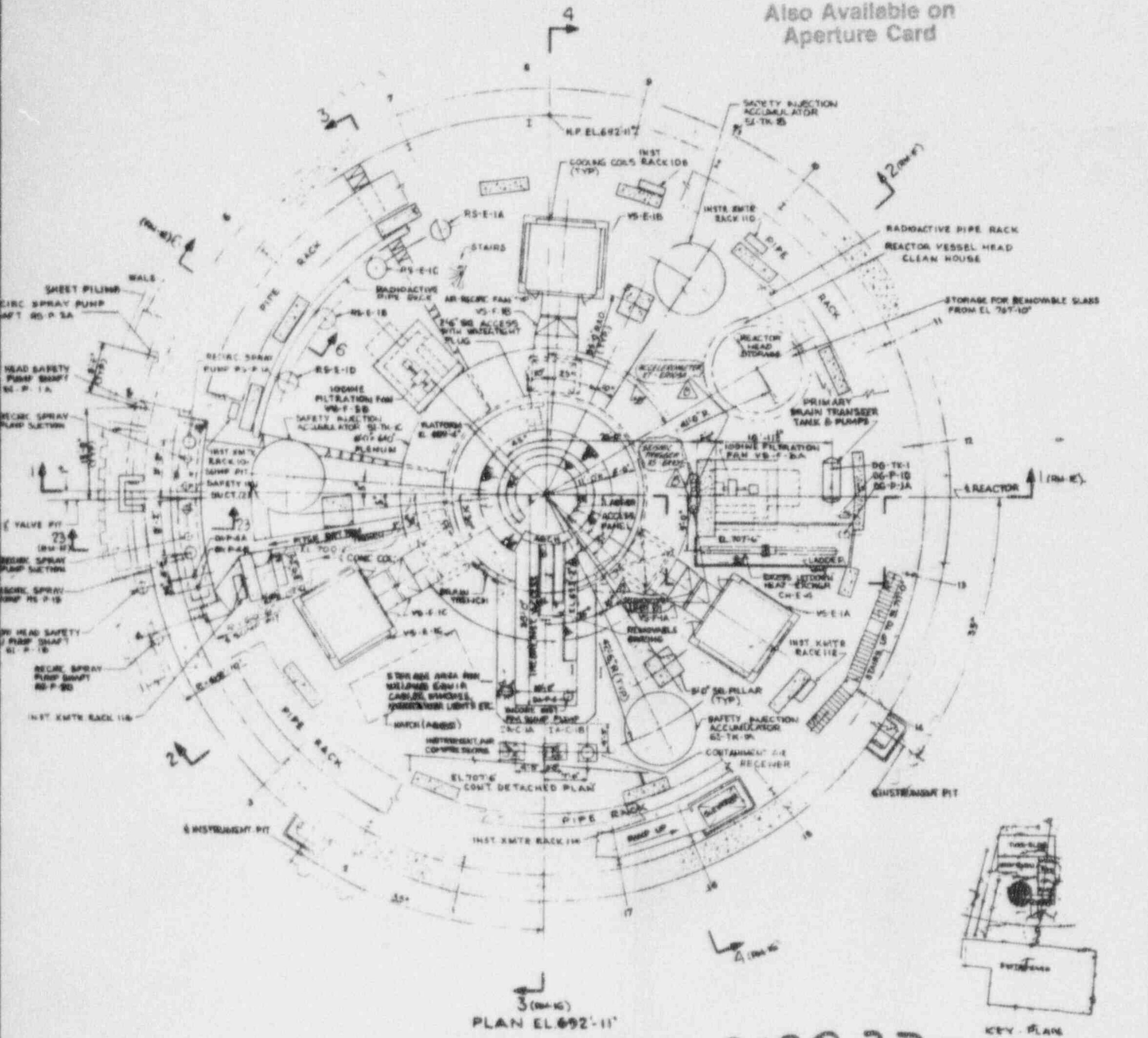
9503130375-04

FIGURE 5.1-3
(RM-1C, REV. 7)
REACTOR CONTAINMENT
GENERAL ARRANGEMENT EL. 718'-6"
BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT



3 (RH-16)
DETACHED PLAN
EL. 707'-6"

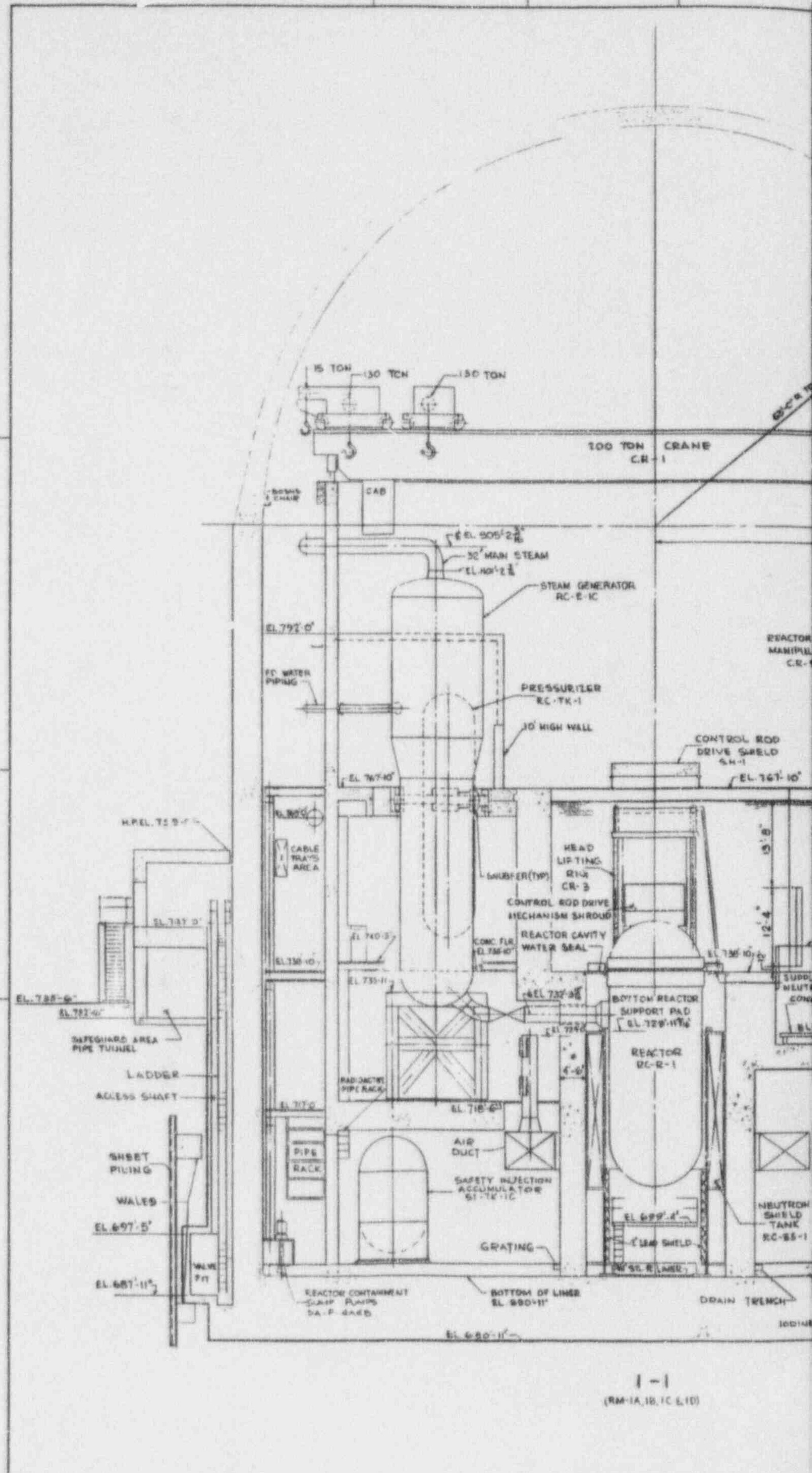
REV. 2 (1-84)



PLAN EL.692'-11"

9503130375-05

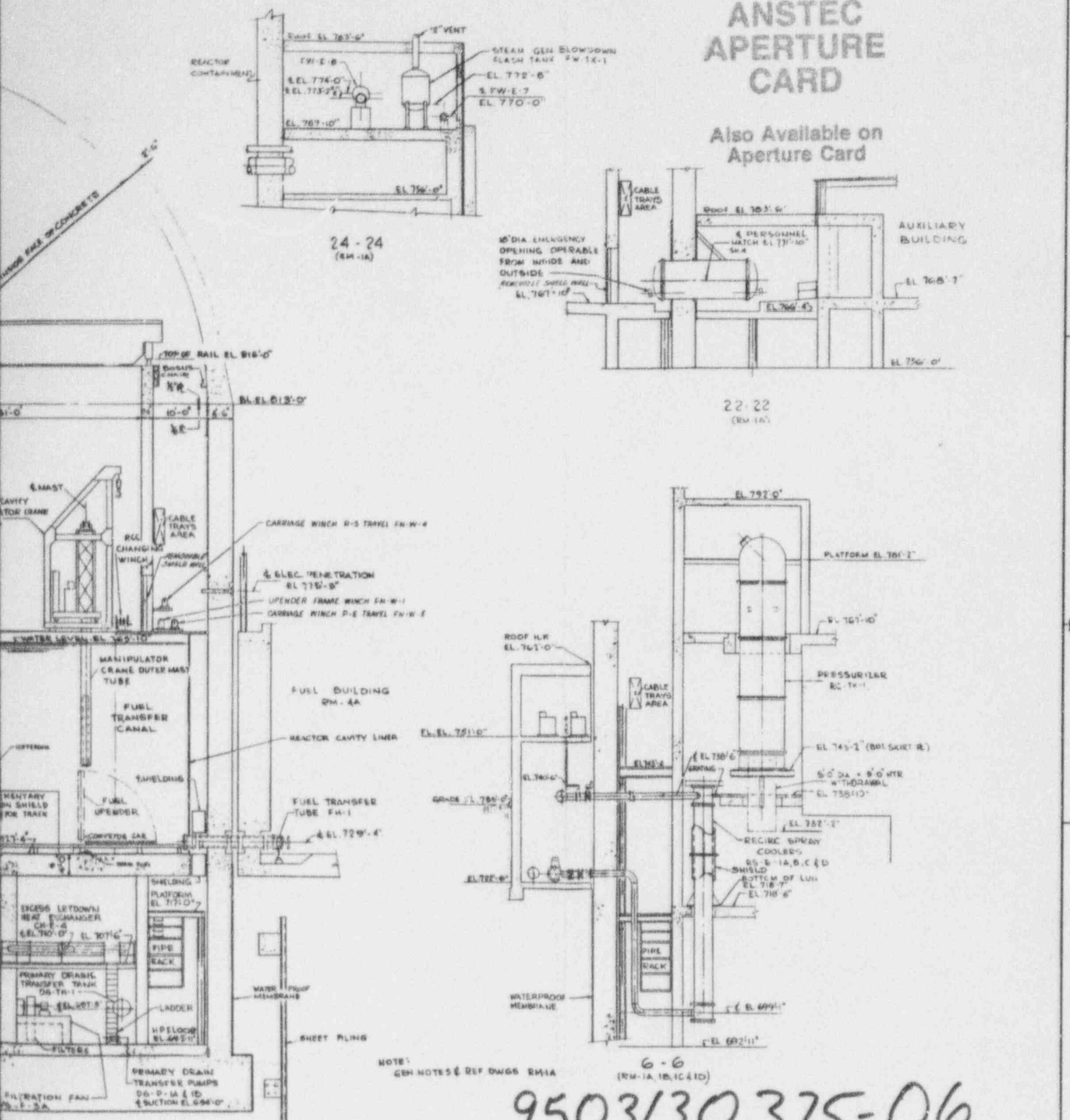
FIGURE 5.1-4
(RM-1D, REV. 8)
REACTOR CONTAINMENT
GENERAL ARRANGEMENT EL. 692' - 11"
BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT


15 TON
130 TON
130 TON
100 TON CRANE
CH-1
CAB
32" MAIN STEAM
STEAM GENERATOR
RC-E-1C
EL. 792'-0"
FC WATER PIPING
PRESSURIZER
RC-TK-1
10' HIGH WALL
CONTROL ROD DRIVE SHIELD
SH-1
EL. 767'-10"
H.P. EL. 759'-
CABLE TRAYS AREA
SNUBBER (TYP)
HEAD LIFTING RIG
CR-3
CONTROL ROD DRIVE MECHANISM SHROUD
REACTOR CAVITY WATER SEAL
BOTTOM REACTOR SUPPORT PAD
REACTOR
RC-R-1
EL. 738'-6"
SAFEGUARD AREA PIPE TUNNEL
LADDER
ACCESS SHAFT
RADIOACTIVE PIPE RACK
PIPE RACK
SHEET PILING
WALES
EL. 697'-5"
AIR DUCT
SAFETY INJECTION ACCUMULATOR
SI-TK-1C
GRATING
NEUTRON SHIELD TANK
RC-SS-1
VALVE PIT
REACTOR CONTAINMENT SUMP PUMPS
BOTTOM OF LINER
DRAIN TRENCH
1-1
(RM-1A, 1B, 1C & 1D)

REV.12 (1-94)

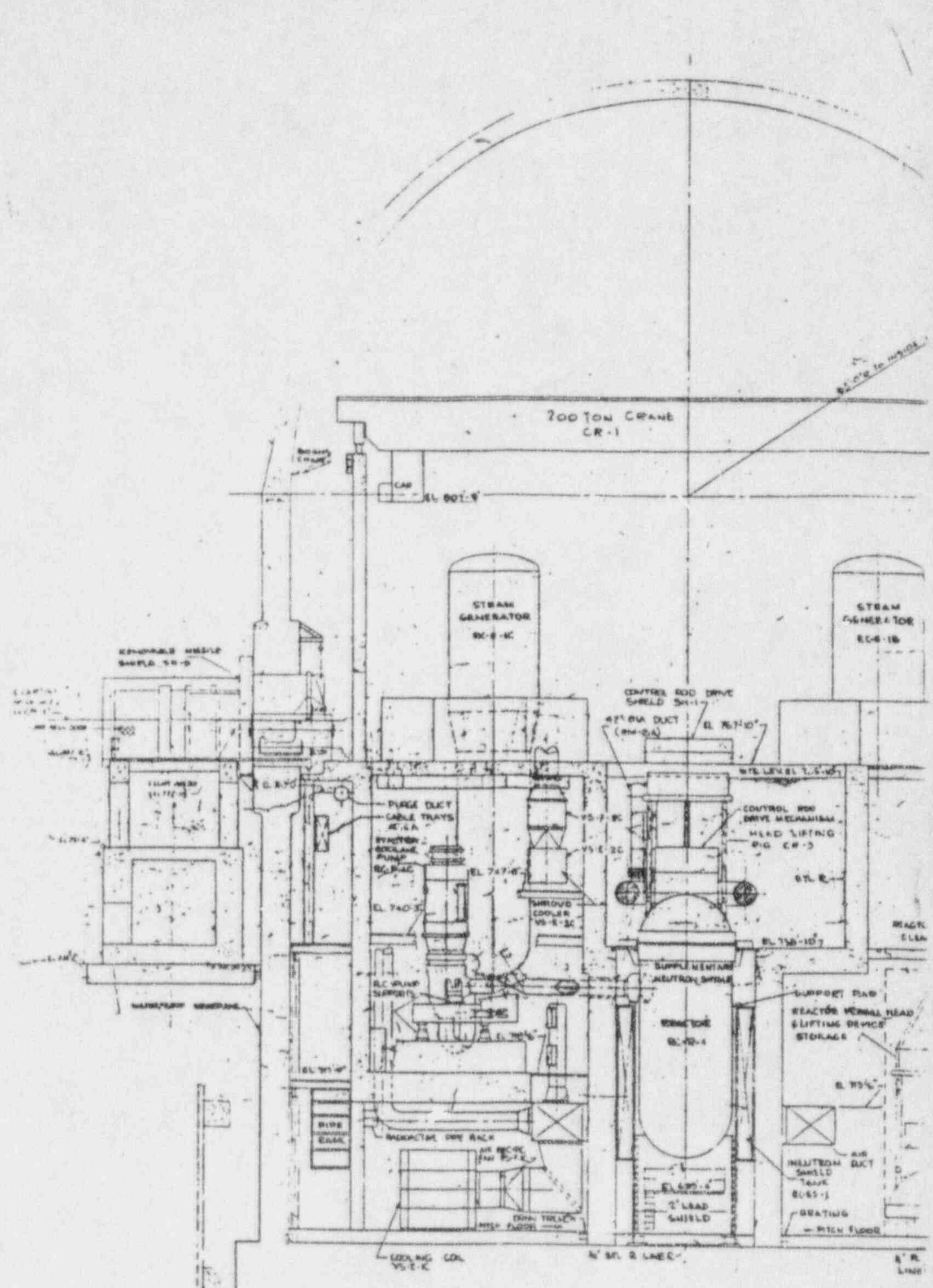
ANSTEC
APERTURE
CARD

Also Available on
Aperture Card

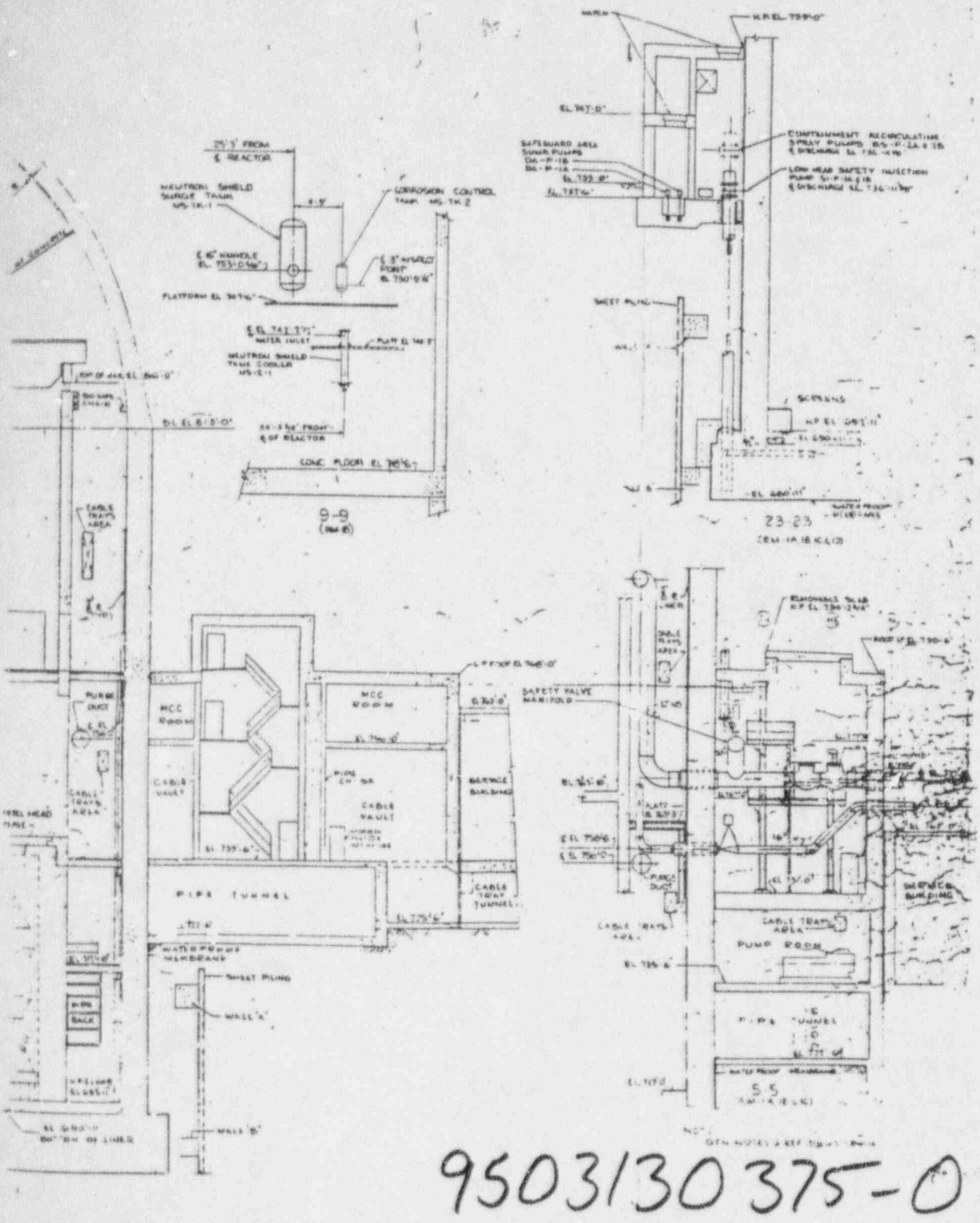


9503130375-06

FIGURE 5.1-5
(RM-1E)
REACTOR CONTAINMENT
SECTIONS 1-1, 6-6, 22-22 AND 24-24
BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT

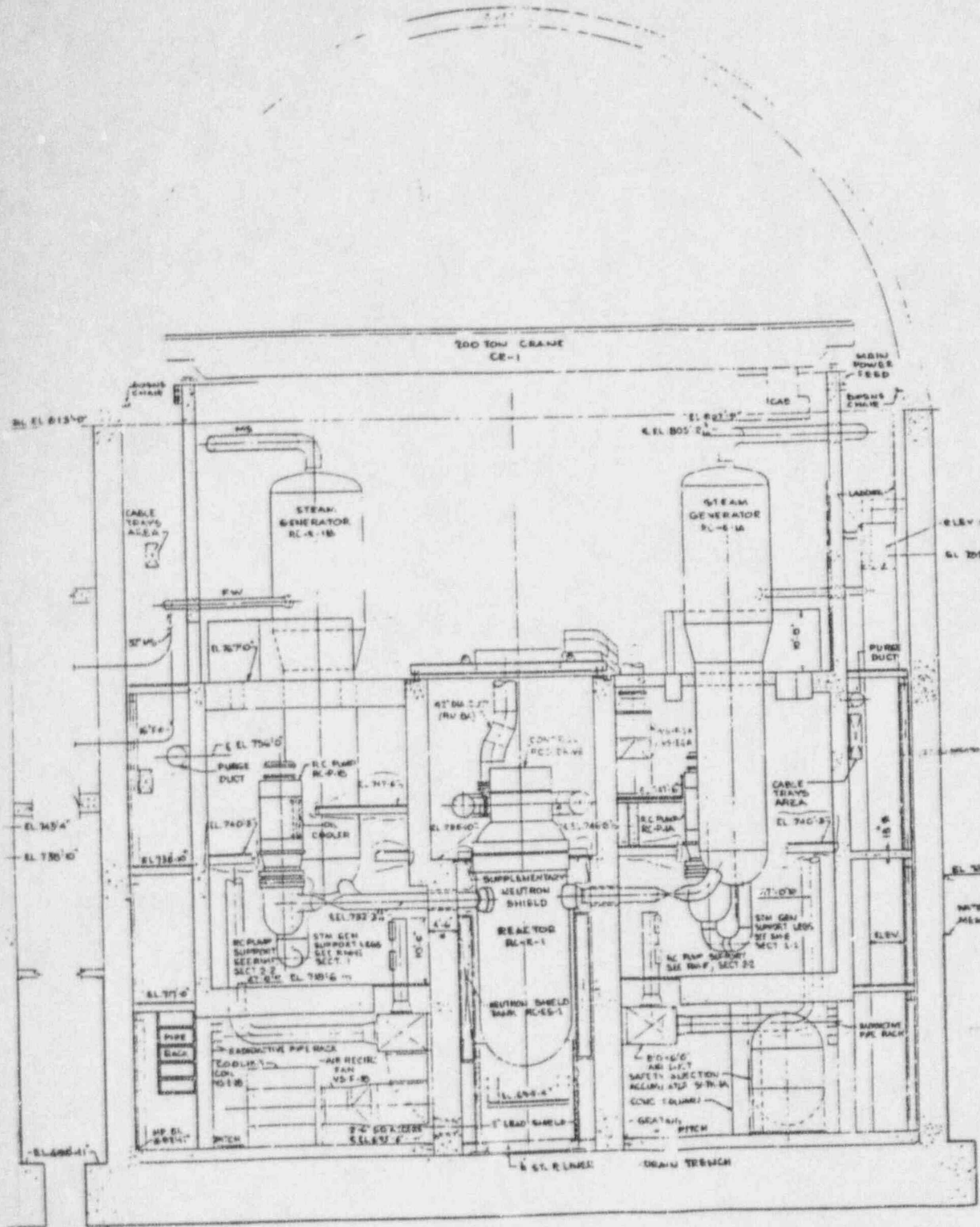

200 TON CRANE
CR-1
STEAM GENERATOR
STEAM GENERATOR
CONTROL ROD DRIVE SHIELD
PURGE DUCT
CABLE TRAYS
CONTROL ROD DRIVE MECHANISM
SHROUD COOLER
SUPPLEMENTARY NEUTRON SHIELD
SUPPORT PAD
REACTOR VESSEL HEAD & LIFTING DEVICE STORAGE
RADIOACTIVE PIPE RACK
2' LEAD SHIELD
GRATING
PITCH FLOOR
COOLING COIL

2 2



9503130375-07

FIGURE 5.1-6
(RM-1F, REV. 7)
REACTOR CONTAINMENT
SECTIONS 2-2, 5-5 AND 23-23
BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT

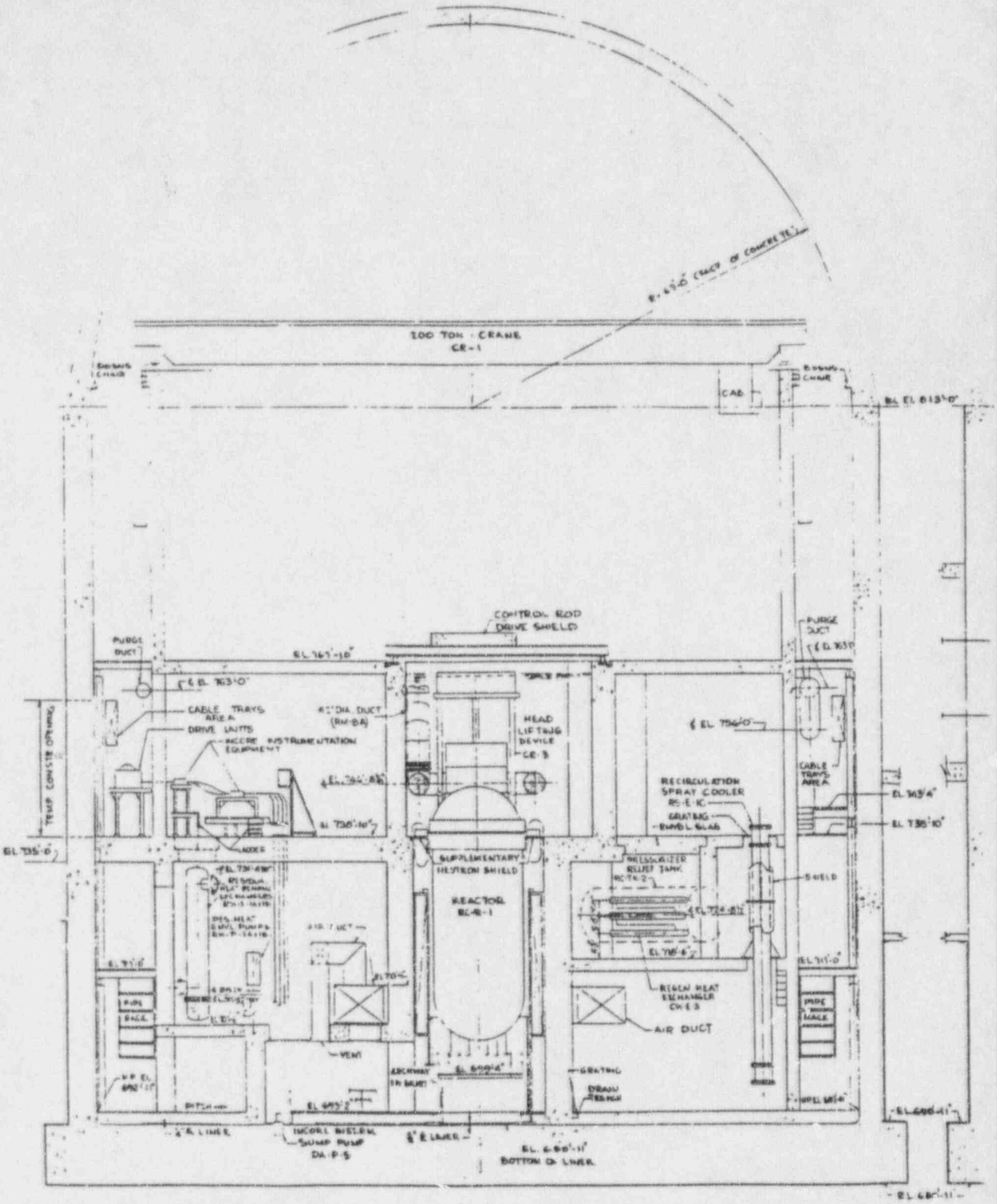


4-4

NOTES:

FIGURE 5.1-7
(RM-1G, REV. 7)
REACTOR CONTAINMENT
SECTIONS 3-3 AND 4-4
BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT

9503130375-08



3-3
(RM-1A 1B 1C & 1D)

(d) Descriptions of the containment failure sizes and locations.

Response:

As noted in Section 4.4-1, the Beaver Valley containments were believed to be very similar to the Surry containment which was analyzed in NUREG-1150. Upon confirmation of this similarity, it was determined that the probability distributions (containment failure and conditional probability of large vs. small failure) for pressure capacity developed for Surry in Reference 7.d could be used for Beaver Valley. As a result, no plant specific evaluation of containment pressure capacity was performed for Beaver Valley. In Reference 7.d, it was noted that a large hole or rupture is one for which the containment would depressurize in less than approximately 2 hours. It was also noted that large, dry containments would depressurize in 2 hours for hole sizes on the order of 0.3 to 0.5 ft^2. It was then stated that a small hole or leak should be of the order of 0.1 ft^2. A failure area of 1.0 ft^2 was identified as "definitely a large hole or rupture." Thus, in the CET quantification process, small and large failures are typified by failure areas of 0.1 ft^2 and greater than approximately 1.0 ft^2, respectively.

The Surry pressure capacity distribution is a composite of four expert analyses and a number of failure modes including hoop failure in the cylinder and dome, shear at the cylinder-basemat junction, liner tearing, and failure at the penetrations. The composite nature of the curve makes it difficult to identify specific failure locations at any given failure pressure. In Reference 7.d, it is noted that "failure location did not turn out to be important since any failure location except shear at the basemat-cylinder junction would result in a direct path to the outside."

As noted in Section 4.7 (see Table 4.7-7), all large containment failures were assumed to release fission products directly to the environment. Some ex-containment retention was credited only for small containment failures. Reference 7.d is as follows:

7.d Breeding, R.J., et al, "Evaluation of Severe Accident Risks: Quantification of Major Input Parameters, Experts' Determination of Structural Response Issues," NUREG/CR-4551, Vol. 2, Rev. 1, Part 3, March 1992

(e) A quantification of contributors to small, early containment failure.

Response:

Table 4.8-4 on pages 4.8-62 and 4.8-63 of the IPE submittal lists the non-guaranteed failure split fraction importance for the major contributors to the small, early containment failure and bypass release category group. The Level 2 top event failures and respective RCG II contributions that lead directly to a small, early containment failure or bypass are listed below:

Top Event BY (Containment bypassed prior to core damage)	= 16.3%
Top Event C1 (Containment failure prior to vessel breach)	= 72.1%
Top Event C2 (Containment failure at vessel breach)	= 13.0%
Top Event CE (Containment failure due to early H_2 burn)	= 0.0%

The reason that the total exceeds 100% is that there is an overlap of sequences which involve both C1 and C2 failures. This is due to the containment event tree structure that still inquires if a containment failure at vessel breach occurs as a result of a HPME, since these types of failures can lead to a large, early containment failure even though small containment isolation failures occurred prior to vessel breach. The majority of failures to this release category group (88.4%) come from small containment isolation failures and bypasses from the Level 1 analysis. Early hydrogen burns below a 12% concentration are not expected to fail the containment structure within 4 hours of vessel breach, and burns above a 12% concentration result in global detonations that lead directly to a large containment failure.

(f) The size assumed for a "large" bypass.

Response:

All large containment bypasses are governed by the Level 1 interfacing systems LOCA events or by a Level 2 induced steam generator tube rupture. The interfacing systems LOCA (i.e., containment bypass LOCA) is addressed in Section 3.1.3.6 of the IPE submittal. As discussed in this section, the LHSI/RCS pathway includes a single 6 in. diameter header that penetrates the containment at Penetration No. 61. The piping from the RCS connection to the normally open motor operated containment isolation valve located outside the containment is designed to withstand normal RCS pressure, as is the valve itself. The 10 in. diameter piping upstream of the isolation valve is not designed for high pressure and is predicted to fail when pressurized to RCS conditions. Break flow through a rupture in the 10 in. piping would be restricted by the flow areas associated with failed check valves, which could be almost as small as the flow areas associated with the LHSI relief valves (105 gpm total choked flow), but not more than that associated with the single 6 in. line through the containment penetration. For the induced SGTR, it was assumed that, if such an event were to occur, the primary system pressure would be high enough to lift the secondary side safety valves, which have a 10 in. diameter outlet, thus creating a containment bypass route during the time of core overheating and fission product release from the fuel. It should be noted that for the induced SGTR the associated minimum bypass area could be restricted by the number of tubes failed.

(g) The contribution of global detonation to conditional containment failure probability (large, early; small, early; and late).

Response:

The reference to "global detonation" is somewhat confusing since no such terminology is used in the submittal. As indicated in the response to Back-End Question 1, a global hydrogen concentration of 12% was used as a "benchmark" for the occurrence of a deflagration-to-detonation transition (DDT). In effect, it was assumed that localized conditions for DDT could be achieved if global concentrations reached 12%.

The percentage contributions of hydrogen burn/DDT to the various release category groups are as follows:

0.2% of RCG I - Large, Early Containment Failures and Bypasses
(based on importance of conditional split fraction LEF)

0.0% of RCG II - Small, Early Containment Failures and Bypasses
(DDTs are assumed to cause large containment failures)

11.3% of RCG III - Late Containment Failures
(based on combined importance of conditional split fractions C3A and C3C)

It should also be noted that containment failures at vessel breach (CET Top Events C2 and L2) due to HPME are due in part to hydrogen combustion. As indicated on page 4.6-15 of the IPE submittal, the distributions used for the pressure rise at vessel breach represent the combined effects of blowdown, hydrogen burning, direct containment heating, and ex-vessel steam explosion.

8. Because of the high contribution of Direct Containment Heating (DCH) to early overpressurization failures, it is not clear what role the induced hot leg failure depressurization plays in reducing potential DCH failures. Please explain.

Response:

As indicated in Table 4.6-4, induced hot leg failure is assumed to occur with a conditional probability of 0.72 when RCS pressure is at or near the setpoint pressure of the relief valves (i.e., PDSs beginning with the letter S). No induced hot leg failures are predicted at lower pressures because the natural circulation flow at lower pressures is insufficient to heat these components significantly.

Table 4.8-6 shows the combined impact on Release Category Group I (large, early containment failures and bypasses) of eliminating induced steam generator tube and hot leg failures. Since induced steam generator tube ruptures are predicted to occur at a much lower conditional probability than hot leg failures, nearly all of the 8.7% increase (9.0×10^{-7} per reactor year) in Release Category Group I frequency to 1.12×10^{-5} per reactor year results from elimination of the induced hot leg failures. On a conditional basis, eliminating induced failures increases the percentage of large, early containment failures and bypasses to approximately 5.3%.

This impact can also be estimated by "walking-through" the containment event tree (CET) quantification process. As noted on page 1.4-6 and in Table 1-3, the contribution to core damage frequency for PDSs where the pressure at UTAF is at the system setpoint is 3.77×10^{-5} per reactor year (or 17.6% of the CDF). Subtracting the containment bypass component (4.0×10^{-7} per reactor year) from this total, a core damage frequency of 3.73×10^{-5} per reactor year is processed by CET Top Event LS (Induced PORV Failure). For RCS pressure at the system setpoint, PORV failures are expected 50% of the time (conditional split fraction LS3 = 0.5). All PORV failures are expected to drop the RCS pressure out of the system setpoint range (see conditional split fraction RPV). Thus, only 1.87×10^{-5} per reactor year ($0.5 * 3.73 \times 10^{-5}$/reactor-year) remains at the system setpoint pressure beyond Top Event LS. Neglecting the minor impact of induced steam generator tube failures, a CDF of 1.34×10^{-5} per reactor year ($0.72 \times 1.87 \times 10^{-5}$/reactor-year) is converted to low pressure prior to vessel breach because of induced hot leg failures, eliminating this frequency from the potential for containment failure due to DCH effects caused by high pressure melt ejection. Given system setpoint pressure at the time of vessel breach, an upper bound estimate of the conditional probability of a large, early containment failure can be derived from the product (0.1) of conditional split fractions ME3 (0.92), C2S (0.1875), and L2S (0.589). Thus, if induced hot leg failures are eliminated, an upper bound increase in the frequency of large, early containment failures of 1.3×10^{-6} per reactor year ($0.1 * 1.34 \times 10^{-5}$/reactor-year) could be estimated. This result compares favorably with the actual sensitivity case which was summarized in Table 4.8-6.

9. Considering the expected similarities between Beaver Valley Units 1 and 2, had any differences been identified that had an impact on either of the Unit's Level 2 findings? If so, please discuss.

Response:

Beaver Valley Units 1 and 2 have containment buildings that are very similar in design and function, as can be seen in Tables 4.1-1, 4.1-2, and the Table shown on page 4.1-2. Therefore, the Unit 1 Level 2 back-end model made use of the logic and split fraction values from the back-end model used for Unit 2. As expected, there were no major differences identified that had an impact on either Units' Level 2 findings. The only difference worth mentioning is that the Unit 1 large containment bypass contribution was 11% of the RCG I total, while Unit 2 had only a 4% contribution. However, this difference was expected since the Level 1 interfacing systems LOCA initiating event frequencies for each unit were so different.

10. With respect to the hydrogen burn issues, please address the following:

(a) Have plant walkdowns been performed to determine the probable locations of hydrogen released into the Containment? Including the use of walkdowns, discuss the process used to assure that: (i) local deflagrations would not translate to detonations given an unfavorable nearby geometry, and (ii) the containment boundary, including penetrations, would not be challenged by hydrogen burns.

Response:

Section 4.1.2 discusses the walkdown that was performed for Beaver Valley Unit 1. As noted, relative to the hydrogen issue, the walkdown consisted of a general visual inspection of the containment geometry and "openness," and a more detailed inspection of the reactor cavity. As noted on page 4.1-5, "the configuration of structures and equipment inside the containment appears to be conducive to good air circulation. The steam generator and pressurizer cubicles and most compartments within the containment are open at their tops to the general containment atmosphere. The reactor vessel head laydown area on the bottom floor of the containment is completely open to the containment dome."

It should again be emphasized that because of the similarity between the Beaver Valley and Surry plants, the approach adopted for the BV1 IPE was to take full advantage of the severe accident analysis that had been performed for Surry in support of the NUREG-1150 study. Accordingly, many of the insights, conclusions, and numerical values for failure probabilities were taken from the Surry evaluation. As noted in the response to Back-End Question 1, detonations were not addressed for Surry.

As indicated in the response to Back-End Question 1, the possibility of deflagration to detonation transitions was addressed in the BV1 IPE submittal. This possibility was inferred from global conditions and a large containment failure was assumed given a DDT. This latter assumption is very conservative. In NUREG-1150, failure of the Sequoyah containment was predicted to be relatively unlikely even if DDT occurred in the ice condenser. A value of 0.1 was used for containment failure, given DDT. The ice condenser geometry and function (condensing steam) is much more conducive to DDT than the configurations of large, dry containments.

(b) Please identify potential reactor hydrogen release points and vent paths. Estimates of compartment free volumes and vent path flow areas should also be provided. Please specifically address how this information is used in your assessment of hydrogen pocketing and detonation. Your discussion (including important assumptions) should cover likelihoods of local detonation and potentials for missile generation as a result of local detonations.

Response:

There are four likely release "points" for the release of hydrogen from the RCS into the containment for the range of accident sequences that was considered in the IPE.

- Hot leg piping
- Cold leg piping
- Pressurizer Relief Tank (PRT) rupture disc
- Bottom head of the reactor vessel

Hot and cold leg piping failures release hydrogen into the lower compartment of the containment. For LOCA initiators, this release would be relatively slow. For thermally induced hot leg failures, the potential exists for a large quantity of hydrogen to be suddenly released into the containment. This phenomena is discussed in detail in the response provided for Back-End Question 1.

The PRT and its rupture disc are located in the lower compartment, which are located approximately 40 feet above the containment floor. The bottom head of the reactor vessel is located in the reactor cavity.

The following table provides estimates of the compartment free volumes.

Containment Compartment	Compartment Free Volume (ft^3)
Reactor Cavity/Instrument Tunnel	8,661
Lower Compartment	430,000
Upper Compartment	1,020,000
Annular Compartment	255,000

In the above table, the reactor cavity/instrument tunnel is specific for Beaver Valley Unit 1 (see Table 4.1-1). The remaining compartment volumes were taken from the Beaver Valley Unit 2 MAAP parameter file, as are the flowpath areas listed in the following table.

Pathway	Cross-sectional Flow Area (ft^2)
Flow area connecting cavity to lower compartment via instrument tunnel	52
Bypass flow area coupling cavity to lower compartment	10
Lower compartment to upper compartment	2,448
Lower compartment to annular compartment	4,360
Upper compartment to annular compartment	3,583

As noted in the response to Back-End Question 1, detonations per se are extremely unlikely since the source of energy required to directly initiate a detonation is extremely large. The transition of deflagrations to detonations (DDTs) has been addressed in the BV1 IPE and was discussed in the response to Back-End Question 1. The treatment of DDT in the BV1 is believed to be very conservative. Whenever conditions for DDT were predicted, a large failure of the containment was assumed, therefore, the potential for missile generation is irrelevant. The most likely region for a DDT to occur, if indeed any could occur, would be in the lower compartment region. It is not clear that the such events would fail the containment boundary. Hence, we are convinced that a more detailed treatment of detonations would show that the BV1 IPE results are conservative in its treatment.

11. How much are the contributions to the CDF from Steam Generator Tube Rupture (SGTR), and interfacing systems Loss Of Coolant Accidents (LOCAs)?

Response:

The contribution to the total CDF from steam generator tube ruptures is 3.45%. This is shown on Figure 1-3, located on page 1.4-5 of the IPE submittal. It should also be noted that after core damage has occurred with the RCS at system pressures (> 2,000 psia) and the steam generators dry, the possibility of induced SGTRs becomes a concern. Therefore, based on the Level 2 Top Event IS, another 0.17% of the total CDF results in induced SGTRs. Interfacing systems LOCAs contributed 0.52% to the total CDF, and is included with the "Other" initiating events on Figure 1-3.

BV1 IPE HUMAN RELIABILITY ANALYSIS (HRA) QUESTIONS

1. Pre-initiator human errors are stated as being evaluated separately and incorporated into each systems analysis as a specific cause for equipment inoperability. However, the submittal does not identify specific pre-initiator actions considered, nor does it discuss the plant-specific analysis conducted to support the quantification estimates of the Human Error Probabilities (HEP) for pre-initiator actions, nor their impact on the system unavailability. It is not clear, for example, whether HEPs were calculated for specific human errors, or whether component failure data was intended to include human-related failures. Please address the following:

   (a) Other PRAs have found pre-initiator human errors to be important and non-negligible contributors to core damage frequency; therefore, if pre-initiator (including restoration and miscalibration) human errors were not specifically addressed, please provide the basis for not including them as part of your analysis. If they were addressed as part of component failure data, please address the basis for your assumption that the data actually captures all the pre-initiator events at BV-1, and that it accurately reflects the impact of the pre-initiators.

   (b) If pre-initiators were specifically addressed, please discuss the following:

      i) If the actions were screened out, what was done to assure that the actions that were screened out actually did have a low contribution to system unavailability and, therefore, their contribution to CDF was indeed negligible?

      ii) Reviews of maintenance, test and calibration procedures for the systems and components modeled that were performed by the Systems Analysts.

      iii) Discussions that were held with appropriate plant personnel (e.g., Maintenance, Training, Operations) on the interpretation and implementation of the plant's test, maintenance and calibration procedures to identify and understand the specific actions, and the specific actions and the specific components manipulated when performing the maintenance, test or calibration task.

      iv) Consideration of plant-specific information such as: plant conditions (e.g., poor lighting), human engineering (e.g., labels, accessibility, etc.), performance by same crew, same time, adequacy of training, and adequacy of procedures in the quantification of pre-initiator events.

      v) How dependencies associated with pre-initiator human errors were addressed and treated. These dependencies could, for example, affect all of the human events simultaneously, or could only affect certain human events such that only a series of human events are determined to fail simultaneously (e.g., complete dependence may be assumed for miscalibration of all reactor water level sensors). Please provide examples demonstrating how the dependencies were treated.

Response:

The IPE submittal did not systematically address pre-initiator human errors in the PRA models or component failure data. The only one that was felt to be important in the Surry analysis (NUREG-1150) was mis-calibration errors associated with the RWST level transmitters. Therefore, for the IPE submittal the PRA staff carefully considered this situation at Beaver Valley Unit 1. Upon review of past operating experience at Beaver Valley Unit 1 from January 1980 through December 1988 (i.e., the freeze dates for the data collection used in the IPE), no such errors occurred for these instruments. Moreover, it was felt by the PRA staff that with the staggered instrument calibrations, and independent verification and restoration checks performed after maintenance, test, and calibration procedures, the occurrence of any such errors would be small when compared to the total failure of a single train. Therefore, while these type of errors were specifically looked at, they were not included in the PRA model (Top Event OR) since they were not expected to have a significant numerical impact on the system model. The PRA model for Top Event OR did, however, account for out-of-calibration errors of the individual RWST level transmitters by including an exposure time equal to half of their monthly channel functional test surveillance period in with the mission time for the basic event equation .

Another type of pre-initiator human error looked at in the Surry analysis was the failure to restore valves to their proper alignment after a pump test. The Beaver Valley Unit 1 IPE did not explicitly include these types of errors in the submittal. A recent review of past operating experience in the plant problem report database did not discover any such errors for the time period of interest (i.e., January 1980 through December 1988) while the plant was operating in Mode 1. Additionally, it is felt that, since the status of key safety-related components are independently verified to be in their normal system alignment position every shift, the occurrence of such errors would be small, certainly less than the frequency of hardware failures due to all other causes. Likewise, important standby systems; e.g., the HHSI, LHSI, quench spray, and recirculation spray systems, are verified every 31 days to insure that each valve (manual, power operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position is in its correct position. It can be argued that the MGL method for quantifying common cause failures implicitly models potential pre-initiator errors that could impact multiple trains of systems. The PRA staff concluded that it was not necessary to model pre-initiator errors that could impact multiple trains because of the thorough treatment of common cause failures already considered.

Furthermore, it should be noted that system mis-alignments and human errors which resulted in a reactor trip are included in with the appropriate initiating event frequencies for such transients. These types of plant trips are included in with the Unit 1 reactor trip events presented in Table 3.3.1-6 of the IPE submittal. An example of this would be Event No. 31, listed on page 3.3-26, in which the main steam trip valves closed due to a test engineer inadvertently isolating the air supply. This event was included with the closure of all main steam isolation valves (AMSIV) initiating event frequency.

2. While the Success Likelihood Index Methodology (SLIM)-based analysis inherently provides a means for systematic incorporation of subjective evaluation of plant-specific performance shaping factors, the judgment of the assessment teams is influenced by the number and type of personnel in the group, and information that is prepared and presented to them. Please identify the number of groups (teams) which participated in the ratings, and the number and type of personnel (i.e., PRA/HRA analyst, operations, maintenance or training personnel) within each group.

Response:

The BV1 personnel which evaluated the human reliability actions consisted of one group with the following team members:

- 1 - PRA/HRA analyst
- 1 - PRA/HRA analyst (former SRO license trainee and STA)
- 1 - Beaver Valley Unit 1 senior nuclear operations instructor (former SRO)
- 1 - Beaver Valley Unit 1 licensed reactor operator

Because of scheduling conflicts and time limitations, it was not practical for the entire group to evaluate all of the human actions. Therefore, only the most important actions deemed by the HRA analysts were rated by the entire group. These actions which were evaluated by this group are shown in Table 3.3.3-6 as having a "Yes" in the "Action Rated by Operations & Training" column. This group was given a brief description of the action to be performed, the scenario and prior conditions for the action, available procedures and time frame required to perform the action, for each human reliability action rated by this group.

Actions not rated by operations and training personnel (i.e., shown in Table 3.3.3-6 as having a "No" in the "Action rated by Operations & Training" column) were analyzed by the HRA analysts using similar previously evaluated actions by the group as a guide, or using the Beaver Valley Unit 2 HRA as a guide for similar actions, or both. The Beaver Valley Emergency Operating Procedures are based on Westinghouse Owners Group Emergency Response Guidelines, and have similar actions between the Units in response to an accident. Therefore, similar actions which were evaluated by Beaver Valley Unit 2 personnel were slightly modified by the PRA/HRA analysts for Beaver Valley Unit 1 actions. The Beaver Valley Unit 2 HRA team used the same PRA/HRA analysts used for Unit 1, different operations and training personnel, and an additional PLG HRA analyst.

3. The submittal emphasizes the strength of the SLIM-based methodology for addressing dependencies among post-initiator response (dynamic) actions through the subjective evaluation process which considers actions in the context of the scenarios in which they are imbedded. The submittal states (in Table 3.3.3-2, Sheet 2 of 7) that, "if necessary, some strongly dependent failures may be accounted for by specific split fractions in event trees". No information is provided as to what constitutes a strongly dependent failure, or the criteria used to identify one. Please identify the criteria used to identify "strongly dependent" failures, and identify if any were considered in the HRA analysis. Provide examples of how

dependencies, and the level of dependency, were factored into the HEP quantification and addressed between operator actions in separate top events in the event trees.

Response:

The SLIM approach to human error rate quantification considers dependencies between actions occurring in the same accident sequence via the second performance shaping factor; i.e., the one for Significant Preceding and Concurrent Actions. This factor, and the others, are evaluated by the plant operating staff and the PRA analysts for every dynamic action. The influence on the final human error rate of this action is then determined by Equation 3.3.3.1 of the submittal for the failure likelihood index and the calibration curves.

When dynamic actions are dependent, the human error rate assigned to the second action should be made dependent on the outcome of the first action. For a particular sequence, the success or failure of the first action can be inferred by the status of the top event in which it is modeled. Then the error rate for the second action in the sequence can be made dependent on the status of the top event in which the first action appears. The split fraction assignment logic used by the event tree quantification code (RISKMAN) is structured so that the split fraction for the top event which contains the second operator action then reflects this dependency.

For example, in the event of a loss of auxiliary feedwater two actions were identified in the event sequence diagrams and modeled for restoring core cooling; i.e., restoration of main feedwater and initiation of bleed-and-feed cooling. These actions are directed by the same emergency procedure. Therefore, they were judged dependent.

The first action is modeled via Top Event OF. The second action is modeled via Top Event OB. In this case, bleed and feed cooling is only required if restoration of main feedwater is unsuccessful. However, there are two categories of reasons for failing to restore main feedwater. The necessary hardware may be unavailable (i.e., Top Event MF fails), or the operating staff may have failed to perform the restoration (i.e., Top Event OF fails).

Since the actions in Top Events OF and OB are dependent, the PRA analysts concluded that separate error rates for Top Event OB should be calculated depending on the status of Top Event OF. The different error rates are reported in Table 3.3.3-5 as ZHEOB1 (1.22×10^{-3}) and ZHEOB2 (1.39×10^{-2}). These different error rates are then used in the quantification of the different split fractions for Top Event OB. The split fraction assignment rules for Top Event OB are then made dependent on the status of Top Event OF. If Top Event OF fails, only Top Event OB split fractions which use ZHEOB2 are assigned. Conversely, if Top Event OF succeeds but Top Event MF fails so that bleed and feed is still required, only split fractions which use ZHEOB1 are assigned.

Numerous other event tree top events which consider multiple dynamic actions are considered in the Beaver Valley Unit 1 IPE models. These can be seen in Table 3.3.3-5. For example, seven different operator action error rates are used for Top Event CD (i.e., for cooldown and depressurization) depending on the specific sequence conditions that exist. The different error rates are used in different split fractions for Top Event CD, and the appropriate split fractions are then selected during event tree quantification based on the split fraction assignment logic.

Similarly, five error rates are used for the actions in Top Event MU and three are used for those in Top Event SL.

Actions appearing in the same accident sequence are identified as strongly dependent if they are directed at the same goal, guidance is provided by the same procedure, and time period in which the actions are to occur are roughly the same time frame. Actions directed at the same goal but separated by several hours in time, are not said to be strongly dependent. No explicit numerical criteria were used for assigning the dependence between two actions as strong. Rather, the PRA analysts used judgment on a case-by-case basis to determine whether the above stated criteria are met.

For strongly dependent actions in the same sequence, it is recognized that the impact of the dependency between actions on the human error rate for the second action may be more pronounced than can be realized through the linear equation for combining performance shaping factors. For such dependencies, the PRA analysts, in many cases, decide to take no credit for the second action; i.e., assign an error rate of 1.0 to it. In such cases, the SLIM quantification model is omitted as the limitations of the model, in extreme cases, are acknowledged. Then during sequence quantification, the split fraction assigned to the top event, which accounts for the second human action, is set to 1.0 for that sequence.

One example of strongly dependent actions modeled for Beaver Valley Unit 1 involves the actions to initiate recirculation from the sump following a small LOCA (Top Event OR) and the action to align for long-term makeup to the RWST (Top Event MU) given recirculation from the containment sump is unavailable. In the split fraction assignment logic for Top Event MU, when Top Event OR fails earlier in the sequence, no credit was taken for Top Event MU; i.e., effectively the operator error rate was set to 1.0 by assigning a split fraction with a value of 1.0.

A second example is that for initiation of manual control rod insertion (Top Event RI) and emergency boration (Top Event OA) during an ATWS following attempts by the operators to manually initiate a reactor trip (Top Event OT). If Top Event OT fails, error rate ZHERI2, which has a value of 1.0, was used for Top Event RI. Also, if Top Event OT fails, no credit was taken for emergency boration via Top Event OA. These dependencies between the three actions were accounted for during event tree quantification by the split fraction assignment logic.

A final point is that the thought process used to apply the event tree linking methodology of RISKMAN is that all split fractions for both hardware failures and human errors are presumed to be dependent until proven otherwise. The identification of dependencies between split fractions is a central task in constructing a RISKMAN PRA model. When completed, all of the dependencies modeled are evident in the split fraction assignment rules files.

4. Timing of operator actions is specifically addressed in the qualitative and quantitative analysis performed in the evaluation of post-initiator actions. The submittal notes that there is a "relatively well-defined time window available for successful operator response". It also notes that timing determines important factors that influence the operators' ability to diagnose the problem, decide what actions are appropriate, and complete those actions within the required

time window. However, little detail as to how time required to perform was actually evaluated, or whether additional factors may have been important for out-of-Control Room actions. Thus, it is not clear that walkdowns were performed for HRA purposes and walkdown-time measurements were taken for time-critical actions or arrived at by simulator runs; whether assumptions about accessibility, availability of tools, etc., were verified by walk-throughs or "simulations" of operator actions in the plant, and how environmental factors and other physiological or psychological "stressors" were accounted.

(a) Discuss by way of examples how such factors were addressed for operator actions, especially for the out-of-Control Room actions.

Response:

Walkdowns and walkdown-time measurements were not performed specifically for the HRA. The specific time available for operator actions to be performed are located on Table 3.3.3-5. These times were either calculated by thermal-hydraulic analyses, past plant experience, other reference analyses, or by simulator runs. In addition to the time window available, the SLIM process also accounts for psychological and cognitive conditions of the operators based on existing procedures, training, and stress factors. It is in these ratings of procedures and training that assumptions about accessibility and availability of tools are accounted for by actual plant procedure implementation experience. This knowledge also provides a sense of how much time is required to perform actions carried out in past experiences, which can be compared to the time frame available. The feasibility of performing each action within the time frame available was discussed among the HRA group before performing the qualitative rankings. Environmental factors and other physiological or psychological "stressors" were accounted for in the stress performance shaping factor rankings.

Examples can be seen in Table 3.3.3-6, e.g., ZHEAF1 and ZHECD4.

ZHEAF1 was rated as a 5 in its procedure PSF, and also as 5's in training and stress. These ratings were evaluated by the HRA group. The time frame available was calculated by thermal-hydraulic analysis based on the time for a steam generator to dryout to 10% on the wide range level indicator with no feedwater flow available. Since this human action is the same action used during surveillance testing or is part of normal training, the operators have a good basis on which to judge accessibility and availability of tools. They also know how much time is required to complete the action and any stress associated with the action through past experience.

Another example, ZHECD4 was rated as being 8's in both procedures and training, and as a 10 in the stress rating. Once again these actions were rated by the HRA group, however the time frame available was based on simulator runs to see how fast the operators could cooldown and depressurize the RCS to below 212 °F before the RWST empties, given an initial 800 gpm SGTR with a stuck-open SG atmospheric steam dump valve. Only vague guidance exists in procedures to perform this action, since it requires local manipulation of steam dump valves to cooldown during the loss of AC orange power. Also this is a non-routine action, but is an option in annual or biannual training.

Even though there is sufficient time available to complete the action at normal speed and to verify results, the pre-existing conditions and local environment in which to accomplish the action, puts tremendous physiological and psychological stresses on the operator. These undue stresses are reflected in the stress rating of 10.

(b) Identify which recovery actions are out-of-Control Room actions.

The following human actions are performed outside of the control room. A brief description of the action to be performed is provided in Table 3.3.3-5 and Table 3.3.3-10.

ZHEAF1	ZHECD7	ZHEIA3	ZHEOF5	ZHERE8
ZHEBV1	ZHECI1	ZHEIA4	ZHERE1	ZHERE9
ZHEBV3	ZHECT1	ZHEIC1	ZHERE2	ZHEREA
ZHECC1	ZHEDF1	ZHEIC2	ZHERE3	ZHERED
ZHECC2	ZHEFL4	ZHEIC3	ZHERE4	ZHEREE
ZHECD2	ZHEHH1	ZHEMA1	ZHERE5	ZHEREH
ZHECD4	ZHEIA1	ZHEMA2	ZHERE6	ZHESL2
ZHECD5	ZHEIA2	ZHEOF3	ZHERE7	ZHESL3

It should be noted that, although the description in Table 3.3.3-5 for action ZHEIA2 states that it is performed from the Control Room, the action was actually evaluated for starting the diesel driven air compressor outside the Control Room.

(d) Descriptions of the containment failure sizes and locations.

Response:

As noted in Section 4.4-1, the Beaver Valley containments were believed to be very similar to the Surry containment which was analyzed in NUREG-1150. Upon confirmation of this similarity, it was determined that the probability distributions (containment failure and conditional probability of large vs. small failure) for pressure capacity developed for Surry in Reference 7.d could be used for Beaver Valley. As a result, no plant specific evaluation of containment pressure capacity was performed for Beaver Valley. In Reference 7.d, it was noted that a large hole or rupture is one for which the containment would depressurize in less than approximately 2 hours. It was also noted that large, dry containments would depressurize in 2 hours for hole sizes on the order of 0.3 to 0.5 ft^2. It was then stated that a small hole or leak should be of the order of 0.1 ft^2. A failure area of 1.0 ft^2 was identified as "definitely a large hole or rupture." Thus, in the CET quantification process, small and large failures are typified by failure areas of 0.1 ft^2 and greater than approximately 1.0 ft^2, respectively.

The Surry pressure capacity distribution is a composite of four expert analyses and a number of failure modes including hoop failure in the cylinder and dome, shear at the cylinder-basemat junction, liner tearing, and failure at the penetrations. The composite nature of the curve makes it difficult to identify specific failure locations at any given failure pressure. In Reference 7.d, it is noted that "failure location did not turn out to be important since any failure location except shear at the basemat-cylinder junction would result in a direct path to the outside."

As noted in Section 4.7 (see Table 4.7-7), all large containment failures were assumed to release fission products directly to the environment. Some ex-containment retention was credited only for small containment failures. Reference 7.d is as follows:

7.d Breeding, R.J., et al, "Evaluation of Severe Accident Risks: Quantification of Major Input Parameters, Experts' Determination of Structural Response Issues," NUREG/CR-4551, Vol. 2, Rev. 1, Part 3, March 1992

(e) A quantification of contributors to small, early containment failure.

Response:

Table 4.8-4 on pages 4.8-62 and 4.8-63 of the IPE submittal lists the non-guaranteed failure split fraction importance for the major contributors to the small, early containment failure and bypass release category group. The Level 2 top event failures and respective RCG II contributions that lead directly to a small, early containment failure or bypass are listed below:

Top Event BY (Containment bypassed prior to core damage)	= 16.3%
Top Event C1 (Containment failure prior to vessel breach)	= 72.1%
Top Event C2 (Containment failure at vessel breach)	= 13.0%
Top Event CE (Containment failure due to early H_2 burn)	= 0.0%

The reason that the total exceeds 100% is that there is an overlap of sequences which involve both C1 and C2 failures. This is due to the containment event tree structure that still inquires if a containment failure at vessel breach occurs as a result of a HPME, since these types of failures can lead to a large, early containment failure even though small containment isolation failures occurred prior to vessel breach. The majority of failures to this release category group (88.4%) come from small containment isolation failures and bypasses from the Level 1 analysis. Early hydrogen burns below a 12% concentration are not expected to fail the containment structure within 4 hours of vessel breach, and burns above a 12% concentration result in global detonations that lead directly to a large containment failure.

(f) The size assumed for a "large" bypass.

Response:

All large containment bypasses are governed by the Level 1 interfacing systems LOCA events or by a Level 2 induced steam generator tube rupture. The interfacing systems LOCA (i.e., containment bypass LOCA) is addressed in Section 3.1.3.6 of the IPE submittal. As discussed in this section, the LHSI/RCS pathway includes a single 6 in. diameter header that penetrates the containment at Penetration No. 61. The piping from the RCS connection to the normally open motor operated containment isolation valve located outside the containment is designed to withstand normal RCS pressure, as is the valve itself. The 10 in. diameter piping upstream of the isolation valve is not designed for high pressure and is predicted to fail when pressurized to RCS conditions. Break flow through a rupture in the 10 in. piping would be restricted by the flow areas associated with failed check valves, which could be almost as small as the flow areas associated with the LHSI relief valves (105 gpm total choked flow), but not more than that associated with the single 6 in. line through the containment penetration. For the induced SGTR, it was assumed that, if such an event were to occur, the primary system pressure would be high enough to lift the secondary side safety valves, which have a 10 in. diameter outlet, thus creating a containment bypass route during the time of core overheating and fission product release from the fuel. It should be noted that for the induced SGTR the associated minimum bypass area could be restricted by the number of tubes failed.

(g) The contribution of global detonation to conditional containment failure probability (large, early; small, early; and late).

Response:

The reference to "global detonation" is somewhat confusing since no such terminology is used in the submittal. As indicated in the response to Back-End Question 1, a global hydrogen concentration of 12% was used as a "benchmark" for the occurrence of a deflagration-to-detonation transition (DDT). In effect, it was assumed that localized conditions for DDT could be achieved if global concentrations reached 12%.

The percentage contributions of hydrogen burn/DDT to the various release category groups are as follows:

0.2% of RCG I - Large, Early Containment Failures and Bypasses
(based on importance of conditional split fraction LEF)

0.0% of RCG II - Small, Early Containment Failures and Bypasses
(DDTs are assumed to cause large containment failures)

11.3% of RCG III - Late Containment Failures
(based on combined importance of conditional split fractions C3A and C3C)

It should also be noted that containment failures at vessel breach (CET Top Events C2 and L2) due to HPME are due in part to hydrogen combustion. As indicated on page 4.6-15 of the IPE submittal, the distributions used for the pressure rise at vessel breach represent the combined effects of blowdown, hydrogen burning, direct containment heating, and ex-vessel steam explosion.

8. Because of the high contribution of Direct Containment Heating (DCH) to early overpressurization failures, it is not clear what role the induced hot leg failure depressurization plays in reducing potential DCH failures. Please explain.

Response:

As indicated in Table 4.6-4, induced hot leg failure is assumed to occur with a conditional probability of 0.72 when RCS pressure is at or near the setpoint pressure of the relief valves (i.e., PDSs beginning with the letter S). No induced hot leg failures are predicted at lower pressures because the natural circulation flow at lower pressures is insufficient to heat these components significantly.

Table 4.8-6 shows the combined impact on Release Category Group I (large, early containment failures and bypasses) of eliminating induced steam generator tube and hot leg failures. Since induced steam generator tube ruptures are predicted to occur at a much lower conditional probability than hot leg failures, nearly all of the 8.7% increase (9.0×10^{-7} per reactor year) in Release Category Group I frequency to 1.12×10^{-5} per reactor year results from elimination of the induced hot leg failures. On a conditional basis, eliminating induced failures increases the percentage of large, early containment failures and bypasses to approximately 5.3%.