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March 15, 1991
ST-HL-AE-3682
File No.: G25, G3.08

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

South Texas Project Electric Generating Station
Units 1 & 2
Docket Nos. STN 50-498, STN 50-499
Additional Information on the External Events
Analysis in the Probabilistic Safety Assessment
South Texas Project Electric Generating Station (STPEGS)

Reference: (1) Letter from the NRC (Mr. George F. Dick, Jr.) to Mr. Donald P. Hall dated October 18, 1990 (ST-AE-HL-92586).

(2) Letter from Mr. M. A. McBurnett to Document Control Desk dated June 19, 1990 (ST-HL-AE-3478).

The purpose of this letter is to provide responses and discussion on all outstanding questions and issued resulting from the review of the South Texas Project (STP) Probabilistic Safety Assessment (PSA) by the NRC and its contractor, Sandia National Laboratory (SNL). With the provision of the responses in the attachments to this letter, Houston Lighting & Power Company (HL&P) considers all outstanding items complete, pending final review by the NRC.

By Reference 1 the NRC identified six (6) questions regarding external events other than fire. Attachment 1 provides the responses to the six (6) questions.

By Reference 2, HL&P provided information previously requested by the NRC and its contractor regarding the STP PSA internal events analysis. As the result of subsequent discussions between Mr. Tim Wheeler of SNL and Mr. R. P. Murphy of our staff on the response to item IE7, HL&P initiated a more detailed and quantitative evaluation of the "V" sequence loss of coolant accident (LOCA) for the STPEGS plant design. Attachment 2 provides the results of this evaluation. In addition, comments made by Mr. Wheeler on the new evaluation and HL&P responses to these comments are included as an addendum to Attachment 2.





A001

On October 24 & 25, 1990, the NRC and its contractor, SNL, visited the STP site for a walkdown of selected plant features related to the internal fire analysis completed as a part of the PSA. Discussions during the plant walkdown identified three (3) items on which HL&P was to provide additional clarification. Attachment 3 provides that additional discussion.

If you should have any questions on this matter, or the attachments, please contact me at (512) 972-7298 or Mr. R. P. Murphy at (512) 972-8919.

AWHarrison

A. W. Harrison
Manager,
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SDP/kmd

Attachment: (1) Nonfire-Related Questions
(2) South Texas V-Sequence Analysis
(3) Fire-Related Questions

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ATTACHMENT 1

ATTACHMENT 1
ST-HL-AE-3682
PAGE 1 OF 20

# NONFIRE-RELATED QUESTIONS

## Question 1

Discuss the basis for generic initiating event frequencies used in the spatial interaction analyses. For example, for flood analysis, discuss "What is a pipe section?" and "How the failure frequencies of $8.0 \times 10^{-7}$ and $8.0 \times 10^{-6}$ are arrived at?" Similarly, discuss the basis for missiles that are generated by rotating machinery and pressurized canisters.

## Response

Internal flood initiating event frequencies were developed from two sources. One source was an event-by-event review of Licensee Event Reports data published in Nuclear Power Experience and classified in terms of location and size. The results of this analysis were presented in Reference 8-7, and the results that were used in the South Texas Project (STP) Probabilistic Safety Assessment (PSA) are listed in Table 8.5-3. This database was updated and reanalyzed in PLG-0624 (Reference 1) to support a shutdown events PSA for Seabrook. Some of the key results of this updated assessment were plotted in Figure 5-1 of that report and are reproduced here. These updated results support flood frequencies that are generally smaller than those used in the STP PSA. Assumptions that were made to support the flood initiating event frequencies for specific locations are documented in the scenario tables in Table D-6 of Appendix D. For locations in which the flood sources were confined to up to a few pipe sections, internal floods were quantified with a frequency of 8E-6 per pipe section per year for nonsafety-grade pipe and 8E-7 per pipe section per year for safety-grade pipe.

The pipe failure rates listed above were derived from WASH-1400 data, but the usage is somewhat different. The value used for nonsafety-grade pipe was taken as the WASH-1400 value for pipe >3 inches in diameter with adjustments made to compute the mean value. WASH-1400 presents a lognormal distribution for this failure rate with the 95th and 5th percentiles of 3E-9 and 3E-12 per hour, respectively. From this, we computed the median and mean values using:

$$Median = \sqrt{\frac{3E\text{-}9}{3E\text{-}12} \times (3E\text{-}12)} = 9.5E\text{-}11 \textit{ per section hour}$$

$$Mean = median \times \exp\left(.1848 \ln^2 \sqrt{\frac{3E\text{-}9}{3E\text{-}12}}\right)$$

Mean = 8.6E-10 per section-hour x 8,766 hours per year
= 7.5E-6 per section-year ≐ 8E-6 per section-year

For safety-grade pipe, the failure rate was assumed in the STP PSA to be a factor of 10 lower than for nonsafety-grade pipe; thus, the 8E-7 per section-year value. The STP PSA used the same definition of pipe section as was used in WASH-1400 as follows:

> Section defined as an average length between major discontinuities such as valves, pumps, etc. (approximately 10 to 100 feet). Each section can include several welds, elbows, and flanges.

If a room contained less than one section of pipe but did not include any major discontinuities, the failure rate was not reduced to account for this.

The above treatment of pipe failure rates is acknowledged to depart from the usage in WASH-1400 in several ways. First, WASH-1400 combined both safety and nonsafety-grade pipe into a single failure rate and argued that the large range factor that was assigned accounts for both periodically tested and nontested pipe segments. Second, WASH-1400 assigned different pipe failure rates for small and large pipe with the value a factor of 10 times higher for pipes <3 inches in diameter. The STP PSA did not discriminate for pipe size. The approach that was followed in the STP study for pipe sections is viewed as reasonable for a screening level examination. We note that there is no evidence in nuclear power experience for significant flooding caused by pipe breaks with the exception of feedwater and steam line breaks.

As discussed on page 8.5-3 of the STP PSA final report, the spatial interactions analysis considered two sources of missile generation, depending on the type of equipment found in the specific location. For rotating equipment such as turbines, motors, diesel engines, compressors, etc., the likelihood of missile generation was based on an assumed missile generation rate of 1.0E-9 event per component operating hour. This value is one-tenth of the missile generation rate for large turbine generators, which was quantified in the Seabrook PSA to be approximately 8.5E-5 event per turbine operating year (Reference 2, page 9.9-5). This annual missile generation rate corresponds to an hourly frequency of approximately 9.7E-9 event per component operating hour. The reduced value of 1.0E-9 event per hour is believed to be conservative for small rotating and reciprocating equipment. Large turbines contain sufficient stored energy to generate missiles that could penetrate the component casing if the control systems fail to prevent overspeed or if material failures occur. With the exceptions of the turbine-driven auxiliary feedwater pump and the diesel generators, it is doubtful that any of the safety-related components considered for missile generation in STP PSA Table 8.2-2 meet this requirement. The rotating mass for most of these components is relatively small.

It is also very unlikely that any motor-driven component will develop a significant overspeed condition.

The second type of equipment considered for missile generation was pressurized canisters, (i.e., gas bottles), when these were located in a given area. It was assumed that these canisters would be handled once per year and that human errors resulting in missile generation would occur at a frequency of 1E-6 per handling. Both of these values and all remaining hazards were assigned based on engineering judgement. Assumptions made in developing these judgements are documented for each scenario in Table D-6 of Appendix D.

In a recently completed study by PLG, Inc. (formerly Pickard, Lowe & Garrick, Inc.) for the chemical industry, an analysis was made of experience with accidental releases of chlorine gas from 1-ton cylinders at U.S. chemical and process plants (Reference 3). Based on data provided by the Chlorine Institute for the U.S. between 1979 and 1987, a total of four serious releases were reported during an estimated 3,730,000-cylinder changeover operations. It is not known whether any of these events generated a missile; however, all involved inadvertent release of the entire cylinder contents. An upper bound of the extent of underreporting in the database for this study was set at 17. This results in a point estimate and upper bound estimate for the frequency of serious release per changeover of about 1.1E-6 to 1.9E-5 per changeover. It is expected that the frequency of handling gas bottles in nuclear power plants is substantially less, and, of course, the conditions are much different. Nonetheless, this chlorine cylinder experience indicates that the engineering judgements made in the STP PSA regarding pressurized container-generated missiles are reasonable.

## References

1. Fleming, K. N., et al., "Internal Flood Frequencies During Shutdown and Operation for Nuclear Power Plants," Pickard, Lowe and Garrick, Inc., prepared for New Hampshire Yankee, PLG-0624, May 1988.

2. Pickard, Lowe and Garrick, Inc., "Seabrook Station Probabilistic Safety Assessment," prepared for Public Service Company of New Hampshire and Yankee Atomic Electric Company, PLG-0300, December 1983.

3. Kazarians, M., and R. Y. LeVine, "Frequency of Serious Release from a 1-Ton Chlorine Cylinder," Pickard, Lowe and Garrick, Inc., prepared for Hoechst Celanese Chemical Group, PLG-0613, February 1988.

ATTACHMENT 1
ST-HL-AE-3682
PAGE 4 OF 20

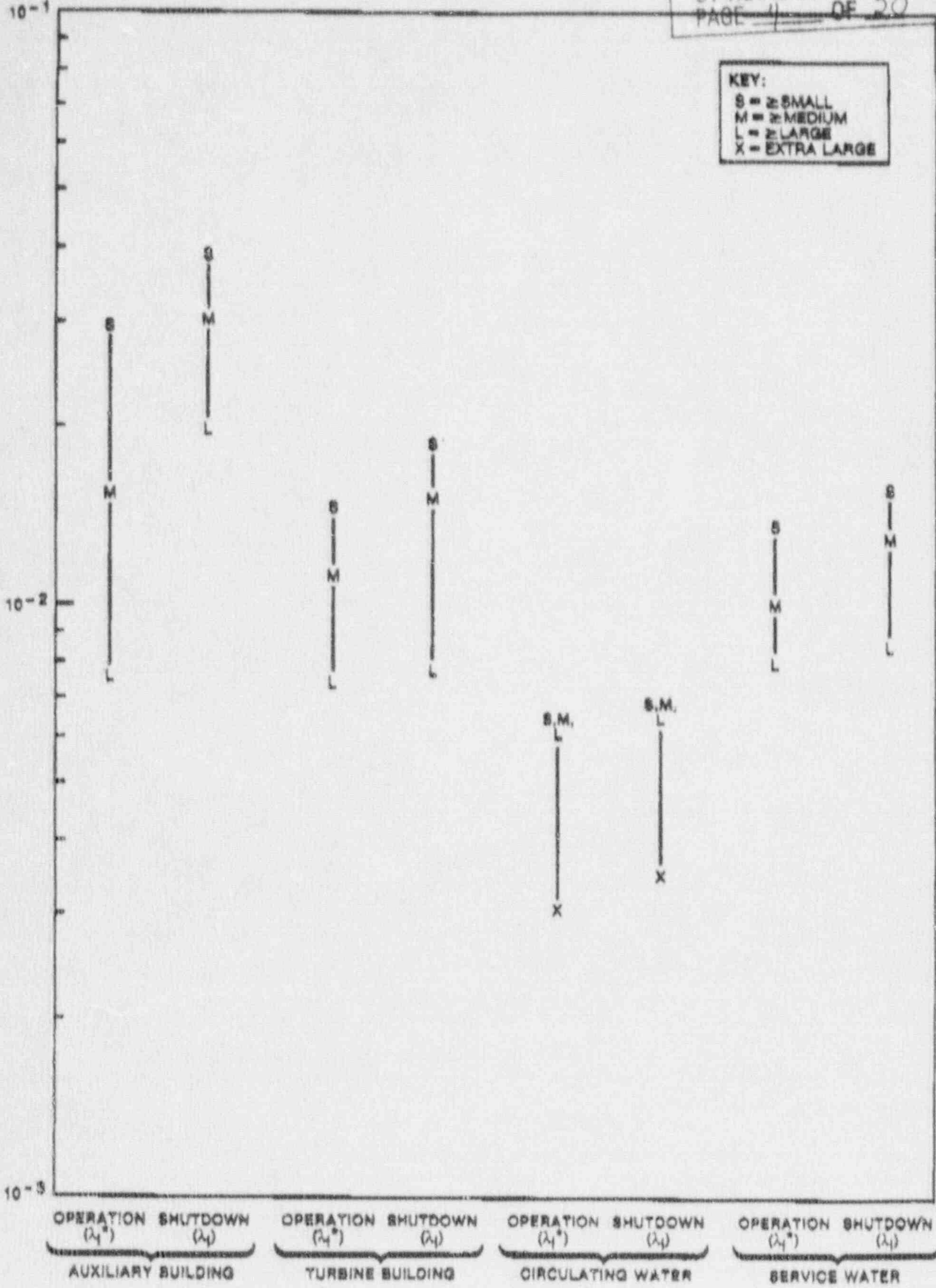



Figure 5-1. Mean Flood Frequencies (taken from Reference 1)

ATTACHMENT 1
ST-HL-AE-3682
PAGE 5 OF 20

## Question 2

Were any of the components that might impact seismic results found to be anchored by plug welds? If so, discuss the fragility calculations of these components with respect to the capacity of plug welds.

## Response

The seismic walkdown of the South Texas Project nuclear generating station (Reference 1) identified 16 mechanical and electrical components that appeared to exhibit the relatively lowest capacities for the representative classes of equipment. Of these, only the 4.16-kV switchgear were anchored by means of plug or puddle welds. Thus, the fragility for the switchgear was taken as the lower of (1) the functional fragility determined from results of the seismic qualification test or (2) failure of the puddle welds.

The switchgear are anchored by eight 17/32-inch puddle welds for each cabinet. The effective portion of each puddle weld was equivalent to a 1.67-inch fillet with a leg dimension of 0.10 inches. Since the footprint of the switchgear is large and the weight of the cabinet was strongly influenced by the weight of the breaker that is located low in the cabinet, the weld was critical only for the shear load.

The functional fragility of the cabinet was calculated by comparing the Test Response Spectrum (TRS) and the worst case location floor response spectrum. The fundamental frequency of the switchgear is 11 Hz in the front-to-back direction, which is near the peak of the response spectrum. This was used as the basis for the functional fragility.

The functional fragility was lower than the fragility based on the failure of the puddle welds. The generic functional failure mode was therefore used to describe the seismic fragility of the switchgear.

## References

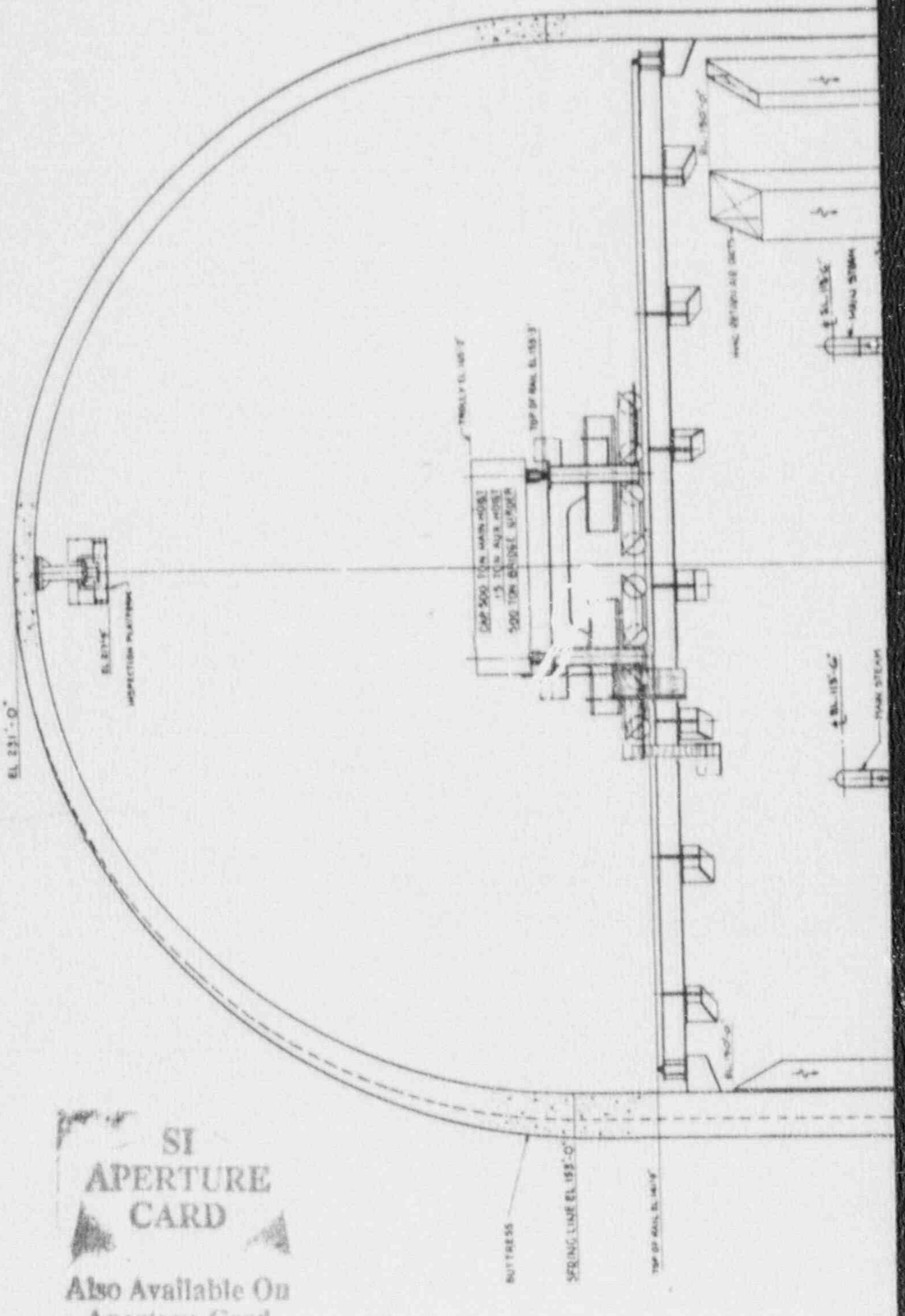

1. Letter from H. F. Perla to R. P. Murphy, "South Texas Project Seismic Walkdown Report," ST-RL-HL-0354, October 16, 1990.

Question 3

In the context of falling objects, Section 3.4.1.3.1, was the potential seismic interaction associated with the movable in-core flux mapping system considered? Portions of the in-core flux mapping system located above the seal table may not have been seismically analyzed. Failure of this equipment during a seismic event could cause multiple failures at the seal table resulting in equivalent small-break LOCA.

Response

The movable in-core flux mapping system at STP is horizontally located as shown on the attached drawings (Drawing Numbers 6C-18-9-N-5008 and 6C-18-9-N-5001). Simplified figures of these drawings are also provided (figures "RCB Section B-B" and "Reactor Containment Building EL (-)11'-3""). This arrangement is unlike the flux mapping system considered, for example, for the Zion plant, which is shown in the attached Figure 2.2 from NUREG/CR-5567. In the case of Zion, portions of the in-core flux mapping system are located above the seal table, and, in the context of falling objects resulting from a seismic event, could interact with the seal table. In the case of STP, this possibility is prevented by its design.






REV | HL&P | DATE

| CHIEF ENGR | | NO. | REVISION | DFT | CKR | GL | EGS | PEM | DATE | NO. | REVISION |
|---|---|---|---|---|---|---|---|---|---|---|---|
| | N/A | 3 | [illegible] | [illegible] | [illegible] | [illegible] | [illegible] | [illegible] | [illegible] | | |
| | N/A | A | GENERAL REVISION | [illegible] | [illegible] | [illegible] | [illegible] | [illegible] | 6.26.84 | | |
| | N/A | B | REVISED PLATE AT EL 90'-0" | [illegible] | DEH | [illegible] | [illegible] | [illegible] | [illegible] | | |
| | N/A | 2 | ADDED VE FAN [illegible] | [illegible] | [illegible] | [illegible] | [illegible] | [illegible] | [illegible] | A | ADDED UNIT-2 SERVICE DESCR[illegible] |
| | N/A | 1 | [illegible] | [illegible] | [illegible] | [illegible] | [illegible] | [illegible] | [illegible] | 1 | ISSUED FOR DOCUMENT TURN[illegible] |
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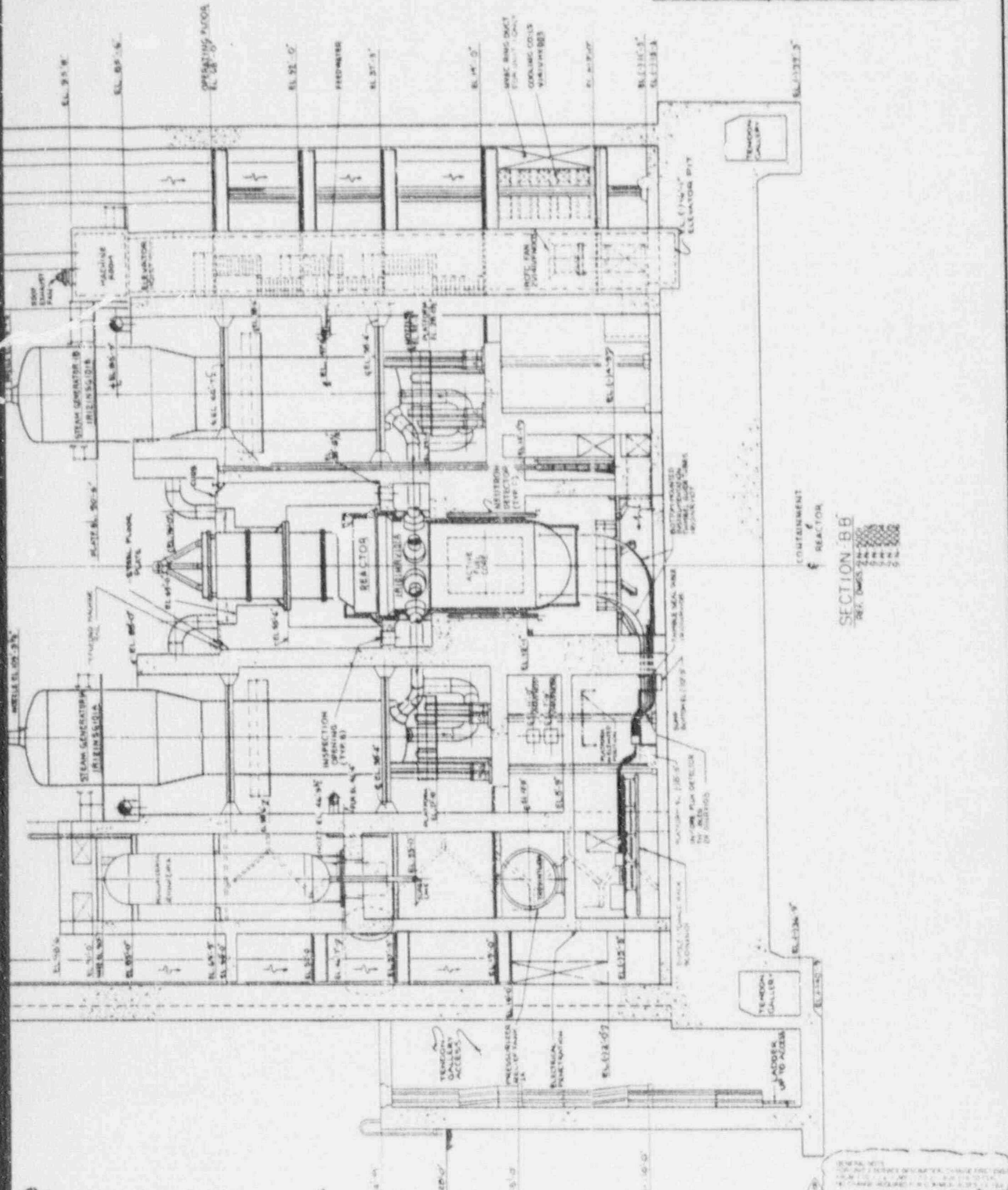
ATTACHMENT 1
ST-HL-AE-3682
PAGE 7 OF 20



ALL UNIT 1 DESIGNATORS SHOWN ON THIS DRAWING SHALL BE INTERPRETED AS BEING UNIT 2 DESIGNATORS WHEN ERECTING UNIT 2 COMPONENTS

9103260232-01

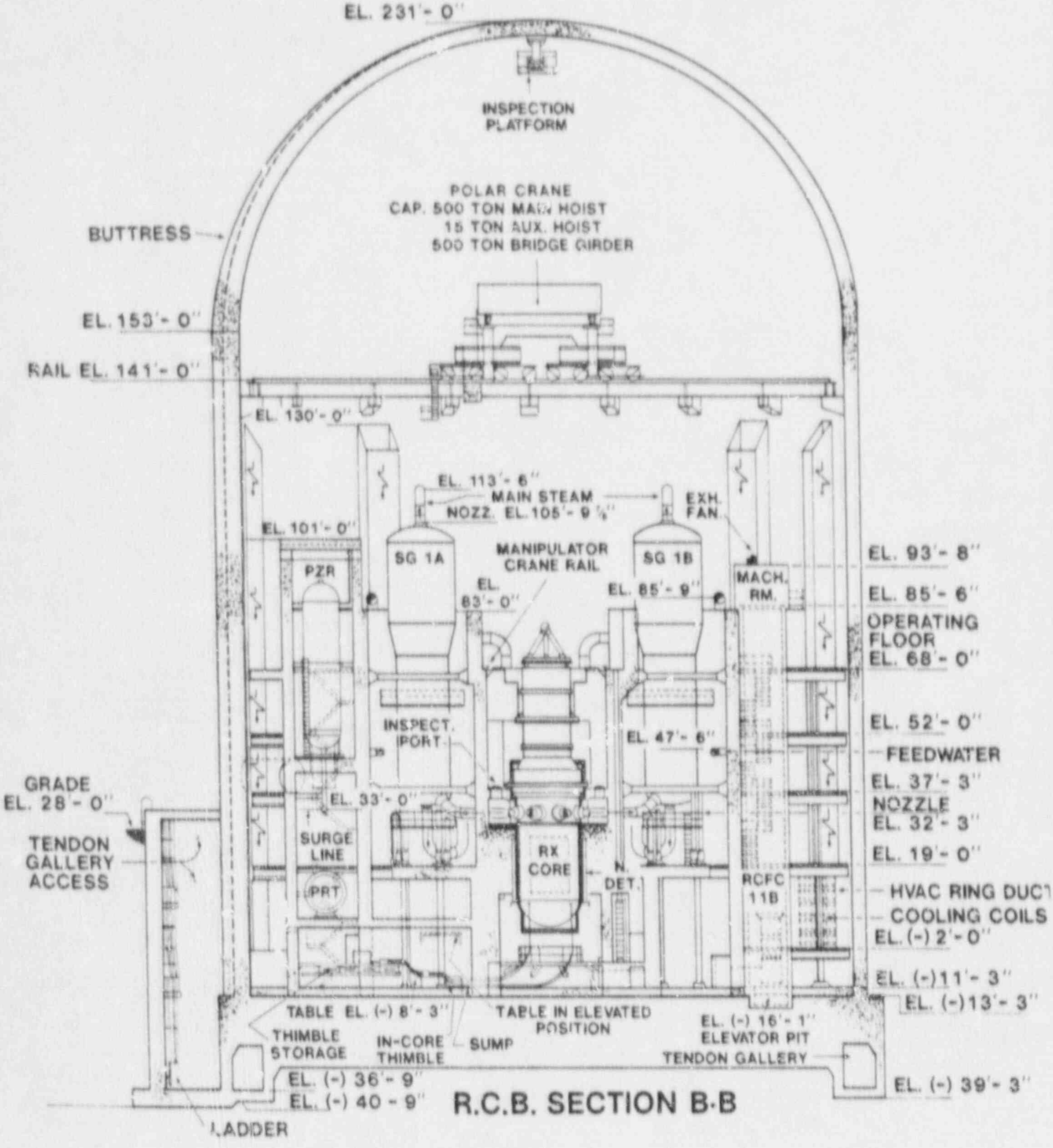
BECHTEL ENERGY CORPORATION
HOUSTON, TEXAS

HOUSTON LIGHTING & POWER CO.
SOUTH TEXAS PROJECT

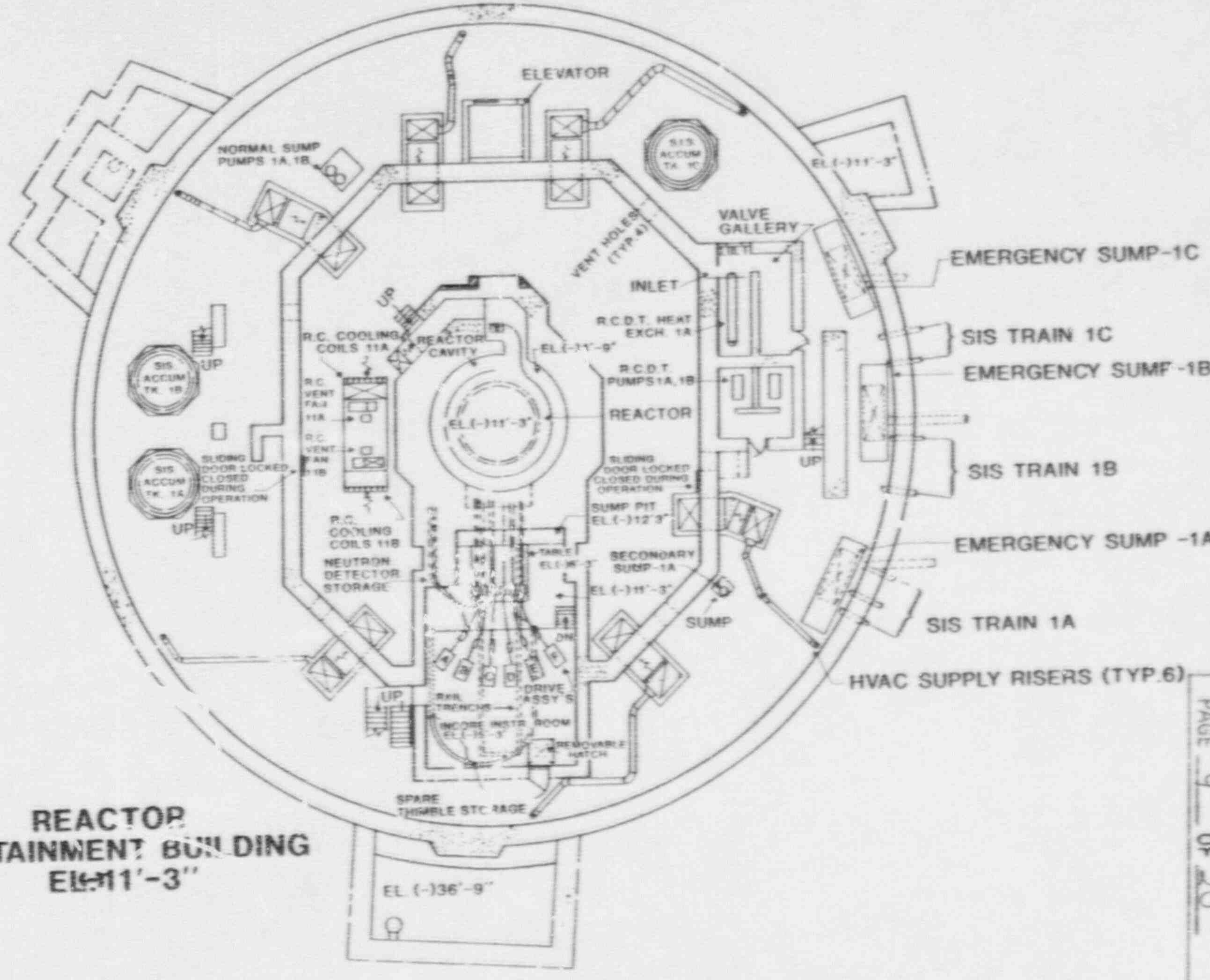
GENERAL ARRANGEMENT
REACTOR CONTAINMENT BUILDING
SECTION B-B AREA 6

| SCALE | JOB NO. | DRAWING NO. | REV. |
|---|---|---|---|
| 1/4"=1'-0" | 14926 | 6C-18-9-N-5008 | 8 |

ATTACHMENT 1
ST-HL-AE-3682
PAGE 8 OF 20




R.C.B. SECTION B-B




**REACTOR CONTAINMENT BUILDING EL.-11'-3"**

ATTACHMENT /
ST-HL-AE-3682
PAGE 9 OF 20

ATTACHMENT 1
ST-HL-AE-3682
PAGE 10 OF 20

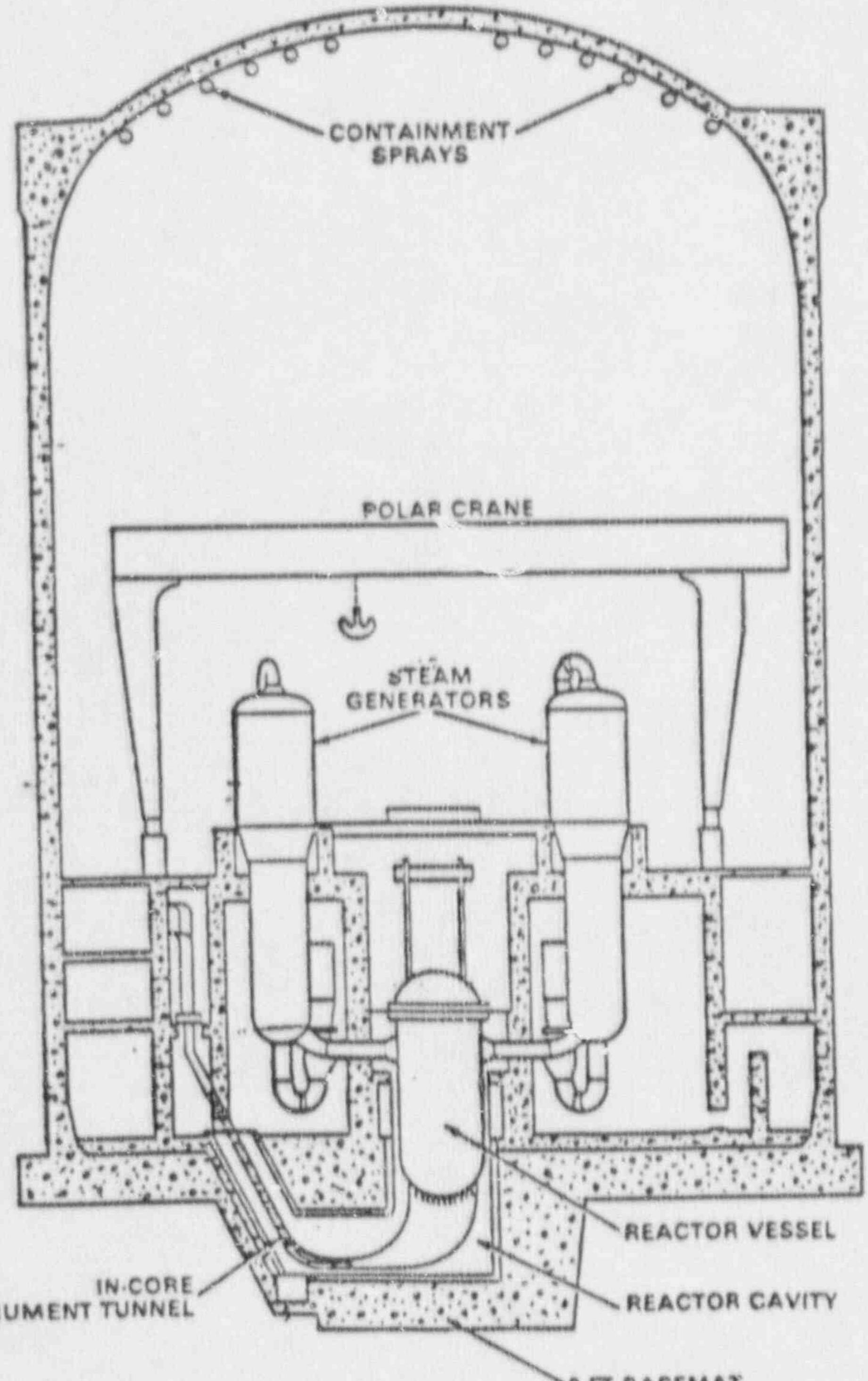



Figure 2.2 Prestressed concrete containment (Zion).

Question 4

As discussed in the draft IPEEE generic letter (88-20, Supplement 4) and guidance document (NUREG-1407), for many external events (other than seismic and fire), if it can be ascertained that a plant has been designed to meet current criteria (1975 Standard Review Plan), additional analysis may not be needed to evaluate severe accidents initiated by those external events. In light of this, it would be very instructive to compare FSAR design criteria (flood level, wind speed, RGs used, etc.) with initiating event frequencies and corresponding criteria used in the PRA analysis in a tabular form. (It is realized that such information is discussed in various sections throughout the PRA, however an organized comparison could be very useful).

It would also be useful to know whether any changes have occurred at or near the site to alter the design information described in the FSAR, for example, construction of a new facility or stacks, or changes related to transportation, chemical or other similar hazards.

Response

Generic Letter No. 88-20, Supplement 4, Draft For Comment, identifies that in addition to seismic events and internal fires, the other events to be considered in any external events analysis are high winds and tornadoes, external floods, and transportation and nearby facility accidents. This same draft letter and the information made available by the NRC at its Workshop in Pittsburgh on September 11-13, 1990, indicated that a "screening type approach as shown in Figure 1 [Attached] is considered adequate" on the latter three hazards. The licensee should first determine if the 1975 Standard Review Plan (SRP) criteria are met. If the plant does not meet the 1975 SRP criteria, the licensee should examine it further using the recommended optional steps. In addition, the draft letter indicates that the licensee may "bypass one or more of the optional steps so long as vulnerabilities are either identified or proved to be insignificant."

Not having had the benefit of this direction at the time (i.e., step 3 relative to the 1975 SRP criteria to which STPEGS generally conforms), the external events PRA was completed on STPEGS (step (6) on the attached figure). Screening analyses on the three hazards identified above are addressed in Sections 13 ("External Flood Analysis") and 14 ("Other External Events") in the STPEGS PSA. Section 14 includes screening analyses on Aircraft Hazard, Turbine Missile, Tornado Wind and Missile, Hazardous Chemical, and [illegible] and ECW Intake Screen Blockage. These analyses show that the impact of those events are insignificant.

However, it may be seen from the attached Table 1 (IPEEE Survey For STPEGS -Design Basis/Regulatory Guides/Standard Review Plan) and the referenced sections in the STPEGS UFSAR that STPEGS conforms [illegible] eneral with SRP guidance as discussed in the UFSAR. Changes [illegible] have occurred are noted in Table 1.

ATTACHMENT
ST-HL-AE-3682
PAGE 12 OF 20

Figure 1

RECOMMENDED IPEEE APPROACH
FOR WINDS, FLOODS, AND OTHERS

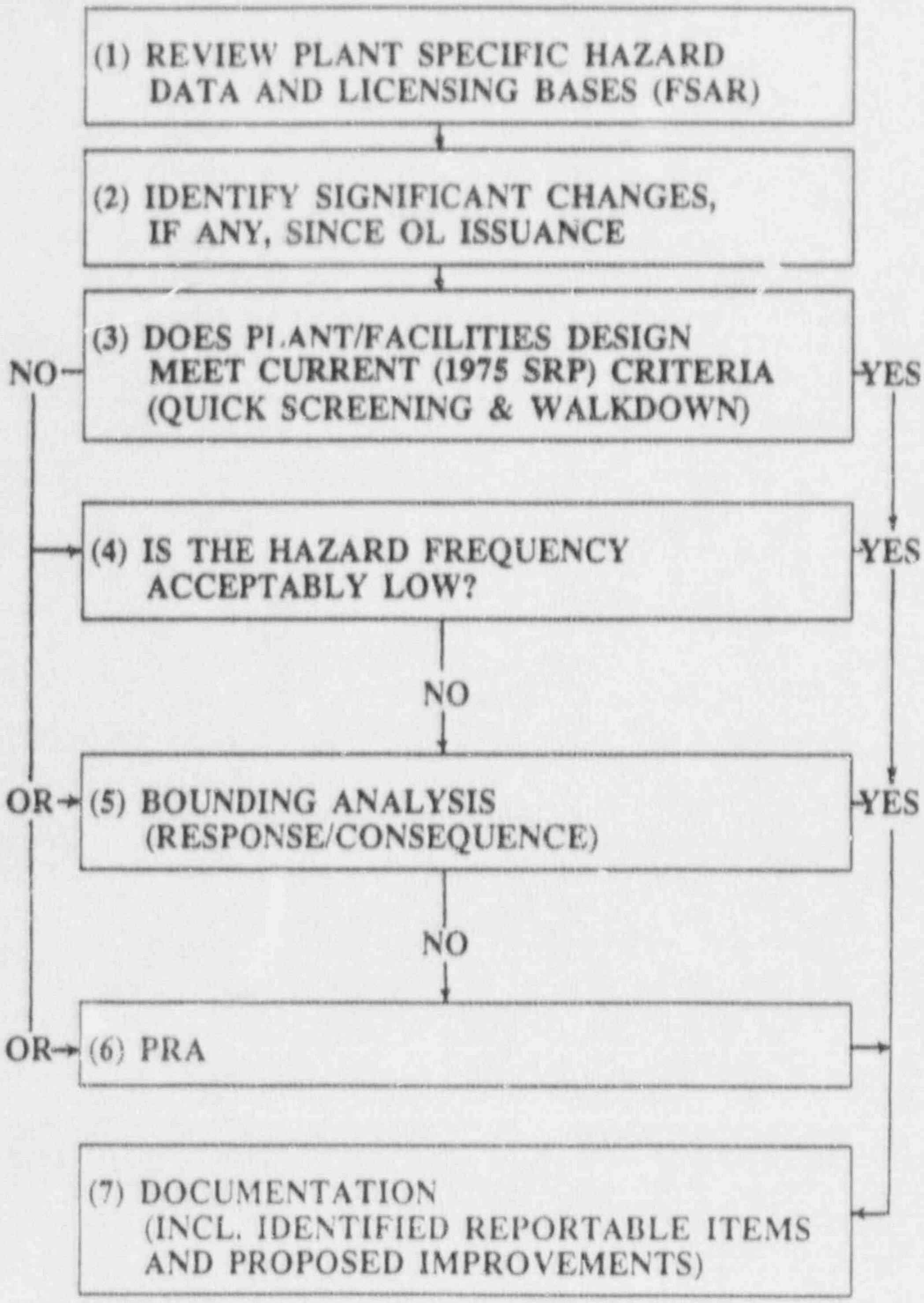


TABLE 1
IPEEE SURVEY FOR STPEGS
DESIGN BASIS/REGULATORY GUIDES/STANDARD REVIEW PLAN

HIGH WIND DESIGN CRITERIA

Straight Wind Speed: 125 mph 30 ft above ground (UFSAR 3.3.1).

Tornado Wind Speed: Tangential 290 mph; translational 70 mph; total 360 mph (UFSAR 3.3.2).

Tornado Pressure Drop: 3 psi (UFSAR 3.3.2).

Radius to Maximum Rotation Wind Velocity: 150 ft (UFSAR 3.3.2).

Tornado Missiles: Table 3.5.9 from UFSAR (attached) (UFSAR 3.5.1.4).

Bases: Straight wind speed - 100 year recurrence interval complies with RG 1.70. Tornado design complies with RG 1.76.

STP PSA: Tornado occurrence frequency at site: 1.67E-5/yr.
Design basis tornado frequency: 8.33E-9.

EXTERNAL FLOODING DESIGN CRITERIA

Project Flood Level: 50.8 ft above MSL (MCR embankment breach) (UFSAR 3.4).

Design Basis Flood Level: Same as above.

Bases: Conform to RG 1.102 and RG 1.59 (Rev 0).

Note: Changes are discussed in the response to Question 5 below.

STP PSA: Initiating event frequency: 6.2E-6.
Flood basis same as UFSAR.

MAN-MADE HAZARDS

ATTACHMENT 7
ST-HL-AE-3682
PAGE 14 OF 20

Transportation Explosion or Impact:

Rail: No impact.

Truck: 8000 gallon gasoline or 5 tons TNT. No adverse effects.

Marine: Less than truck.

Bases: Satisfies RG 1.91 Rev 0 (UFSAR 2.2.3.1).

STP PSA: No impact. Qualitative discussion only in Section 8. Previously discussed in Preliminary Scoping Study submitted to NRC.

Aircraft Crash: No impact.

Bases: See UFSAR 3.5.1.6. Total frequency in UFSAR from both general aviation and US air carriers is 2.85E-6.

STP PSA: IE Frequency: 6.95E-7.

Toxic Gas:

Chlorine: No impact.

Other: Detection, alarm and isolation for vinyl acetate and anhydrous ammonia. Detection and alarm for HCl, acetic acid and naptha.

Bases: Methods and assumptions are in agreement with the guidance given in RG 1.78 and the methodology presented in NUREG-0570. (UFSAR 2.2.3 and 6.4.4.2).

STP PSA: No Impact. Qualitative discussion only.

REG GUIDES AND SRP

High Winds, Tornadoes and Missile

| | |
|---|---|
| RG 1.76 Rev 0 | Conform to guidance. |
| RG 1.117 Rev 0 | N/A but conform to guidance. |
| SRP 3.3.1 | |
| 3.3.2 | |
| 3.5.1.4 | |
| 3.5.2 | |
| 3.5.3 | SRP position is included in UFSAR chapters. Generally in conformance with guidance in SRP. |

External Flooding

| | |
|---|---|
| RG 1.102 Rev 1 | Conform to guidance. |
| RG 1.59 Rev 0 | Conform to guidance. |
| SRP 2.3.1 | |
| 2.3.2 | See High Winds above. |

Man-Made Hazard

| | |
|---|---|
| RG 1.78 Rev 0 | Partial exception (no seismically qualified toxic gas instruments available in industry). |
| RG 1.91 Rev 0 | N/A but conform. |
| RG 1.95 | N/A. See UFSAR 6.4.4.2. |
| SRP 2.2.3 | See High Winds above. |

ATTACHMENT I
ST-HL-AE-3682
PAGE 16 OF 20

TABLE 3.5-9

CHARACTERISTICS OF TORNADO-GENERATED MISSILES

| Missile | Length (ft) | Weight (lb) | Velocity (ft/sec)* |
|---|---|---|---|
| 4" x 12" wood plank | 12 | 200 | 420 |
| 3"-diameter schedule 40 steel pipe | 10 | 78 | 210 |
| 1"-diameter steel rod (reinforcing bar) | 3 | 8 | 310 |
| 6"-diameter schedule 40 steel pipe | 15 | 285 | 210 |
| 12"-diameter schedule 40 steel pipe | 15 | 743 | 210 |
| 13.5"-diameter wooden utility pole | 35 | 1,490 | 210 |
| Automobile (4' x 5' frontal area) | 15 | 4,000 | 100 |

The first five missiles are considered at all altitudes and the last two missiles at altitudes up to 30 ft above grade levels (except the Cooling Reservoir embankment) within one-half mile of the safety-related structures. There are no utility poles atop the embankment within one-half mile of the safety-related structures. There is an access road on top of the embankment, but there will be limited traffic on the road and then only on rare occasions, consisting only of authorized vehicles being used during inspection or maintenance activities. No part of the embankment is closer to safety-related structures, systems, or components than 650 ft.

* Assuming a Region I tornado, as defined in Regulatory Guide 1.76 (April 1974).

ATTACHMENT 1
ST-HL-AE-3652
PAGE 17 OF 20

Question 5

The flooding event associated with failure of the upstream dams, considers two dams, the Mansfield and Buchanan dams. The PRA analysis indicates that the water level at the site due to the failure of these two dams would be at El. 32.0 ft. The following have not been addressed in this analysis. Please explain:

a. Failure/effect of the Columbus Bend or other proposed/built dams.

b. Effect of upstream dam breaks on Main Cooling Reservoir (MCR) and Essential Cooling Pond (ECP).

Response

a. At the time the analysis for the STP PSA was prepared, the Columbus Bend Dam had been proposed. A flooding analysis including this proposed dam was (and still is) addressed in the STP UFSAR Section 2.4.4.

The status of this dam remains unchanged at this time. For that reason, the failure of the proposed Columbus Bend dam has not been included in the STP PSA. Including the effect of failure of this dam along with the Mansfield and Buchanan Dams, all on the Colorado River as discussed in the UFSAR, leads to the conclusion that the resulting flood level is below that for the design basis flood resulting from breach of the main cooling reservoir embankment.

A fourth dam (the Stacy Dam, also known as the Freeze Dam) which is also located on the Colorado River upstream of Buchanan Dam has recently been constructed. The filling of the reservoir, which is currently underway, is scheduled for completion in 1992. An amendment to the UFSAR is currently being prepared to include the effects of failure of this dam. The draft amendment indicates that the flood level resulting from a cascaded failure of the Mansfield Dam and all major dams upstream from it (including Buchanan and Stacy Dams) is also below that for the design basis flood (breach of the MCR embankment). This assumed flood level also includes an antecedent flow between Mansfield and Bay City (in which area would be located the proposed Columbus Bend Dam) equal to the Spillway Design Flood (SDF) from the proposed Columbus Bend Dam in coincidence with the peak Standard Project Flood (SPF). The failure of the Stacy Dam has also not been included in the STP PSA due to the timing of its analysis, and its completion and inclusion in the UFSAR.

The effect of failure of upstream dams in addition to the Mansfield and Buchanan dams will be included in the STP PSA as it is updated in the future.

b. The effect of the upstream dam breaks on the Essential Cooling Pond (ECP) is discussed in Section 13.9 of the PSA. The ECP is designed to withstand the hydraulic forces resulting from failure of upstream dams on the Colorado River or a breach of the MCR embankment (UFSAR, Section 9.2.5).

Cooling is provided to essential safety systems from the water in the ECP in the case of an accident. The crest of the circumferential embankment around the ECP is at El. 34 ft. and of the dividing dike is at El. 38 ft. The embankment and dike are separated from the pond by a 30-ft-wide berm at El. 26.0 ft. The natural ground surface is at approximately El. 26 ft. The minimum water level in the ECP is at El. 25.5 ft. The ECP embankment is designed to be over-topped during severe flood conditions without impacting the primary function of the pond. The higher elevation of the central dike prevents warm discharge water from flowing directly into the intake facility. Thus no impact on the ECP by external flooding has been included in the analysis since this impact would be expected to be negligibly small.

The water in the Main Cooling Water Reservoir (MCR) does not provide a safety-related function. The South Embankment of the ECP is designed to withstand the dynamic and hydrostatic forces caused by the flood wave propagating from an MCR embankment breach (UFSAR Section 2.4.4). Thus, loss of the MCR as the result of an MCR embankment breach does not impact the ability to provide safety-grade cooling for plant essential safety features.

The downstream slope of the MCR embankment has a slope angle of 3 on 1. The inside embankment slope is 2.5 on 1. The inside slope has been the subject of numerous slope stability analyses using a variety of submerged conditions much more severe than forecast conditions on the downstream slope due to external flooding. Thus no impact on the MCR by external flooding has been included in the analysis.

Question 6

Several recovery actions are identified in Tables 15.5-12 through 15.5-17 with respect to seismic-initiated sequences. Discuss the basis for error rates associated with these actions under seismic environment. Also, discuss impact of not including these recoveries on seismic-induced core damage frequencies. Recovery actions HECH03 and HECH04 are associated with starting the technical support diesel generator (TSDG) and the positive displacement pump; however, discussion in the seismic section implies guaranteed failure of TSDG with seismic-induced loss of offsite power (LOOP), and Table 3.4.4-4 indicates very low capacity of the TSDG. In light of this, clarify how these recovery actions are used in the analysis.

Response

The recovery actions identified in Tables 15.5-12 through 15.5-17 were developed specifically for scenarios initiated by seismic events. The analysis of these recovery actions was performed in the same way as the dynamic human actions described in Section 15.4, with the exception that detailed operator surveys for performance-shaping factor (PSF) evaluation were not done for the recovery factors in Section 15.5. Evaluations of the PSF ratings, weights, and calibration tasks were performed solely by the human actions analysis team, using the same data sources and computer tools as were used in Section 15.4.

The recovery actions identified in Tables 15.5-12 through 15.5-17 were used in the analysis of several seismic scenarios. If these six recovery actions are not included in the quantification of core damage, the seismic-induced core damage frequency increases from 2.04E-7 per year to 3.05E-7 per year. This calculation assumes that all seismic initiating events, as defined in Section 11, that used any of these recovery actions result directly in core damage if these recovery actions are removed.

The availability of the positive displacement pump is questioned in Top Event PD of the frontline event tree. Top Event PD is questioned only after failure of the centrifugal charging pumps to provide RCP seal cooling. The positive displacement pump is normally powered from 480V MCC 1G8 but, given loss of offsite power, must be powered from the Technical Support Center (TSC) Diesel Generator (DG). The TSC DG is assumed to be unavailable for every seismic-induced loss of offsite power event. Therefore, no credit was taken for recovery actions that require operation of the positive displacement pump during seismic LOOP event sequences. For non-LOOP seismic sequences, two positive displacement pump recovery actions were analyzed involving operator actions HECH03, given a loss of AC power due to switchgear ventilation failure, and HECH04, given seismic-induced component cooling water failure. After a loss of onsite AC power caused by seismic-induced ventilation failures, the split fraction for Top Event PD (PDR) includes hardware failures of the positive displacement pump and the TSC DG in addition to operator action HECH03. If split fraction PDR is set to 1.0 (guaranteed failure), seismic-induced core damage increases from 2.07E-7 to 2.48E-7 per year.

ATTACHMENT 2

# SOUTH TEXAS V-SEQUENCE ANALYSIS

## INTRODUCTION

Presented here is a model that bounds the upper limit of the frequency of containment bypass sequences at the South Texas Project (STP) Unit 1 or 2. The various locations of high pressure-low pressure boundaries between the reactor coolant system (RCS) and the interfacing systems are discussed in detail in Reference 1. Of these, the lines that are most likely to be subject to the bypass sequences are discussed in this analysis; these are the three low head safety injection (LHSI) lines. Figure 1 is a simplified diagram of the configuration and major components in LHSI train A. The other two trains are essentially identical. The section of each line closest to the RCS is rated to withstand full RCS design pressure. This section contains two check valves (SI0038 and RH0032) and a motor-operated valve (MOV)(RH0031). The MOV is normally open, and its power supply is locked out at the motor control center (MCC). Beyond this MOV, the system is rated for a lower pressure. In this section are the residual heat removal (RHR) heat exchangers and their flow control valves.

Discharge lines from both the LHSI pump and the RHR pump feed into the inlet side of each heat exchanger. The RHR pump is separated from the heat exchanger by a check valve (RH0065), but the entire RHR system is situated inside the containment. The LHSI pump is separated from the heat exchanger by a similar check valve inside the containment (SI0030), but the rest of the LHSI system is outside the containment.

## ASSUMPTIONS AND ADDITIONAL INFORMATION

1. The leakage/rupture failure rates for the first two check valves (e.g., valves SI0038A and RH0032A) are assumed to be the same. These valves are both rated for pressures that exceed normal reactor coolant system operating pressure. The leakage/rupture failure rate for the other two check valves is different; e.g., valves RH0065A and SI0030A. Each of these valves is rated for a pressure of approximately 600 psig.

2. The space between the first two check valves is not continuously monitored. Minor leakage past the first valve (e.g., valve SI0038A) may pressurize this space and cause undetected high differential pressure across the second valve; e.g., valve RH0032A. It is conservatively assumed that both of these valves are exposed to full system pressure for the entire period between refueling outages. It is also assumed that if the space between these valves is pressurized and one of the valves fails catastrophically, the other valve will be exposed to a sudden pressure pulse.

3. The RHR relief valve (e.g., valve PSV3934) is rated to open at approximately 600 psig, and it has a rated flow capacity of approximately 20-gpm water at that pressure.

4. The RHR relief valve will open if minor leakage occurs through the first two valves (e.g., valves SI0038A and RH0032A), and this section of line is pressurized above approximately 600 psig. The relief valve discharges to the pressurizer relief tank. This leakage will be quickly detected, and the plant will be shut down and depressurized. Therefore, the RHR and LHSI isolation check valves (e.g., valves RH0065A and SI0030A) are not pressurized until both of the first two valves fail.

5. The first two check valves (e.g., valves SI0038A and RH0032A) are confirmed closed by functional tests performed at the end of every refueling outage. These tests are also performed during all other unscheduled cold shutdown outages before the plant enters Mode 2. This analysis accounts only for the tests performed every 18 months during the regular refueling outage.

6. The minimum allowable pressure in the accumulators is approximately 586 psig. The RHR relief valve setpoint is approximately 600 psig. Therefore, if the first check valve is intact (e.g., valve SI0038A) and only the second check valve develops a leak (e.g., valve RH0032A), it is assumed that the RHR relief valve will not open. This leak will remain undetected. However, if both the second and the third check valves develop leaks (e.g., valves RH0032A and SI0030A), the accumulator will drain into the emergency core cooling system (ECCS) pump room sump through the LHSI pump and its suction line relief valve; e.g., valve PSV3935. The resulting loss of accumulator level will alert the operators to this condition, and the plant will be shut down. Therefore, accumulator level provides an effective method for determining that at least one of these valves is intact during normal plant operation; e.g., valve RH0032A or valve SI0030A.

7. No functional tests are performed to verify that either the RHR isolation check valve or the LHSI isolation check valve is closed while the plant is operating at power; e.g., valve RH0065A or valve SI0030A. Either of these valves may be stuck in the open position if it failed to reclose after previous RHR or LHSI system operation.

8. As long as the leakage of the RCS past the first two check valves (e.g., valves SI0038A and RH0032A) is within the capacity of the relief valve, the two low pressure pump

isolation check valves (e.g., RH0065A and SI0030A) will not be exposed to pressures above 600 psig. If the leaks are beyond the capacity of the relief valve, pressure will start rising in the RHR heat exchanger piping unless another relief path is available. It is assumed for this analysis that the heat exchanger and the piping survive the increased pressure. This assumption may be quite conservative, depending on the relative strengths of the heat exchanger, its connected piping, and the low pressure isolation check valve disks; e.g., valves RH0065A and SI0030A. If the heat exchanger shell or its piping fails before either of the check valves, the RCS leakage will be confined to the containment, and the resulting scenario is a small or a medium loss of coolant accident (LOCA). However, it is assumed for this analysis that the two isolation check valves are the weak points in the system due to failure of the check valve disk. As soon as one of the two check valves fails, the pressure will no longer challenge the heat exchanger, the piping, or the other check valve. Since the two check valves are identical in design, either of them is equally likely to fail first.

9. As long as the leak past the first two valves (e.g., valves SI0038A and RH0032A) is within the capacity of the charging pump, the leak will be treated as a very small LOCA whether it is inside or outside the containment. The letdown line will be isolated, normal charging pump operation will maintain pressurizer level, and the charging pump suctions will automatically transfer to the refueling water storage tank (RWST) when the volume control tank (VCT) is drained. Plant shutdown can be attained before the RWST water is exhausted. There is a range of leak rates beyond the capacity of the charging pump for which the plant may also be shut down and depressurized before the RWST is drained. However, it is conservatively assumed for this analysis that any leak greater than the makeup capacity of the charging pump (120 gpm) is a potential bypass sequence and that the RWST will be drained before the plant can be shut down and depressurized.

10. The onset of any significant leakage past the first two check valves (e.g., valves SI0038A and RH0032A) will be detected from increasing temperature and pressure in the pressurizer relief tank. The specific line that is leaking can be determined from temperature and pressure alarms from the pressurized line: TA857, TA874, and PA861. To terminate the leak, the operator would have to close the MOV in the high pressure line; e.g., valve RH0031A. This valve is normally in the open position with its power locked out at the MCC. Therefore, to close the valve, operators must first locally reenergize its power supply at the MCC. The valve can then be closed from either its MCC controls or its switch in the main control room.

## MODEL

**Failure of Check Valves SI0038 and RH0032**

In general, the frequency of failure for two valves, $V_1$ and $V_2$, in series ($V_1$ is assumed to be nearest to the RCS) can be expressed as

$$\lambda_s = \lambda(V_1)^* P(V_2|V_1) + \lambda(V_2)^* P(V_1|V_2) \tag{1}$$

where

$\lambda_s$ = the frequency of failure of both series valves.

$\lambda(V_1)$ = the frequency of random, independent failure of $V_1$.

$P(V_2|V_1)$ = the conditional likelihood that $V_2$ is failed, given that $V_1$ fails.

$\lambda(V_2)$ = the frequency of random, independent failure of $V_2$ (events per hours).

$P(V_1|V_2)$ = the conditional probability that $V_1$ is failed, given that $V_2$ fails.

$P(V_2|V_1)$ and $P(V_1|V_2)$ are composed of both random independent and demand-type failures of the second valve.

In some cases, the random independent failure frequencies and conditional probabilities for the two valves will be approximately equal, but, in other cases, they will not. For example, if $V_1$ leaks slightly but $V_2$ does not, $V_2$ would be exposed to the differential pressure loading to which $V_1$ is normally exposed. In this situation, $V_1$ would have RCS pressure on both sides of the disc and would be expected to have a lower failure rate than $V_2$, which is exposed to a greater differential pressure. Thus, Equation (1) could be written as

$$\begin{aligned}\lambda_s = {} & \lambda(V_1)^* P(V_2|V_1)^*(1-P_I) + \lambda'(V_1)^* P'(V_2|V_1)^* P_I \\ & + \lambda(V_2)^* P(V_1|V_2)^*(1-P_I) + \lambda'(V_2)^* P'(V_1|V_2)^* P_I\end{aligned} \tag{2}$$

where

$P_I$ = the probability that the space between valves is pressurized to RCS pressure.

$\lambda'(V_1)$ = the frequency of a random, independent failure of $V_1$, given that the space between valves is pressurized (events per hour).

$P'(V_2|V_1)$ = the conditional probability that $V_2$ fails, given that $V_1$ has failed, and the space between valves is pressurized.

$\lambda'(V_2)$ = the frequency of a random, independent failure of $V_2$, given that the space between valves is pressurized.

$P'(V_1|V_2)$ = the conditional probability that $V_1$ fails, given that $V_2$ has failed and the space between valves is pressurized.

On the basis of the loadings across the valve discs, the following assumptions appear to be reasonable for the lines that contain the check valves:

1. $\lambda'(V_2) \simeq \lambda(V_1)$. The two valves are assumed to be physically identical. Therefore, if valve $V_2$ is exposed to full reactor coolant system pressure, its random, independent failure rate should be the same as that normally observed for valve $V_1$.

2. $\lambda'(V_1)$ is small compared to $\lambda(V_1)$. If the space between the valves is pressurized, the differential pressure across valve $V_1$ will be very small. The loading across the valve disc will be much lower than if the valve were exposed to full reactor coolant system differential pressure. If the loading across the valve disc is reduced, its random, independent failure rate should be significantly lower than the failure rate normally observed at full reactor coolant system differential pressure.

3. $\lambda(V_2)$ is small compared to $\lambda'(V_2)$ . If the space between the valves is not pressurized, there will be essentially no differential pressure across valve $V_2$. Therefore, there will be essentially no loading across the valve disc. If the loading across the valve disc is reduced, its random, independent failure rate should be significantly lower than the failure rate normally observed at full reactor coolant system differential pressure.

4. $P'(V_1 | V_2) \simeq P(V_2 | V_1)$. If the space between the valves is pressurized and valve $V_2$ fails, pressure will be rapidly removed from the downstream side of valve $V_1$. This depressurization transient will expose the disc of valve $V_1$ to a sudden differential pressure pulse. It is assumed that this pressure pulse is comparable to the sudden differential pressure that would be experienced by valve $V_2$ if the space between the valves were not pressurized and valve $V_1$ failed. If these differential pressure pulses are comparable, the conditional probabilities that each exposed valve fails should be approximately equal.

Substituting for $\lambda'(V_2)$ and $P'(V_1 | V_2)$,

$$\begin{aligned}\lambda_s \simeq\ & \lambda(V_1)^* P(V_2 | V_1)^*(1-P_l) + \lambda'(V_1)^* P'(V_2 | V_1)^* P_l \\ & + \lambda(V_2)^* P(V_1 | V_2)^*(1-P_l) + \lambda(V_1)^* P(V_2 | V_1)^* P_l\end{aligned} \tag{3}$$

or

$$\begin{aligned}\lambda_s \simeq\ & \lambda(V_1)^* P(V_2 | V_1) + \lambda'(V_1)^* P'(V_2 | V_1)^* P_l \\ & + \lambda(V_2)^* P(V_1 | V_2)^*(1-P_l)\end{aligned} \tag{4}$$

Application of assumptions 3 and 1 from above indicates that the third term in Equation (4) is small compared to the first; therefore,

$$\lambda_s \simeq \lambda(V_1)^* P(V_2 | V_1) + \lambda'(V_1)^* P'(V_2 | V_1)^* P_l \tag{5}$$

Assumption 2 from above indicates that $\lambda'(V_1)$ should be much smaller than $\lambda(V_1)$. However, a conservative upper bound for the overall failure frequency can be calculated by setting these two failure rates equal and applying assumption 4 from above.

ATTACHMENT 2
ST-HL-AE-3682
PAGE 6 OF 25

$$\lambda_s \simeq \lambda(V_1)^* P(V_2|V_1)^*(1+P_l) \tag{6}$$

Because only a minute amount of leakage is required to pressurize the space between valves, it is assumed that $P_l$ approaches 1.0. Therefore,

$$\lambda_s \simeq 2^* \lambda(V_1)^* P(V_2|V_1) \tag{7}$$

Examination of Equation (7) and its preceding derivation shows that this formulation of the combined failure rate model is quite conservative. It accounts for the fact that leakage through valve $V_1$ is quite likely to pressurize the space between the valves, but it also assumes that both valves are always exposed to full reactor coolant system differential pressure. If one valve fails, it is assumed that the second valve is exposed to a sudden differential pressure pulse.

Given that $V_1$ has failed independently, $V_2$ could fail on demand (due to the sudden pressure challenge) or it may fail randomly in time, sometime after failure of $V_1$. The latter failure mode is represented by the standby redundant system model.

The term $P(V_2|V_1)$ in Equation (7) contains two components: one representing random failures of the second valve, given that the first valve has failed, and the second representing a demand failure at the time that the first valve failed.

The determination of the frequency of occurrence of random failures is facilitated by assuming that the two series check valves in each path represent a standby redundant system, and that failure of the downstream check valve cannot occur until failure of the check valve nearest to the reactor coolant system loop has occurred. The probability of random failure (unreliability) for a single injection path is given by

$$Q_{path} \simeq 1 - e^{-\lambda t}(1 + \lambda t) \tag{8}$$

where $\lambda$ is the appropriate failure rate of a single check valve. In this study, $\lambda$ is the frequency of exceeding leakages of 120 gpm. This expression was then used to derive a failure (or hazard) rate for the path; that is,

$$\lambda_{path}(t) = \frac{-1}{(1-Q_{path})} \frac{d}{dt} [1-Q_{path}] \tag{9}$$

or

$$\lambda_{path}(t) = \frac{\lambda}{(1+\frac{1}{\lambda t})} \tag{10}$$

As noted earlier, the plant is expected to go to cold shutdown once every 18 months, at which time these valves will be inspected. If it is determined that the system is not functioning, it is repaired at that time. Therefore, the time-dependent failure rate is bounded at 1 year. The average failure rate over a time period, T, is given by

$$< \lambda_{\text{path per reactor-year}} > = \frac{1}{T}\int_0^T \frac{\lambda dt}{(1+\frac{1}{\lambda t})}$$
$$= \frac{1}{T}\left[\lambda T - \ell n(1+\lambda T)\right] \tag{11}$$

When $\lambda T << 1$, this result can be expanded to obtain

$$< \lambda_{\text{path}} > = \frac{1}{2}\lambda^2 T \tag{12}$$

The demand component of the path failure frequency is merely the product of $\lambda$ and the demand failure rate, $\lambda_d$. Thus,

$$< \lambda_{\text{path}} > = \lambda\left[\frac{\lambda T}{2} + \lambda_d\right] \tag{13}$$

Finally, the above expression for $< \lambda_{\text{path}} >$ is multiplied by a factor of 2 to account for the logic used in developing Equation (7). This logic is that the two valves can fail in either sequence because of an assumed high likelihood of inboard valve leakage and pressurization of the space between valves. Thus, the final expression for the series valves in the injection lines is

$$< \lambda_{\text{path}} > = 2\lambda\left[\frac{\lambda T}{2} + \lambda_d\right] \tag{14}$$

**Failure of Check Valve RH0065 or SI0030**

If the leakage through check valves SI0038 and RH0032 exceeds the capacity of the RHR relief valve, the RHR heat exchanger and its connected piping will pressurize. It is assumed for this analysis that the heat exchanger and piping will survive these pressurization transients and that the disks for low pressure isolation check valves RH0065 and SI0030 are the weak points in the system pressure boundary. This assumption provides a conservative upper bound for the frequency of containment bypass events. If the heat exchanger shell or its connected piping fails before either of these check valves, the leakage will be confined to the containment, and the resulting scenario is identical to a small or medium LOCA event.

Check valves RH0065 and SI0030 are rated to withstand approximately 600 psig. The check valves are the same size and are essentially identical in construction. Therefore, when the RHR heat exchanger piping is pressurized, it is equally likely that either check valve will fail. If valve RH0065 fails first, it is assumed that the RHR pump seals or other portions of the RHR piping will fail, and the resulting leakage will be confined to the containment. If valve SI0030 fails first, it is assumed that the LHSI pump seals or other portions of the LHSI piping will fail, and the leakage will flow outside the containment. It is assumed for this analysis that one of these check valves will fail during every event that pressurizes the RHR heat exchanger piping. This assumption may also be quite conservative because standard design criteria require these check valves to hold pressures well in excess of their nominal 600-psig rating. However, these criteria do not extend to full RCS pressure.

Since it is equally likely that either check valve will fail when the RHR heat exchanger piping is pressurized, the model uses a conditional frequency of 0.5 to represent the fraction of leakage events that bypass the containment through valve SI0030.

**Failure To Isolate Leak before RWST Is Drained**

Motor-operated valve RH0031 is designed to withstand full-rated RCS pressure. It is normally open, and its power supply is locked open at the MCC. Any leakage through check valves SI0038 and RH0032 will open the RHR relief valve and will cause control room alarms from increasing temperature and pressure in the pressurizer relief tank. The operators can then determine which valves have failed by checking the individual pressure and temperature indicators for each RHR heat exchanger line. When the affected line has been identified, the emergency procedures instruct the operators to locally restore power to valve RH0031 at the MCC and to close the valve. The valve can be closed from either its local controls at the MCC or its switch in the main control room.

If valve SI0030 has failed, RCS leakage is flowing outside the containment. Depending on the size of the leak, high pressure or low pressure injection will continue until the RWST is drained. Since no water is available in the containment sump, subsequent recirculation cooling is not possible. Therefore, if valve RH0031 is closed before the RWST is drained, the leak will be stopped while sufficient coolant inventory remains to prevent core damage.

**Total Frequency of Containment Bypass Events That Lead to Core Damage**

The expression for a containment bypass sequence in any of the three LHSI injection lines can then be written as:

$$Q_v = 3 * 8760 * \lambda[\lambda T + 2\lambda_d] * 0.5 * [HE + Q_d] \qquad (15)$$

where

$Q_v$ = the annual frequency of containment bypass events that lead to core damage (events/year).

$\lambda$ = the failure rate for excessive leakage or rupture of check valve SI0038 or check valve RH0032 (events/hour).

$T$ = the exposure time between tests of check valves SI0038 and RH0032, the time between refueling outages (18 months = 13,140 hours).

$\lambda_d$ = the conditional failure rate for rupture of check valve SI0038 or check valve RH0032 during a pressure pulse caused by failure of the companion valve (failure/demand).

0.5 = the conditional frequency that check valve SI0030 fails before check valve RH0065 (failure/demand).

HE = the error rate for operator failures to correctly diagnose the cause for the control room alarms and close valve RH0031 before the RWST is drained (error/demand).

$Q_d$ = the hardware failure rate for valve RH0031 failures to close on demand (failure/demand).

The initial factor of 3 accounts for the three LHSI injection lines that may experience these failures. The factor of 8,760 hours per year converts the hourly frequency of these events to an annual initiating event frequency.

ATTACHMENT 2
ST-HL-AE-3482
PAGE 10 OF 25

## FAILURE DATA

Figure 2 is taken from Reference 2. It displays a set of curves that estimate the frequency of check valve leakage failures as a function of the leak rate. For this analysis, it is assumed that a containment bypass event will progress to core damage only if the total leakage rate exceeds normal charging flow capacity. The nominal capacity of one charging pump at normal RCS pressure is approximately 120 gpm. If the leak rate is less than this amount, charging flow will maintain normal pressurizer level. The charging pump suctions automatically transfer to the RWST when the VCT is drained. The normal inventory in the RWST is sufficient to maintain this amount of charging flow for approximately 2 days without any additional makeup. Therefore, failures of check valves SI0038 and RH0032 are of concern for this analysis only if the total leakage rate exceeds approximately 120 gpm. Figure 2 shows that the relevant median failure rate for each check valve is approximately 1.45E-08 failures per hour.

- Check Valve SI0038:

| | | | |
|---|---|---|---|
| – Develops Leak > 120 gpm | Mean: | 4.00E-08 | failure/hour |
| | 5th percentile: | 1.40E-09 | |
| | 50th percentile: | 1.45E-08 | |
| | 95th percentile: | 1.45E-07 | |
| – Fails To Hold under Pressure Pulse | Mean: | 2.26E-04 | failure/demand |
| | 5th percentile: | 2.66E-05 | |
| | 50th percentile: | 1.37E-04 | |
| | 95th percentile: | 6.82E-04 | |

- Check Valve RH0032:

| | | | |
|---|---|---|---|
| – Develops Leak > 120 gpm | Mean: | 4.00E-08 | failure/hour |
| | 5th percentile: | 1.40E-09 | |
| | 50th percentile: | 1.45E-08 | |
| | 95th percentile: | 1.45E-07 | |
| – Fails To Hold under Pressure Pulse | Mean: | 2.26E-04 | failure/demand |
| | 5th percentile: | 2.66E-05 | |
| | 50th percentile: | 1.37E-04 | |
| | 95th percentile: | 6.82E-04 | |

- Check Valve SI0030:

| | | | |
|---|---|---|---|
| – Fails To Hold under Pressure Pulse | Guaranteed Failure | (1.0) | failure/demand |

- Check Valve RH0065:

| | | | |
|---|---|---|---|
| – Fails To Hold under Pressure Pulse | Guaranteed Failure | (1.0) | failure/demand |

- Motor-Operated Valve RH0031:
  - Fails To Close on Demand

| | | | |
|---|---|---|---|
| Mean: | 4.30E-03 | failure/demand |
| 5th percentile: | 7.49E-04 | |
| 50th percentile: | 2.84E-03 | |
| 95th percentile: | 1.05E-02 | |

The model conservatively assumes that either check valve RH0065 or check valve SI0030 will fail every time that the RHR heat exchanger piping is pressurized above 600 psig. A conditional frequency of 0.5 is used in Equation (15) to account for the fraction of these events during which valve SI0030 fails before valve RH0065.

## OPERATOR ACTIONS

After failure of the first two check valves in the high pressure section of the injection lines, several control room indications are available to alert the operators that reactor coolant has leaked into the RHR heat exchanger line. The RHR relief valve discharges into the pressurizer relief tank. When the relief valve opens, the control room operators will receive alarms for increasing temperature, pressure, and level in this tank. Each RHR heat exchanger line also contains temperature and pressure instrumentation with displays in the main control room. These indications can be used to quickly determine which line has been affected.

The emergency operating procedures will instruct the operators to close valve RH0031 after they have identified the specific line with the leaking valves. To close valve RH0031, an operator must first locally restore power to the valve at its MCC. The valve can then be closed either from the local controls at the MCC or its switch in the main control room. It is conservatively estimated that these actions may be completed within approximately 30 minutes after the operators make the decision to close the valve.

The available time window for diagnosing the specific cause for the control room alarms and closing the correct isolation valve depends on the size of the leak and the operators' familiarity with the emergency procedures for these events. To effectively terminate the leak before the core begins to uncover, the operators must close valve RH0031 before the RWST is completely drained. If the leak is at the lower end of the range for this analysis (approximately 120 gpm), normal inventory in the RWST is sufficient to maintain high pressure injection flow for approximately 60 hours without any additional makeup. The largest leak possible for this scenario is limited by an 8"-diameter hole (the size of the LHSI piping). In this case, with all of the high head safety injection (HHSI) and LHSI pumps operating, normal RWST inventory will maintain injection flow for at least 1 hour. (Containment spray flow is not actuated if the leakage is flowing outside the containment.)

A detailed human reliability analysis was not performed for this screening evaluation. Three sensitivity calculations are presented that illustrate how this operator recovery action affects the total frequency of core damage caused by these containment bypass failures. A more thorough and realistic analysis would develop an integrated recovery model that probabilistically combines the possible ranges of leak rates, their corresponding occurrence frequencies, the associated operator response time windows, and a detailed evaluation of dynamic human response under each set of conditions.

The first sensitivity calculation uses a mean operator error rate of 1.0 to provide an absolute upper bound for the frequency of these events if no credit is given for operator recovery. The results from this calculation are clearly unrealistic, but they provide insights about the hardware failure contribution to these scenarios for comparison with other analyses of different plant configurations.

The second sensitivity calculation uses a conservative screening value of 0.1 for the mean operator error rate. A lognormal range factor of 3 is used to produce the error rate distribution shown below.

- Screening Value for Operator Failures To Close RH0031

| | | |
|---|---|---|
| | Mean: | 1.00E-01 |
| | 5th percentile: | 2.57E-02 |
| | 50th percentile: | 7.88E-02 |
| | 95th percentile: | 2.35E-01 |

Error rates in this range are appropriate for only the most limiting of these containment bypass scenarios in which all three check valves are fully open and the LHSI piping fails catastrophically. In all other cases, several hours are available for the operators to thoroughly investigate the alarms, review the appropriate procedures, and consult with offsite emergency response personnel.

The third sensitivity calculation uses a "reasonable" value of 0.01 for the mean operator error rate. A lognormal range factor of 5 is used to produce the error rate distribution shown below.

- "Reasonable" Value for Operator Failures To Close RH0031

| | | |
|---|---|---|
| | Mean: | 1.00E-02 |
| | 5th percentile: | 1.18E-03 |
| | 50th percentile: | 6.07E-03 |
| | 95th percentile: | 3.02E-02 |

Error rates in this range remain quite conservative for the most likely of these containment bypass scenarios in which injection flow continues for several hours before the RWST is drained. During these extended scenarios, several additional alarms and local observations in the fuel handling building would almost certainly identify the specific failed piping. If the scenario continues for more than 8 hours, additional control room operators are certain to arrive onsite. Under these conditions, it is quite likely that the operator error rate will be very low and that the inability to isolate the leak will be limited by hardware failures of valve RH0031.

ATTACHMENT 2
ST-HL-AE- 3682
PAGE 14 OF 25

## RESULTS

Three different results of this analysis are presented here:

1. After the failures of the check valves, no actions are taken to terminate the leak:

| | |
|---|---|
| Mean: | 1.73E-06 |
| 5th percentile: | 3.82E-09 |
| 50th percentile: | 8.72E-08 |
| 95th percentile: | 3.11E-06 |

2. After the failures of the check valves, a conservative upper-bound screening value is applied for the operator error rate to terminate the leak by closing valve RH0031:

| | |
|---|---|
| Mean: | 1.71E-07 |
| 5th percentile: | 2.79E-10 |
| 50th percentile: | 7.37E-09 |
| 95th percentile: | 2.85E-07 |

3. After the failures of the check valves, a "reasonable" value is applied for the operator error rate to terminate the leak by closing valve RH0031:

| | |
|---|---|
| Mean: | 2.28E-08 |
| 5th percentile: | 3.28E-11 |
| 50th percentile: | 9.43E-10 |
| 95th percentile: | 3.95E-08 |

The results from the third sensitivity calculation show that hardware failures of valve RH0031 begin to contribute significantly to the total frequency of isolation failure if the operator error rate is less than approximately 0.01.

## REFERENCES

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

2. Pickard, Lowe and Garrick, Inc, "Seabrook Station Risk Management and Emergency Planning Study," prepared for New Hampshire Yankee Division, Public Service Company of New Hampshire, PLG-0432, December 1985.

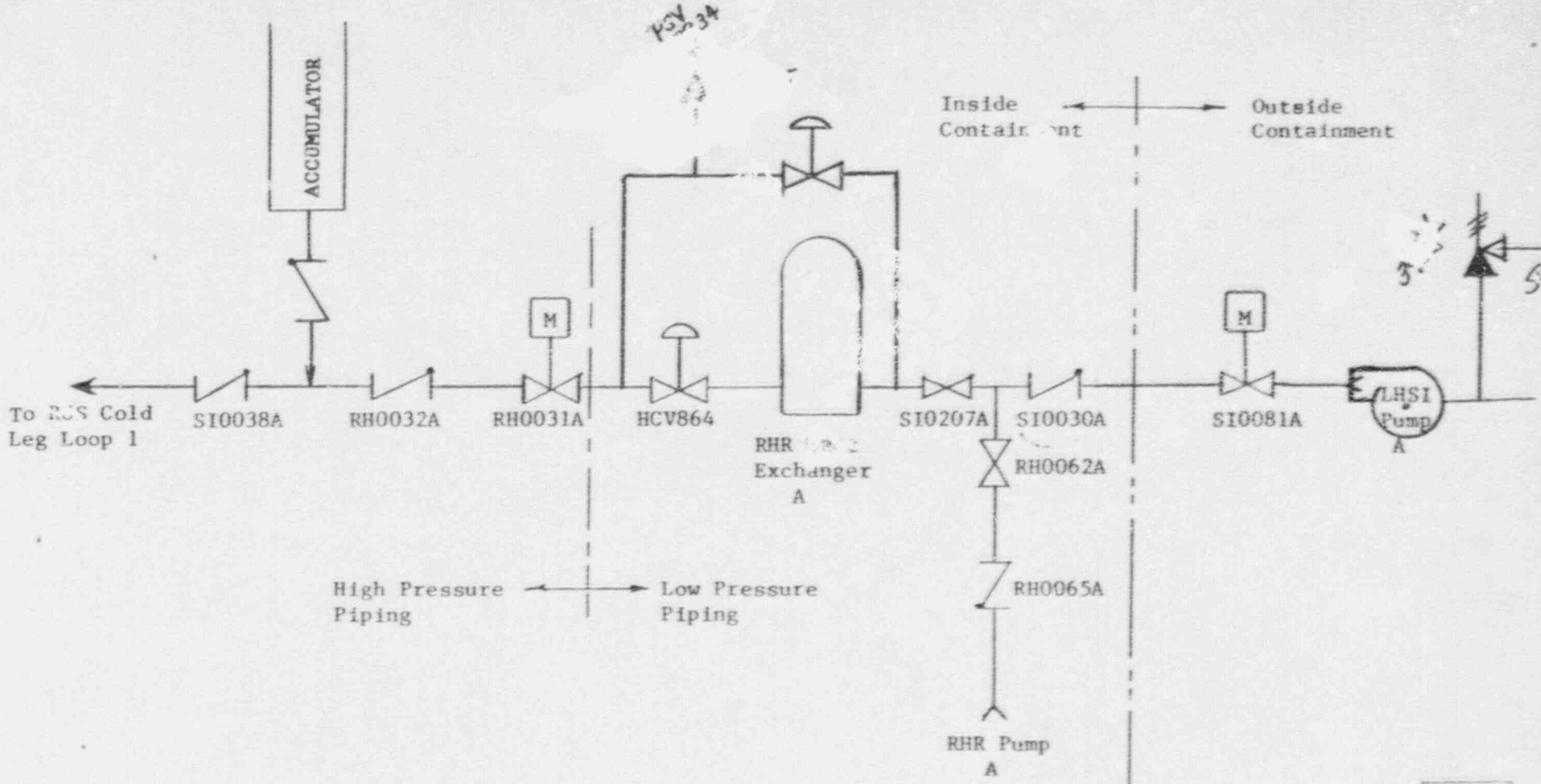



Figure 1. High/Low Pressure Piping Interface (Trains B and C Identical)

ATTACHMENT 2
ST-HL-AE-3682
PAGE 16 OF 25

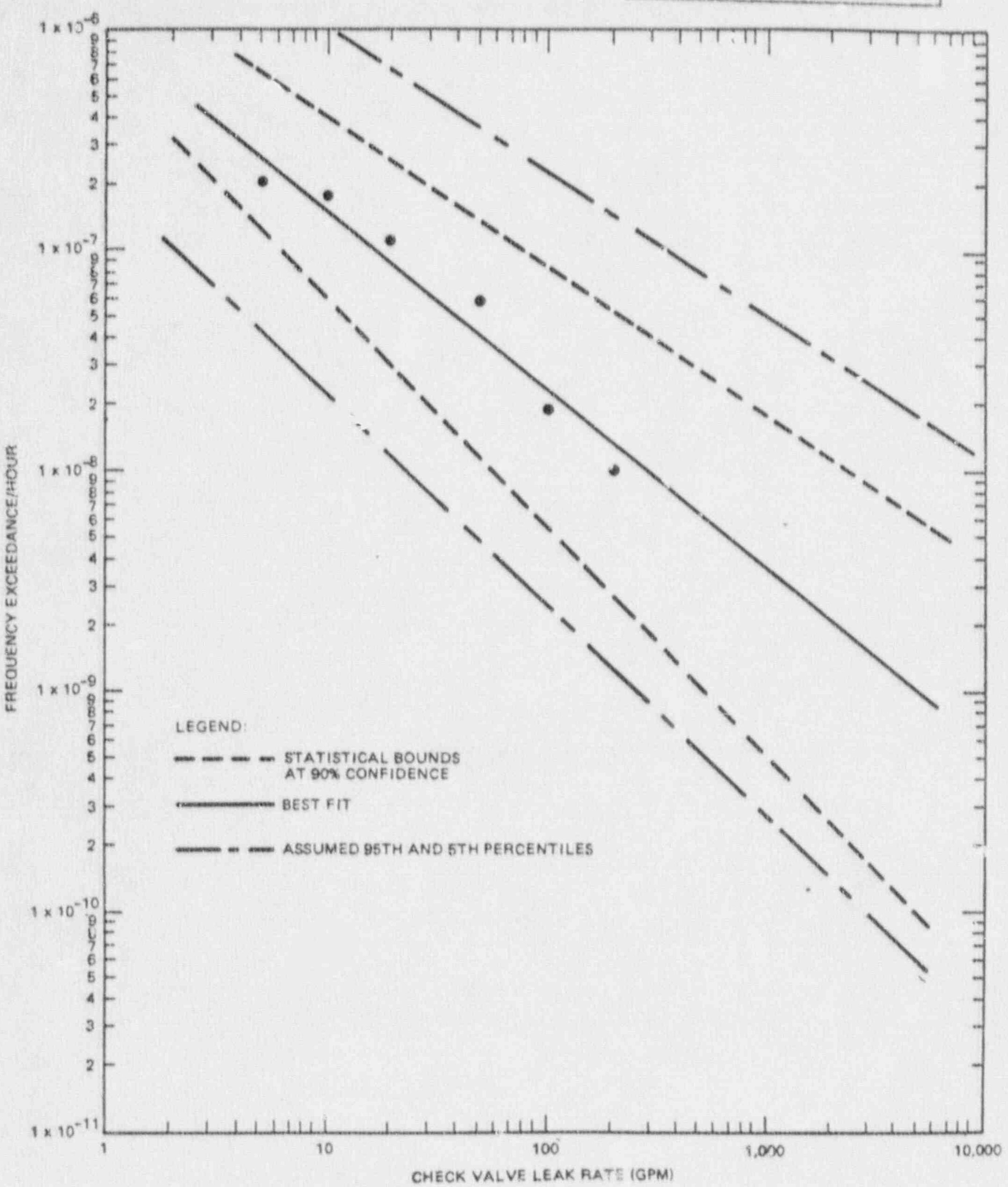



Figure 2. Frequency of Check Valve Leakage Events

Addendum to South Texas V-Sequence Analysis

Comment 1:

The latest analysis includes a simplified schematic of the ECCS injection lines. However, the analysis refers to several components which are not shown on the schematic. Using P&ID #5R169F20000, sheet 1, the pertinent missing components were located, and it was verified that the analysis correctly incorporates these components into the model.

Response:

Figure 1 included in the "South Texas V-Sequence Analysis" has been updated to include the major components discussed. Certain sensors such as for temperature and pressure alarms are not shown on this simplified diagram. For information, P&ID #5R169F20000, Sheet 1, is attached.

Addendum to South Texas V-Sequence Analysis

Comment 2:

The new analysis uses a mean frequency of random failure of a high pressure check valve of 4E-8/hr, compared to 5.4E-7/hr in the earlier analysis. The later value corresponds to data item ZTVCOL in the PSA data tables. The lower value of 4E-8/hr should be justified.

Response:

The basis for the value of the high pressure check valve failure rate of 4E-8/hr is now discussed on page 10 in the "Failure Data" section of the "South Texas V-Sequence Analysis". This value is consistent with the value of ZTVCOL used in the PSA after adjustment to include only valve leakage greater than 120 gpm.

Addendum to South Texas V-Sequence Analysis

Comment 3:

The new analysis incorporates operator actions to mitigate the impact of an interfacing systems LOCA. These operator actions are evaluated within the context of a good discussion of the temperature and pressure instrumentation available to the operators as indicators of the existence of a leak. The analysis estimates that operators will successfully isolate the leak 90% of the time without detailed procedural guidance, and 99% of the time with such guidance. The values for the operator failures for these two scenarios (guidance versus no guidance) appear to be reasonable, but there is no discussion of any HRA done to arrive at these values. Such a discussion would be helpful in validating this analysis.

Response:

The "Operator Actions" section on page 12 of the "South Texas V-Sequence Analysis" has been augmented to provide a better discussion of the basis for the assumptions relative to the operator failure rates used. It is also clarified that a detailed human reliability analysis was not performed for this screening evaluation. However, the factors identified in the "Analysis" and in this discussion are the same as those which must be considered in a more quantitative way in determining operator success or failure.

The reviewer has acknowledged a good discussion of temperature and pressure indications available to operators to diagnose the existence of a leak. In addition, there is a realistic discussion of the limits of time for diagnosis and for recovery actions to be taken before core uncovery may occur. An HRA would probabilistically combine the possible ranges of leak rates, their corresponding occurrence frequencies, the associated operator response time windows, and a detailed evaluation of dynamic human response under each set of conditions. The last of these is the purpose of the surveys which were performed for other situations evaluated in the PSA. Each of the other considerations has been qualitatively discussed in this "Analysis".

Of the three types of human actions evaluated in the PSA (see Section 15, Human Actions Analysis), this action would fall under the second type, i.e., scenario-specific, directed-mission activities. In this type of action, the operators must accomplish well-defined tasks for manual initiation, control, and alignment of plant emergency equipment. As noted in Section 15, these tasks are generally guided by the plant emergency response procedures, which consider the time window for successful response, the type of action that must be taken, and other factors that influence operator stress and confusion which are determined by the type of event being evaluated (i.e., performance shaping factors).

At STP, Emergency Operating Procedures (EOPs) provide guidance to the operators for identifying and isolating loss of coolant accidents (LOCAs) both inside and outside of containment. Plant operating procedure 0POP05-EO-EC12, "LOCA Outside Containment", provides specific guidance for isolation of a LOCA outside of containment. This procedure is entered from 0POP05-EO-E000, "Reactor Trip or Safety Injection", and 0POP05-EO-E010, "Loss of Reactor or Secondary Coolant", on abnormal radiation in the Fuel Handling Building due to a loss of RCS inventory outside containment. As discussed in the "Operator Actions" section on page 12, the operators will have up to 60 hours to take action for a leak at the lower end of the spectrum. In the worst case, operators will have an hour or more to take action.

Therefore, it is considered that the value of 1E-2 for operator failures to close RH-0031 is reasonable for this analysis for STP.

Addendum to South Texas V-Sequence Analysis

Comment 4:

The new analysis does not consider the possibility that the operator successfully acts to close the necessary MOV to mitigate the leak, only to have the MOV fail to close on demand. Since no data is provided for failure of the valve to close, it appears that the new analysis assumes that failure of the valve to close on demand is much less likely than failure of the operator to initiate the closure action. In most circumstances we would agree with this assumption, but it requires further justification in this instance. The valve in question (MOV RH-0031) may not be able to close against reverse flow unless its torque switches are set to allow for this flow. This is an important point since without successful mitigation, the V sequence frequency is comparable to other dominant core damage sequences, and should not be screened.

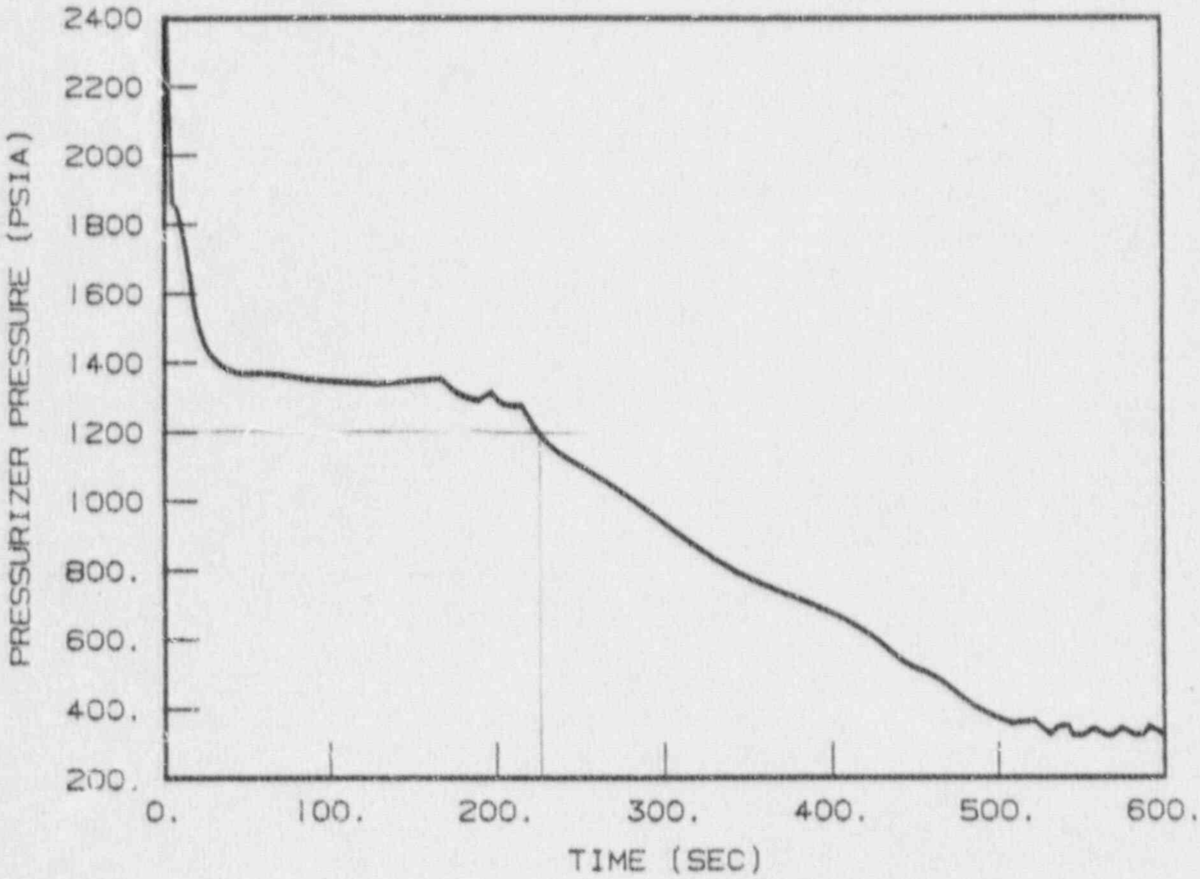
Response:

Failure of the MOV RH-0031 to close after the operator successfully acts to restore power and close the valve is included in the analysis. Inadvertently, the value of the demand failure rate, $Q_d$, shown in equation (15) on page 8 of the "South Texas V-Sequence Analysis" was omitted from the first draft. However, the quantified results correctly include this failure. The value of $Q_d$ used in this analysis is now given on page 11 of the "South Texas V-Sequence Analysis".

MOV RH-0031 is a spring-compensated 8" gate valve which has a 6.06" diameter opening and is designed to statically withstand a seat differential pressure equal to 2485 psi. Closure of MOV RH-0031 is determined by a limit switch for 95% of the stroke closed and then by a torque switch. At 2250 psid (pounds per square inch differential) and 95% of nominal bus voltage, the valve operator will close to at least 95% of stroke without consideration of torque switch set point. This results in approximately 98% gate closure without taking credit for gate closure on torque. That is, if the torque switches are not set to allow for this flow, failure of the valve to close beyond the 95% position will leave the valve in a substantially closed position (approximately 98% closed).

Figure 15.6-48 of the STPEGS UFSAR (attached) indicates that for a 6" break, the RCS depressurization transient would result in an RCS (saturation) pressure of approximately 1200 psia within about 220 seconds (3.7 minutes). The operator actions to close RH-0031 include manual actuation after locally restoring power at the MCC as described in the "Analysis", which will likely occur at greater than 3.7 minutes into the transient. The reduced RCS saturation pressure conditions are within the valves's capacity to close without consideration of valve closure on torque.

ATTACHMENT 2
ST-HL-AE-3482
PAGE 24 OF 25




**SOUTH TEXAS PROJECT**
**UNITS 1 & 2**

RCS DEPRESSURIZATION
TRANSIENT

Figure 15.6-48 Revision 0

Addendum to South Texas V-Sequence Analysis

Comment 5:

HLP presented three values for the V sequence frequency which correspond to three different mitigation scenarios:

1) No mitigation action taken (1.73E-06)
2) No specific guidance given (1.71E-07)
3) Specific guidance given (2.28E-08).

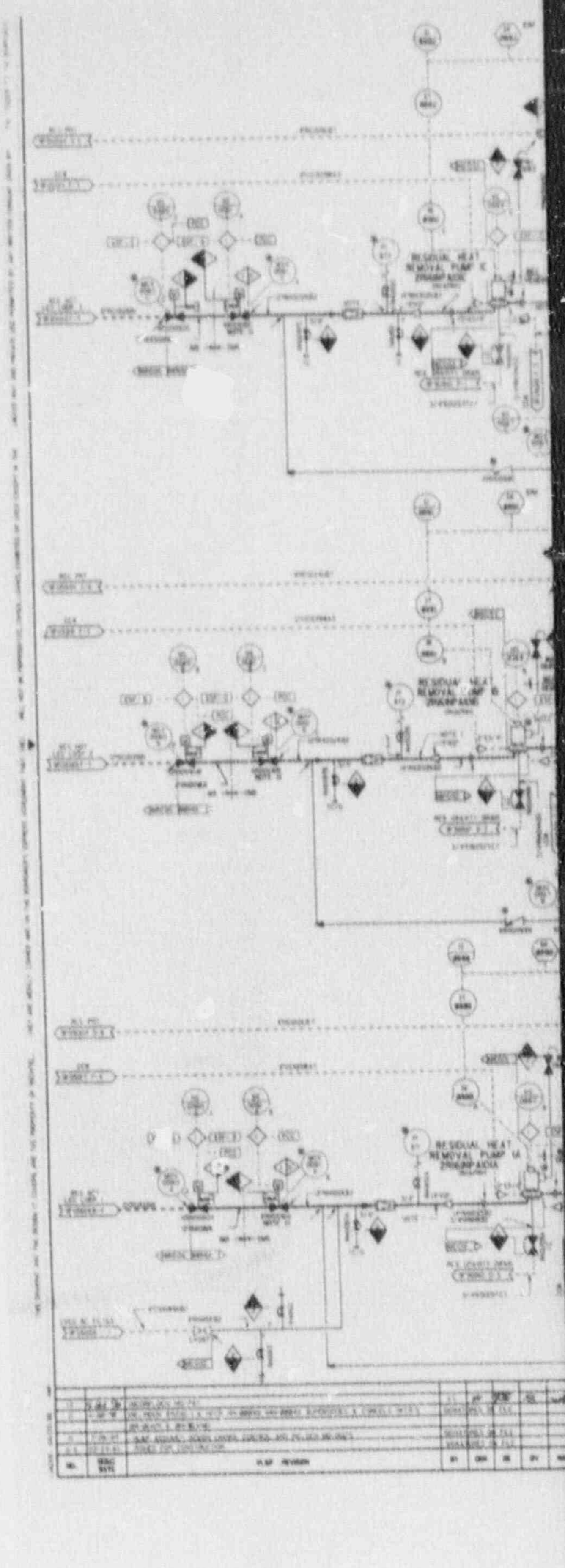
HLP did not make a final statement as to which scenario is most appropriate for their STP PSA. This should be clarified. Also, we recommend that units be clearly provided for all basic data and calculations. This should help to eliminate confusion between sequence frequencies, component failure rates, and failure probabilities.

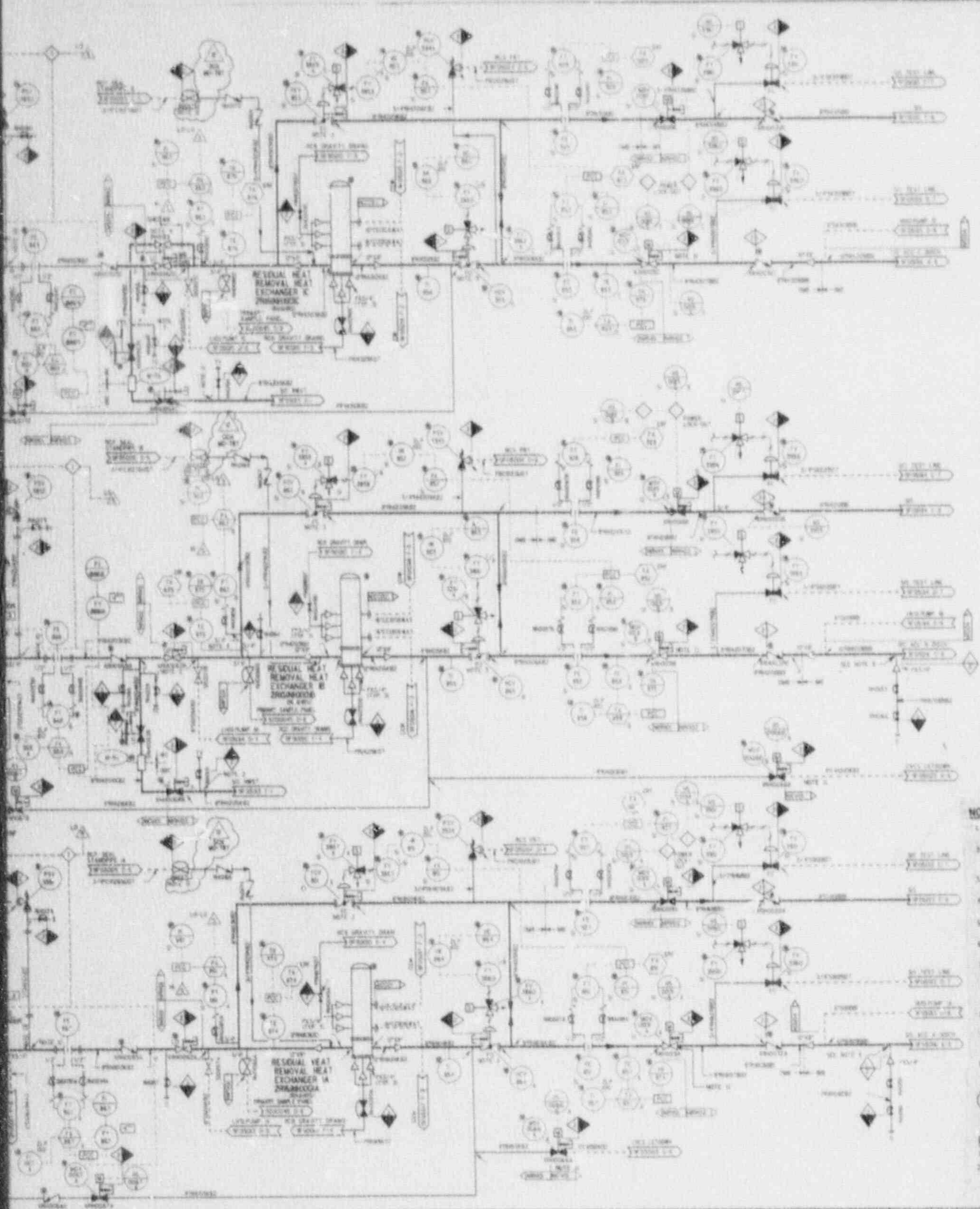
Response:

Further analysis will be performed as a part of the IPE process for evaluation of Accident Management Strategies (Supplement 2 of Generic Letter 88-20 dated April 4, 1990) to determine the scenario applicable to STP. The discussion provided in the response to comment 3 indicates that HL&P considers that specific guidance is given to operators for isolation of a LOCA outside of containment. The discussion provided in the response to comment 4 indicates that without consideration of torque switch set point, MOV-0031 can be expected to achieve at least 98% closure (i.e., to within approximately 0.7 square inches) against 2250 psid and would be expected to close under the more realistic saturation pressure conditions in the RCS for a break the size of the LHSI pipe outside of containment (i.e., 6 to 8 inches). Evaluations will be performed to determine the need for further action and, if necessary, identify and implement strategies for closure under other potential conditions.

The units for data and calculations are now shown in the "Analysis".

ATTACHMENT 2
ST-HL-AE-3682
PAGE 19 OF 25






**NOTES:**

1. SEE ... STANDARD DESIGN CRITERIA ... LATEST ...
2. LOCATE TEST CONNECTION AS CLOSE AS POSSIBLE TO ISOLATION VALVE. TEST CONNECTIONS ATTACHED TO BOTTOM OF PIPING TO FACILITATE DRAINING OF LINES.
3. MATING FLANGE FOR VALVE MUST BE ... REQUIREMENTS.
4. LOCATE ... AS CLOSE AS POSSIBLE TO THE ... TEST CONNECTION.
5. ALL INSTRUMENTS ON THIS P & ID HAVE SYSTEM DESIGNATOR ... UNLESS OTHERWISE NOTED.
6. ALL INSTRUMENTS ON THIS P & ID HAVE ... GROUP DESIGNATOR N UNLESS OTHERWISE NOTED.
7. UNLESS OTHERWISE NOTED, UNIT 1 IS SHOWN ON THIS DIAGRAM. FOR APPLICATION OF THIS DIAGRAM TO UNIT 2 USE UNIT 2 IDENTIFICATION NUMBERS FOR PIPING, VALVES, INSTRUMENTS, EQUIPMENT, ETC. REFER TO DWGS ... AND ... FOR IDENTIFICATION NUMBER DETAILS.
8. PROVIDE ... FLOW RESTRICTOR FOR TRANSITION FROM SAFETY CLASS 1 TO SAFETY CLASS 2.
9. FOR UNIT 2 CONNECTION ... IS LOCATED BETWEEN BRANCH CONN'S. FOR ... AND ...
10. BREAKOUT FLANGES ARE REQUIRED FOR PUMP REMOVAL FOR MAINTENANCE.
11. THIS VALVE IS POWER LOCKED OUT AT THE MCC.
12. STANDPIPE VALVE IS TO BE LOCKED CLOSED IN MODES 4, 5, & 6.

12 DCN ND-78'

**REFERENCES:**

1. FOR PIPING & INSTRUMENT SYMBOLS SEE DRAWINGS ... & ...
2. WESTINGHOUSE DRAWING NO. ... SHEETS 1, 2, & 3.

| | | | |
|---|---|---|---|
| BECHTEL ENERGY CORPORATION<br>HOUSTON, TEXAS | PIPING AND INSTRUMENTATION DIAGRAM<br>RESIDUAL HEAT REMOVAL SYSTEM | | |
| HOUSTON LIGHTING & POWER CO.<br>SOUTH TEXAS PROJECT | SCALE | JOB NO. 14926 | DRAWING NO. 5R069F20000 * | REV. 12 |

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ATTACHMENT 3

ATTACHMENT 3
ST-HL-AE-3682
PAGE 1 OF 9

# FIRE-RELATED QUESTIONS

## Question 1

Provide the basis for using the fire occurrence frequency for auxiliary buildings for the analysis of the STP cable spreading rooms instead of the frequency of fires in cable spreading rooms used in previous PRAs.

## Response

Five of the fire zones in the STP PSA fire risk screening analysis are typical of cable spreading rooms in other plants. These zones are Z010, Z026, Z047, Z057, and Z060. All of these zones are located in the mechanical and electrical auxiliary building (MEAB). Other zones in the MEAB are also predominantly populated by cable trays. However, these other zones include corridors, cable vaults, and cable penetration rooms that are similar to areas found in other plant auxiliary buildings. It seems reasonable to include these other areas in the population of fire zones allocated to the "auxiliary building" fire frequency and to include the five noted zones in the population of "cable spreading rooms."

Examination of Table 8.5-2 in the STP PSA (Reference 1) indicates some apparent discrepancies in the allocation of fire frequencies among these five cable zones. For example, on page 8 of Table 8.5-2, zones Z026 and Z047 are correctly included in the "Cable Spreading" category. The total annual frequency of cable spreading room fires (6.70E-03 fire per year) is distributed between these two zones. However, on page 2 of Table 8.5-2, zone Z047 is also included in the "Mechanical and Electrical Auxiliary Building" category with a correspondingly lower annual fire frequency. Zones Z010, Z057, and Z060 are also included in the "Mechanical and Electrical Auxiliary Building" category. The fire frequency of 1.07E-03 fire per year from page 2 of Table 8.5-2 was assigned to Zone Z047 in the quantitative screening analysis. Because of the time that has transpired and the unavailability of some key personnel who performed the original analysis, we are unable to reconstruct the reasons for theses apparent discrepancies.

A sensitivity study was performed to examine the quantitative effects from reassignment of the five questionable fire zones to the "Cable Spreading" category. The first step of this study was to determine an appropriate generic annual fire frequency for cable spreading rooms. It is noted that the generic database for cable spreading room fires includes three events (Reference 2). One of these events involved a relay fire. This event is not applicable for any of the five STP cable zones because none of these zones contain any relay cabinets. However, the event was retained in the database for this sensitivity study, and the generic annual fire frequency of 6.70E-03 fire per year shown on page 8 of Table 8.5-2 was conservatively used as the basis for these calculations. It has also been noted that many "generic" plants have somewhat less equipment and fewer cables than STP. To account for the possibility that STP has more cable spreading room area than a "typical" plant, the generic annual cable spreading room fire frequency was conservatively increased by 50% to yield a value of 1.01E-02 fire per year. This scaling practice is not typically applied in other fire risk analyses, and it was used in this sensitivity study only to provide an upper bound estimate for the quantitative effects from reevaluating these five fire zones. A more realistic analysis would remove the relay fire event from the generic cable spreading room fire database and would more carefully assess the actual cable spreading room area at STP compared with areas in "typical" two-train plants.

The scaled total annual cable spreading room fire frequency was allocated among the five STP fire zones according to their floor areas shown in Table 8.5-2. The results from this allocation are shown below and are compared with the annual fire frequency used for each zone in the original quantitative screening analysis.

| Zone | Area | Original Annual Fire Frequency | Revised Annual Fire Frequency |
|---|---|---|---|
| Z010 | 7,877 | 1.15E-03 | 2.48E-03 |
| Z026 | 7,907 | 3.48E-03 | 2.48E-03 |
| Z047 | 7,320 | 1.07E-03 | 2.30E-03 |
| Z057 | 5,779 | 8.46E-04 | 1.82E-03 |
| Z060 | 3,100 | 4.54E-04 | 9.74E-04 |
| **Total** | 31,983 | 7.00E-03 | 1.01E-02 |

It is interesting to note that the original total annual fire frequency for these five zones is very close to the unscaled "generic" cable spreading room value of 6.70E-03 fire per year. However, the allocation of this total among the zones is somewhat skewed by the different treatment of zone Z026. It is expected that a more realistic evaluation of the revised annual cable spreading room fire frequency (removing the relay fire event and appropriately scaling the generic frequency to account for the STP cable room area) would yield a total that is also close to this value.

The revised annual fire event frequency for each cable zone was next propagated through the quantitative screening process applied for all STP fire zones. This process is described in Section 9.3 of the STP PSA final report and in responses to previous review questions. The original fire zone screening analysis applied a quantitative criterion that stated that an end state was screened from further investigation if its estimated annual core damage frequency was less than one-tenth of one percent of the total core damage frequency from all other internal initiating events; i.e., less than 1.7E-07 event per year. The results from this sensitivity study indicate that one fire event scenario end state from zone Z010 and five end states from zone Z047 fail to meet the original quantitative screening criterion when the revised initiating event frequencies are applied. The frequencies of all end states from zones Z026, Z057, and Z060 remain below the criterion. The six end states and their revised estimated core damage frequencies are shown below.

| Zone | End State | Original Estimated Core Damage Frequency | Revised Estimated Core Damage Frequency |
|---|---|---|---|
| Z010 | 6 | 1.36E-07 | 2.93E-07 |
| Z047 | 53 | 1.44E-07 | 3.16E-07 |
| | 54 | 1.63E-07 | 3.58E-07 |
| | 90 | 9.05E-08 | 1.99E-07 |
| | 101 | 7.87E-08 | 1.73E-07 |
| | 107 | 9.16E-08 | 2.01E-07 |

Reduction factors to account for the fire zone geometry and fire severity were applied during the original screening analysis for only end state 53 from zone Z047. No reduction factors were applied for any of the other end states, and no additional reduction factors were applied for end state 53 during this sensitivity study. Based on experience from the original analyses, it is expected that application of conservative geometry and severity factors would reduce the frequency of each of the other end states well below the screening criterion.

The original fire event frequencies for zones Z010, Z047, Z057, and Z060 were derived from data for "auxiliary building" fires rather than "cable spreading room" fires. However, it is concluded from this sensitivity study that reallocation of the annual fire frequency for "Cable Spreading" areas among the five relevant fire zones at STP has a negligible quantitative impact on the results or conclusions from the original analysis. Only 6 of a total of 72 end states from these five zones failed to meet the original quantitative screening criteria after their frequencies were adjusted. The estimated total core damage frequency from these end states is less than one percent of the core damage frequency from all other internal initiating events.

Several significant sources of conservatism remain in the calculations performed for this sensitivity study. One of the three events in the generic database for cable spreading room fires involved a relay fire that is not applicable to the cable zones at STP. Removal of this event from the database would reduce the applicable generic annual fire event frequency. The generic annual fire event frequency was also arbitrarily increased by 50% to account for the possibility that STP contains significantly more cable areas than a "typical" plant. This assertion has not been confirmed. The practice of scaling generic fire event frequency data has also not been typically applied in other contemporary fire risk analyses. Conservative reduction factors to account for the fire zone geometry and fire severity have been applied during the analysis of only one of the six end states that fail to meet the quantitative screening criterion. It is expected that the application of similar conservative reduction factors to each of the other end states would reduce their frequencies well below the screening criterion. It is also expected that a more detailed assessment of end state 53 for zone Z047 would reduce its frequency. Based on this sensitivity study and its associated conservatisms, the conclusion that fires at STP are an insignificant contribution to the total frequency of core damage remains valid.

**References**

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

2. PLG, Inc., "Database for Probabilistic Risk Assessment of Light Water Nuclear Power Plants," PLG-0500, Volume 8, Fire Data, Revision 0, September 1990.

**Question 2.** Provide the basis for screening area Z032 from further analysis in the STP Internal Fire Analysis.

**Response**

The STP PSA spatial interactions analysis identified zone Z032 as a potentially important fire area. This is documented by inclusion of scenario Z032-FS-01 in the "List of Important Hazard Scenarios for Further Analysis in STP PSA," Table 8.6-7 of the STP PSA final report. However, this scenario was inadvertently omitted from the list of mechanical and electrical auxiliary building fires evaluated in Section 9.3 of the STP PSA report and from the list of control room fire scenarios evaluated in Section 9.4.

To consistently evaluate the potential risk significance from fires in this zone, a sensitivity study was performed for zone Z032, using the same methodology previously documented for all other fire scenarios listed in the STP PSA final report, Table 9.3-1. The most important equipment in this zone consists of the first row of cabinets and their associated cables. This row contains solid state protection system (SSPS) train R logic cabinet ZRR01, engineered safety features actuation system (ESFAS) train A actuation cabinet ZRR02, ESFAS train A test cabinet ZRR03, ESFAS train B actuation cabinet ZRR04, ESFAS train B test cabinet ZRR05, ESFAS train C actuation cabinet ZRR06, ESFAS train C test cabinet ZRR07, and SSPS train S logic cabinet ZRR08. All cabinets are separated from each other by double wall construction. An air gap of approximately 2 inches is also provided between each set of cabinets for different safeguards functions. (For example, there is an air gap between SSPS cabinet ZRR01 and ESFAS train A cabinets ZRR02 and ZRR03; there is also an air gap between ESFAS train A cabinets ZRR02 and ZRR03 and ESFAS train B cabinets ZRR04 and ZRR05.) There are no lateral penetrations between any cabinets in this row. All cables exit through either risers into the overhead cable tray network or floor penetrations into the cable spreading area on the next floor below.

All ESFAS train A cables exit cabinets ZRR02 and ZRR03 through the cabinet floors into the train A cable spreading room below. Some nonessential equipment cables (designated division "N") exit through the tops of these cabinets into the overhead trays. However, none of the overhead trays in this zone contain any cables that affect operation of safeguards train A equipment.

ESFAS train B cables exit through the tops of cabinets ZRR04 and ZRR05 into an overhead vertical stack of four horizontal cable trays that run parallel to the cabinet row and are offset approximately 8 inches to the east of the closest cabinet edges. These trays distribute the train B cables to risers on the south end of the room that penetrate the ceiling into the train B cable spreading room above.

ESFAS train C cables exit through the tops of cabinets ZRR06 and ZRR07 into an overhead vertical stack of four horizontal cable trays that run parallel to the cabinet row and are offset approximately 8 inches to the west of the closest cabinet edges. These trays distribute the train C cables to risers on the north end of the room that penetrate the ceiling into the train B cable spreading room above.

None of the train B cable trays pass over the train C cabinets or any of the trays containing train C cables. However, some of the train B trays are routed relatively close to and above the train A cabinets. None of the train C cable trays pass over the train A or train B cabinets or any of the trays containing train B cables. All cables in this zone, including the nonessential cables in division "N," meet the flammability criteria of IEEE Standard 383.

The spatial interactions analysis identified zone Z032 as potentially important because it was assumed that any fire in this area would completely disable all three trains of safeguards equipment and lead directly to core damage. This assumption is inappropriately conservative. A quantitative screening analysis was performed to more realistically estimate the potential core damage frequency contribution from fires in this zone. This analysis evaluated the effects from small cabinet fires, large cabinet fires, cable tray fires, and transient combustible fires based on data from the PLG fire event database. Propagation of extremely large cabinet fires to adjacent overhead cable trays was also considered.

During this screening analysis, all of the original fire frequency modification and reduction factors were reviewed for consistency with the sensitivity calculations performed for other fire zones in STP PSA Table 9.3-1. As a result of this review, the initiating event frequency for all fires in zone Z032 was revised from the value of 9.84E-05 fire event per year shown in STP PSA, Table 8.5-2, to a value of 5.90E-04 fire event per year. The higher frequency was then used as the basis for allocating fires among the cabinets and cable trays located in this zone.

The screening analysis results indicate that the largest core damage frequency contribution from any credible fire scenario in zone Z032 is approximately 4.0E-08 core damage event per year. This value is well below the quantitative screening criterion of one-tenth of one percent of the total core damage frequency from internal initiating events; i.e., less than 1.7E-07 core damage event per year. The most important fire scenario includes a large cabinet fire that damages the train A ESFAS cabinets and propagates to the nearest train B cable tray. It is assumed to cause a small LOCA due to short circuits that open pressurizer PORV PCV-655A, and it is assumed to disable all safeguards equipment in trains A and B.

It is noteworthy that the stuck-open PORV could be isolated by closing its motor-operated block valve MOV RC0001A. Operability of this valve is not affected by any fires in zone Z032. It is also noteworthy that fires in this zone can disable only automatic safeguards actuation signals and manual signals from the main control room switches. The operators could manually start and operate all necessary safeguards equipment from the auxiliary shutdown panels by disconnecting the normal control circuits at the switchgear room transfer panels. However, neither of these possible recovery actions were included in the screening analysis.

The results from this sensitivity study confirm the fact that fires in zone Z032 are negligible contributors to the frequency of core damage at STP. The quantitative impact from all fires in rooms classified within the control room envelope is completely dominated by the small set of main control panel fires evaluated in Section 9.4 of the STP PSA final report.

Question 3

Provide a discussion of the effect of weighting the fire initiating event frequency for personnel traffic on the overall fire contribution.

Response

The rules for allocating the frequency of MEAB fires among the individual MEAB fire zones are not documented in the STP PSA final report. However, the rules can be inferred by examination of the actual numerical frequency assignments. These rules are shown in Table 1.

**Table 1. Inferred MEAB Fire Zone Frequency Allocation Rules**

| Rule | Condition | $f_{mod}$ |
|---|---|---|
| 1 | Occupancy = "Cable" | 0.25 |
| 2 | Occupancy = "Cable, Cabinets" | 0.75 |
| 3 | Occupancy = "Piping"; Traffic <0.25 | 0.125 |
| 4 | Occupancy = "Piping"; Traffic >0.50 | 0.375 |
| 5 | Occupancy = "Power Cable" | 0.75 |
| 6 | Occupancy = "Power Cable, Cabinets" | 1.00 |
| 7 | Occupancy = "Power Cable, Cabinets, Battery" | 1.50 |
| 8 | Occupancy = "Power Cable, Switchgear" | 1.875 |
| 9 | Occupancy = "Pumps" | 1.50 |
| 10 | Occupancy = "Power Cable, Valves" | 1.00 |
| 11 | Occupancy = "Transient" | 0.125 |

The rules shown in Table 1 were applied directly to 95 of the 111 fire zones in the MEAB. The table shows that the zone traffic level enters the allocation rules only for zones whose primary occupancy consists of piping. These zones are

Z030, Z032, Z062, Z063, Z065, Z066, Z082, Z105

The first level of the screening analysis eliminated all of these fire zones as quantitatively insignificant. Since the traffic level does not enter into the frequency allocation for any of the remaining 87 zones, it can be concluded that the assessed traffic levels shown in Table 8.5-2 of the STP PSA final report have an insignificant impact on the overall fire risk contribution from these zones.

The rules documented in Table 1 were applied to 95 of the 111 MEAB fire zones. For the remaining 16 zones, additional modification factor adjustments were made to account for zone-specific conditions. These 16 zones are

Z006, Z019, Z023, Z028, Z033, Z061, Z093, Z096, Z104, Z117, Z123, Z124, Z125, Z141, Z142, Z143

None of the numerical adjustments to these zones are very large. The first level of the screening analysis eliminated 14 of these fire zones as quantitatively insignificant. The remaining two zones, Z006 and Z142, were evaluated more extensively in the second and third levels of screening. For zone Z006, an adjusted final modification factor of 1.50 was applied. This factor is higher than the factor of 1.00 that is normally assigned to this type of zone. Therefore, the estimated fire event frequency for this zone in the STP PSA is approximately 50% higher than the frequency that would be calculated by other methods. The quantitative screening evaluation for zone Z006 has shown this zone to be an insignificant contributor to the overall risk from fires. For zone Z142, an adjusted final modification factor of 0.75 was applied. This factor is somewhat lower than the factor of 1.00 that is normally assigned to this type of zone. The detailed fire scenario end states for this zone were reexamined to determine the effects from increasing the initiating event frequency by 33%. All end state frequencies remain below the applied quantitative screening criterion of one-tenth of one percent of the core damage frequency from all other internal initiating events; i.e., less than 1.7E-07 event per year.

Based on these observations, it is concluded that neither the assessed traffic levels documented in Table 8.5-2 of the STP PSA final report nor the additional adjustments to the 16 specific fire zone frequency allocation factors have a significant impact on the overall contribution of fires to core damage at STP.