PSNH

Public Service of New Hampshire

New Hampshire Yankee Division

SEABROOK STATION
Engineering Office

October 31, 1986

SBN- 1225
T.F. B7.1.2

United States Nuclear Regulatory Commission
Washington, DC 20555

Attention: Mr. Steven M. Long, Project Manager
PWR Project Directorate No. 5
Division of PWR Licensing - A

References:
(a) Facility Operating License NPF-56, Docket No. 50-443
(b) USNRC Letter, dated October 8, 1986, "Request for Additional Information for Seabrook Station, Units 1 and 2, Emergency Planning Sensitivity Study", S. M. Long to R. J. Harrison
(c) USNRC Letter, dated October 23, 1986, "Request for Additional Information for Seabrook Station, Units 1 and 2, Emergency Planning Sensitivity Study", S. M. Long to R. J. Harrison

Subject: Response to Request for Additional Information (RAIs)

Dear Sir:

Enclosed herewith are the majority of the responses of the Requests for Additional Information forwarded in References (b) and (c). Attachment A identifies responses that are included in this transmittal. Attachment B is the responses.

An additional submittal addressing the remainder of the RAIs will be forthcoming in the near future.

Very truly yours,

John DeVincentis

John DeVincentis
Director of Engineering

Attachment

cc: Atomic Safety and Licensing Board Service List

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United States Nuclear Regulatory Commission
Washington, DC 20555

A045
1/1

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ATTACHMENT A

Responses are included in this transmittal for the following RAIs:

1	12	23	41	55	69
2	13	24	42	57	70
3	14	25	43	59	71
4	15	26	44	60	72
5	16	28	45	61	73
6	17	33	46	62	
7	19	34	49	63	
8	20	35	50	64	
9	21	40	51	67	
10	22		53	68	
11					

Responses to the following RAIs will be forthcoming in an additional submittal:

4	36	54
18	37	58
27	38	65
29	39	66
30	47	74
31	48	75
32	52	

ATTACHMENT B

RESPONSES TO REQUESTS FOR INFORMATION

RAI 1

Describe how the overpressurization calculations made by SMA were checked or design reviewed.

RESPONSE 1

All SMA calculations on the Seabrook overpressure capacity were performed by Dr. Ralf Peek, currently on the Dept. of Civil Engineering faculty, University of California Berkeley. All assumptions and methods of analysis were reviewed by Dr. D.A. Wesley. Numerical checks of calculations for the capacities of the controlling failure modes were conducted by Dr. Wesley and other members of the SMA staff. The SMA report was reviewed by Dr. Torri and the SSPSA technical review board. These reviews were conducted and documented as part of a QA program that is described in Section 1 of SSPSA.

In addition a numerical check of all calculations and an independent review of assumptions is currently in progress. This independent review is scheduled for completion on November 25th, 1986.

RAI 2

A meeting should be arranged with the originator of these calculations to assist the BNL reviewers in following these calculations and understanding the assumptions.

RESPONSE 2

Assumptions need to be made in all engineering calculations. The difference between the usual design calculations and this probabilistic evaluation of the containment integrity is that here the median ultimate strength needs to be estimated. This requires evaluating the load carrying capacity in a limit state condition in which integrity of the liner is lost. In most cases this involves large inelastic deformations and extensive redistribution of stresses. This requires different assumptions compared to design code calculations, and these assumptions must be based on an understanding of the behavior of the structure in its ultimate condition.

The intent of the calculations is to document in a conventional manner what was done and what assumptions were made, rather than to explain in detail how the structure is expected to behave, and why the various assumptions are justified. Dr. Wesley and Dr. Peek participated in a detailed discussion of the calculational methodology with your staff on October 16th and 17th at BNL.

RAI 3

Document the basis for the assumptions in the calculations. In particular, explain the uncertainty factors assigned to various pressure capacities.

RESPONSE 3

The codes and standards used in design assure large margins of safety beyond the design and test pressures. However ultimate capacities are not computed in design. Virtually all design calculations are limited to elastic conditions whereas ultimate load response involves large nonlinear effects. Consequently, only limited data are available to quantify the uncertainties associated with many areas of the calculated ultimate pressure capacities. Uncertainty is introduced due to inexact knowledge of the structures' material properties as well as the structure behavior at extreme loads. Where data exist, as for instance on material test properties for Seabrook, it is a straight-forward procedure to introduce it into the overall variability for a given failure mode. Other areas such as expected accuracy of analytical methods are estimated based on the judgement of experienced engineers, and are so noted in the calculations and the report.

In addition, as noted in the response to question 1, the bases for assumptions will be independently verified.

RAI 4

Explain the mechanism for transferring the load from the penetration sleeves to the containment wall, inparticular, the equipment hatch, when subjected to high strain conditions. Explain how the rebars around the penetrations were assessed to assure that they can resist these loads in addition to the primary pressure induced loads.

ANSWER 4

At the ultimate containment pressure of 216 psi, the force per unit length along the circumference of the equipment hatch penetration that must be transferred from the sleeve to the containment wall is 18 kips/in. The 1 3/4 in x 10 in annular plate shown in the calculations (Fig 14.5, Section 14, Part I, as reproduced from FSAR report), is more than sufficient to support this load, thus preventing a blowout of the penetration.

A typical calculation demonstrating that small diameter penetration sleeves do not punch thru the containment wall at ultimate containment capacity will be provided with the response to RAI 31.

RAI 5

The calculations use a rebar ultimate stain value of 4.7%, i.e., more than 21 feet of linear extension for the hoop bars. This linear extension under the high pressure load will be accommodated by formation cracks in the concrete totaling approximately 21 feet in width. Justify the assumption that the pressure loads will be carried proportionately by the linear plate and the rebars (similar to the elastic condition) in this highly cracked condition. Also address the potential for developiong a crack large enough for the local extension of the liner plate to lead to its failure at that point.

RESPONSE 5

Since the strain is distributed in the steel and concrete between the cracks, the total elongation is not all accumulated as open cracks. The reinforcing steel assures the cracks will be essentially uniformly distributed and the crack growth controlled until failure. Cracking of the concrete and strains above yield in the liner and reinforcing steel assure that the local discontinuity elastic stresses are proportionately less important at internal pressure which result in a stress condition above yield. Both the reinforcing steel and liner are ductile materials with flat stress-strain curves in the range of ultimate strength. This assures that brittle failures will not occur in isolated elements and the proportional strengths of the rebar and liner can be relied until failure.

The potential for significant liner strain concentration in the vicinity of concrete cracks is considered highly unlikely. The surface of the reinforcing bars is deformed in order to create a bond with the concrete whereas the liner is smooth and is in contact with the concrete on only one side. At the failure pressures of interest, the required coefficient of friction between the liner and the concrete necessary to achieve the same concentration of strain in the liner as in the reinforcing steel is not considered credible, even if the thermal strains in the liner are neglected. Inclusion of the thermal strains will increase the required coefficient of friction and for the liner to fail prior to the rebar significantly higher strains must accumulate in the liner since it has higher elongation.

RAI 6

Was compatibility of strains in the rebars and the liner plate satisfied in the calculations? For example, the outermost hoop bars will fail before the inside bars and the liner plate reach their respective ultimate strengths. Was this fact reflected in the calculations? In addition how is the biaxial stress-strain state of the liner plate considered.

RESPONSE 6

The controlling modes of failure in the concrete structure occur due to membrane tension. For these modes, all reinforcing bars in a given direction have essentially equal strain at failure and the outermost hoop bars will not fail before the inside bars. The inner bars are likely to have somewhat lower strengths due to the higher temperatures toward the inside surface of the containment. However, the strain at which the ultimate strengths of these bars is attained is not significantly altered for temperature increases up to 700^{o}F. The bi-axial stress-strain state of the liner was considered in the calculations. A factor of 1.73 was computed to account for the expected change in elongation properties for the liner.

RAI 7

The combined tension, shear and bending effect at base and spring line levels was not considered in the calculations (Ref. p. 35, assumption 6). Verify that the combined effect does not change the conclusions of the analysis.

RESPONSE 7

Bending and shear effects occur not only at the springline, but wherever the hoop reinforcement content changes. However, these stresses are secondary in nature. This means that these stresses are not necessary to carry the internal pressure loads, they merely arise from displacement incompatibilities in the membrane solution. It was judged that such displacement incompatibilities could be accommodated if necessary by plastic rotations of the containment wall (flexural yielding due to meridional bending stresses) without significant loss in the load carrying capacity of the containment wall. The hoop bars especially, which are the most critical for carrying the membrane stresses, are unaffected by these plastic rotations.

Furthermore, the plastic rotations limit the shear forces which are developed. This is important because whereas plastic rotations are not expected to affect the integrity of the liner, the effects of extensive shear cracking could be more detrimental. The possibility of failure due to secondary shear stresses was considered for the junction between the base slab and the containment wall. This is the location where such secondary shear stresses were judged to be most critical. The estimated median pressure capacity for this failure mode is 408 psig as indicated in Table 3-3 of the SMA report. Therefore failure modes involving secondary stresses in the containment wall and dome are not considered critical.

RAI 8

Since 31 cadwelds out of a total of 169 test samples failed at a stress lower than the rebar ultimate strength and there was apparently a construction problem concerning staggering of these welds, provide justification for not using a reduced ultimate strength for the rebar.

RESPONSE 8

There was no construction problem involving the staggering of cadwelds in the containment shell. Cadwelds are distributed throughout the structure as specified in design.

Failure of the containment shell in the controlling hoop direction is expected to begin with vertical cracks forming in the concrete. These cracks are expected to initiate at locations of vertical, rolled section steel supports for the liner which are spaced uniformly approximately 20 inches apart around the circumference. This results in approximately 260 vertical cracks. As the pressure is increased, these cracks wil continue to open until the ultimate capacity of the combined reinforcing bars and liner plate is reached across the crack with the lowest capacity. The location of this crack cannot be predicted since the available information predicts essentially equal probability of failure of any crack location.

However, it is important to realize that failure does not occur due to failure of a single bar. The reinforcing bars and liner develop a ductile system with the ability to provide significant load sharing and load redistribution between the liner and a large number of reinforcing bars. An estimate of the number of reinforcing bars over which failure of a single bar is averaged can be obtained by multiplying the six equivalent bars across the shell thickness by the meridional length required for the perturbation damp out.

For the shell in the post yield condition, this length is conservatively estimated at greater than twice the shell characteristic length, or over a hundred bars.

The cadwelds are staggered throughout the structure and there is no way to establish the location of those cadwelds with relatively high or low capacities, just as there is no way to establish the location of the crack with the lowest median rebar strength. From the cadweld test data available for Seabrook, less than 20% of the cadwelds can be expected to have capacities below the median reinforcing bar strength, whereas by definition, 50% of the reinforcing bars will have strengths below the median. Hence, the test data shows that a conservative value of the median strength was used. The relatively few cadwelds which may be expected to have lower capacities than the median reinforcing steel are only slightly weaker (less than 10% reduction). This corresponds approximately to the lower bound reinforcing steel strength so that the minimum cadweld strength and the minimum reinforcing steel strength may be expected to be approximately equal. Away from the crack, the stress in the rebar decreases due to the concrete bond, and the effect of the cad-weld strength does not influence the strength of the bar once the load in the weld is decreased to less than the strength of the rebar across the crack.

Even if the cadwelds were completely ineffective, the total reduction in the hoop capacity would be less than 1% due to the averaging effect of the adjacent bars. Because of the limited number of cadwelds, there is very little probability that more than one cadweld will be located in the same crack in the area where the averaging effect occurs, and an even smaller probability that two cadwelds with low capacities would be so located. Thus, the effect of a few randomly distributed cadwelds with less than the median reinforcing bar strength is considered neglibible, and this effect is accounted for in the variabilities associated with the various structural failure modes.

RAI 9

The containment analysis is based on an axisymmetric geometry and loading. This is not the case due to the presence of adjoining structures such as the fuel building and main steam and feedwater pipe chase. Identify these axisymmetric conditions and assess their impact on the failure modes and analysis.

RESPONSE 9

The effect of local non-axisymmetric conditions is not expected to effect the capacities computed for the axisymmetric failure modes (i.e. cylinder hoop and meridional membrane failures, dome membrane failure, and base slab bending and shear failures etc.). This is because the local effects damp out rapidly for the inelastic case. Local failure modes such as interference between the fuel storage building and the containment were evaluated and found to have significantly higher capacities than the controlling axisymmetric failure mode (hoop failure).

Also refer to response to RAI 13.

RAI 10

Only a sample of pipe penetrations are considered in some detail (X-23, X-26 and X-71). The justification to consider only these should be provided.

RESPONSE 10

Virtually all the piping isometrics were reviewed by members of the SMA staff and members of the Yankee Atomic staff. The purpose of this review was to identify the lines considered most likely to fail, based on support spacing from the penetration both inside and outside the containment. Based on this review "worst case" lines for a multiple pipe penetration, a thin wall (sch. 40) pipe penetration, and a thick wall (sch. 160) pipe penetration were selected. Evaluations of these "worst case" lines indicated that failure of the penetration (i.e. breach of liner integrity) was not likely to occur for the thin wall and multiple pipe penetrations, irrespective of relative displacement, although fluid leaks or flow restriction in the pipes could occur. Thus, any further investigation of lines associated with this type of penetration was unnecessary. The evaluation indicated that sufficient force could be generated in a thick-walled pipe to potentially fail the penetration. However, there are only a few schedule 160 pipes entering the containment and the expected leak path area associated with this type of failure is so small that even if all schedule 160 pipe penetrations should fail, the resulting leak area is insufficient to prevent a continuing increase in internal pressure with eventual failure from an independent failure mode. Therefore detailed evaluations of individual penetrations were not warranted, once the controlloing penetrations were identified.

RAI 11

A structural evaluation of electrical penetrations should be provided.

RESPONSE 11

The electrical penetrations were reviewed and it was determined their pressure capacity would not be a controlling mode of failure for the following reasons:

- penetration are not subject to rigid pipe reaction which interact with containment wall dispacement
- it was judged that the failure of these penetrations is dominated by thermal effect (i.e. leaks)
- a thermal analysis was performed in lieu of a structural analysis

Consequently, a detailed evaluation was not conducted in order to concentrate on the more likely modes of structural failure modes. This is consistent with the overall approach used for both the seismic and overpressure capacity evaluations where detailed investigation is only conducted on conceivable modes of failure.

RAI 12

The basis for the leakage area assigned to the flued head at failure should be provided.

RESPONSE 12

The leakage area due to failure of the flued head is a rather uncertain quantity as indicated by the large variability assigned to it. As a median centered estimate, the flued head is expected to fail at a radius 0.5 in less than the inner radius of the sleeve. With a slight increase in containment pressure and the associated radially outward displacement of the containment wall, the part of the fluid head which remains attached to the pipe is pulled into the sleeve. As a result a clearance of 0.5 in is available all around, as a leak area. The logarithmic standard deviation of 0.6 indicates that the 95% confidence intervals for this clearance width are about 0.2 in and 1.3 in respectively. Thus no claim is made that this leak area can be accurately predicted. This uncertainty is taken into account in the probabilistic consequence analysis.

RAI 13

A more detailed evaluation of the impact of punching shear at the Fuel Transfer Building should be provided.

RESPONSE 13

Punching shear stresses at the Fuel Storage Building begin to develop only at a (median) containment pressure of 172 psi. This pressure, at which the containment wall begins to bear against the Fuel Storage Building, is not "very approximate". The main source of uncertainty for this pressure is the uncertainty of the effect of bonding on the stress-average strain relation for the hoop reinforcing bars, and could not be eliminated by a more detailed analysis. What is "very approximate" is the existing evaluation of the increase in containment pressure that could be supported after the containment pressure begins to bear against the Fuel Storage Building. The large uncertainty in this increment in internal pressure is reflected by the large logarithmic standard deviation assigned to this quantity, and has been included in the risk evaluation process. In doing so, it is found that the failure mode is not critical. This conclusion is not expected to change even if a more detailed analysis resulted in a slightly different value for the median pressure increment or slightly reduced uncertainties for the punching shear failure.

Furthermore, even if some of the uncertainty in modeling could be removed by a more detailed analysis, the uncertainty in strength would remain. Finally the lack of knowledge of the behavior of construction materials under such extreme loading condition renders the applicability of the results of the most sophisticated analysis open to debate and is not warranted for this evaluation.

RAI 14

Clarify the extent to which double ended piping failures have been considered in the overall containment performance assessment. Provide isometric drawings of all piping attached to containment penetrations.

RESPONSE 14

The possibility of a pipe failure on the inside and outside of the containment due to the pressure induced displacements of the containment wall was considered. For some of the thick walled pipes, it is likely that the penetration will fail prior to the pipe and thus relieve the load on the pipe. For thin wall pipes, the most likely mode of failure is bending of the pipe thus reducing fluid flow but not resulting in a leak. However, there is some possibility that fracture of thin walled pipes can occur, and a much reduced possibility that a fracture on both sides of the penetration can occur for a given pipe. This possiblity was considered for Penetrations X-23, X-26, and X-71 and found to have a negligible contribution to the overall risk. For the flued head pipes (Penetration X-8) however, the double-ended pipe break failure mode contributed about 55% of the total probability of this type of penetration failing before the concrete structure membrane failure mode.

Isometric drawings of all piping attached to containment penetrations were supplied to BNL in September 1986.

RAI 15

In PLG-0465, page 2-10, Figure 2-3, the conditional frequency of exceeding whole body dose vs distance apprears to be driven by the S2 source term. If this is the case, please describe all accident sequences (internal and external events) that contribute to the frequency of the S2 source term given in Table 4-2, pg. 4-7. In particular, define how the timing and size of containment leakage was determined for each of these classes of accident sequences. Justify the appropriateness of the bounding of each of the accidents into this particular source term.

RESPONSE 15

Our response to the overall question of how the contributors have changed from the SSPSA to recent updates, is comprised of the response to question 23 along with the following additional information:

ASSIGNMENT OF SEQUENCES TO RELEASE CATEGORY S2

As discussed in Section 11.6.3 (starting on page 11.6-5) of the SSPSA (PLG-0300), the release categories S2, $\overline{S2}$ and $\overline{S2V}$ were originally defined in the SSPSA to cover a class of accidents not modeled in the Reactor Safety Study (WASH-1400). Of these, the category $\overline{S2V}$ was found to be a significant risk contributor and was found to dominate the 200 rem dose vs distance curve in the EPZ sensitivity study (PLG-0465). Note that for simplicity, the release category notataion was simplified in the sensitivity study. Hence S2 on the sensitivity study is actually the same as $\overline{S2V}$ in the SSPSA.

The $\overline{S2V}$ release category was first defined to bound the releases that could occur as a result of small penetration failure during hydrogen burn pressure spikes that would occur shortly after the time of reactor vessel melt through (see discussion on page 11.6-12 of PLG-0300). Then to avoid an excessive number of release categories (to 14), additional sequences were conservatively assigned to $\overline{S2V}$ (now S2) as well as including those in the 3FP and 7FP plant damage states (as well as some steam generator tube rupture sequences). As noted in the response to question 23, it turned out that the 3FP and 7FP sequences fully dominated $\overline{S2V}$. The 3FP and 7FP sequences are dominated by station blackout, RCP seal LOCA sequences in which the release path is the 3 inch seal return line with failed open motor operated valves. There are many different sequences in but they all have the same release path. As noted above, the dominant sequences are initiated by seismic events partly because no credit was taken for operator recovery (closure of the outboard MOV manually) after seismic events. As discussed in Section 11.6.6.4 the assignment of seal return path sequences to $\overline{S2V}$ is conservative becuase the actual leak rates would be much less than calculated for this category.

RAI 16

Provide justification for the liner yield stress increase from the specified yield stress of 32 ksi to a mean yield stress of 45.4 ksi.

RESPONSE 16

Attached are liner plate certified materials test reports (CMTR) representing ten distinct "heats" or "charges" of liner plate material. These samples were selected at random. The average yield stress in this ten charge samples exceeds the mean yield stress of 45.4 ksi which was utilized in the analysis.

FILE NO. : 3P071-36/4D(A)
CONTRACT NO. : 2139-20089
SUPPLIER : YAMAMOTO

KAWASAKI STEEL CORPORATION MIZUSHIMA WORKS

INSPECTION CERTIFICATE

MIZUSHIMA KAWASAKI DORI
KURASHIKI JAPAN

NAME OF CLIENT : I.H.I. YOKOHAMA NO.3 WORKS.

PROJECT : PITTSBURGH-DES MOINES STEEL CO.
SEABROOK UNIT 1 & 2

WORK NO : 5501-382
SPEC.NO. : 3828500001

MATERIAL : ASME SA516GR.60HT

DATE OF ISSUE : MARCH 25, 1975
DATE OF TEST : MARCH 25, 1975

CHARGE NO.	ROLL NO.	DIMENSIONS (mm)	WEIGHT (kg)	TENSILE TEST (GL-8") Y. P. (KSI)	T. S. (KSI)	El. (%)	REDUC. OF AREA (%)	HARDNESS TEST ( )	BEND 180° R-0.5 t	CHARPY IMPACT TEST Notch : mmV Ave. Min. At ℃ ( ) 1	2	3	Ave.	REMARKS
				Min 32.0	60.0 -72.0	Min 21	Min.	Max.						
2-1110	2J046A	9.60 x 3,070 x 11,205 (SB-C2-9-1)	2,592	46.2	64.6	31			GOOD					

DOCUMENT REVIEWED
4 77

CHEMICAL ANALYSIS (%)

CHARGE NO.		C × 100	Si × 100	Mn × 100	P × 1000	S × 1000	Cu × 100	Ni × 100	Cr × 100	Mo × 1000	V × 1000			Ceq × 100
		Max. 21	15 -30	85 -120	Max. 35	Max. 40	Max.							
2-1110		10	24	111	14	7								

HEAT TREATMENT :

Plate :
Normalized...1652°F for 30 minutes

We hereby certify that the material described herein has ... of the contract.

MIZUSHIMA KAWASAKI DORI
KURASHIKI JAPAN

KAWASAKI STEEL CORPORATION MIZUSHIMA WORKS

INSPECTION CERTIFICATE

FILE NO. : [illegible]
CONTRACT NO. : 2139-20069
SUPPLIER : YAMAMOTO

NAME OF CLIENT : I.H.I. YOKOHAMA NO3 WORKS.

PROJECT : PITTSBURGH-DES MOINES STEEL CO.
[illegible] UNIT 1 & 2
WORK NO. : 5501-[illegible]
SPEC. NO. : 38-[illegible]0001

MATERIAL : ASME SA516GR.60MT

DATE OF ISSUE : MARCH 25, 1975
DATE OF TEST : MARCH 25, 1975

CHARGE NO.	ROLL NO.	DIMENSIONS (mm)	WEIGHT (kg)	TENSILE TEST GL 8" Y. P. (KSI) Min 32.0	T. S. ( ) 60.0 -72.0	El (%) Min 21	REDUC. OF AREA (%) Min	HARDNESS TEST ( ) Max.	BEND 180° R= 0.5 t	CHARPY IMPACT TEST Notch: mmV Ave. Min. At ℃ ( ) 1	2	3	Ave.	REMARKS
2-1784	2B249A	9.60 x 3,070 x 11,205 (SB-C2-9-1)	2,592	46.9	65.7	31			GOOD					

DOCUMENT REVIEWED
[illegible] 4 '77 J.S.V.
By [illegible]

CHEMICAL ANALYSIS (%)

CHARGE NO.		C ×100	Si ×100	Mn ×100	P ×1000	S ×1000	Cu ×100	Ni ×100	Cr ×100	Mo ×1000	V ×1000			Ceq ×100
		Max. 21	15 -30	85 -120	Max. 35	Max. 40	Max.							
2-1784		10	22	108	12	8								

HEAT TREATMENT :

Plate:
Normalized.......1652°F for 20 minut[illegible]

[illegible signature] INTECO

[illegible handwriting]

We hereby certify that the material described herein has been made in accordance with the rules of the [illegible] ASME SA516GR.60.

T. E[illegible]
CHIEF INSPECTOR

MIZUSHIMA KAWASAKI DORI
KURASHIKI JAPAN

KAWA[illegible]KI STEEL CORPORATION MIZUSHIMA [illegible]RKS

INSPECTION CERTIFICATE

FILE NO. : JP010-4,5/60(A)
CONTRACT NO. : 8139-33002
SUPPLIER : YAMA[illegible]IO

NAME OF CLIENT : I.H.I. YOKOHAMA NO.3 WORKS.
[illegible]ROJECT : PITT[illegible]-DES MOINES STEEL CO.
SEA[illegible] UNIT 1 & 2
WORK NO. : 5501-3[illegible]2
[illegible].NO. : 382-[illegible]001

MATERIAL : ASME SA516GR.60NT

DATE OF ISSUE : FEBRUARY 7, 1975
DATE OF TEST : FEBRUARY 7, 1975

CHARGE NO.	ROLL NO.	DIMENSIONS (mm)	WEIGHT (kg)	TENSILE TEST (GL- 8" [illegible]) Y. P. (KSI) Min. 32.0	T. S. (KSI) 60.0 -72.0	El. (%) Min. 21	REDUC. OF AREA (%) Min.	HARDNESS TEST ( ) Max.	BEND 180° R- 0.5 t	CHARPY IMPACT TEST Notch : mmV Ave. Min. At °C ( ) 1	2	3	Ave.	REMARKS
1-3830	1C102A	9.60 x 3,070 x 11,205 (SB-C2-9-1)	2,592	47.6	69.1	24			GOOD					
	B	9.60 x 3,070 x 11,205 (SB-C2-9-1)	2,592											

DOCUMENT REVIEWED [illegible] 5 '77 I.S.W

CHARGE NO.		C × 100	Si × 100	Mn × 100	P × 1000	S × 1000	Cu × 100	Ni × 100	Cr × 100	Mo × 1000	V × 1000			Ceq × 100
		Max. 21	15 -30	85 -120	Max. 35	Max. 40	Max.							
1-3830		12	23	109	15	7								

CHEMICAL ANALYSIS (%)

HEAT TREATMENT :

Plate :

Normalized...1652°F for 20 minutes.

MIZUSHIMA KAWASAKI DORI
KURASHIKI JAPAN

KAWASAKI STEEL CORPORATION MIZUSHIMA WORKS

INSPECTION CERTIFICATE

FILE NO. : JP068-1/18(A)
CONTRACT NO. : 2139-20080
SUPPLIER : YAMAMOTO

NAME OF CLIENT : I .I. [illegible] NO.3 [illegible]
[illegible] : [illegible] STEEL CO.
[illegible] 1 & 2
[illegible] : [illegible]
[illegible] : [illegible] 1

MATERIAL : ASME SA516GR.60NT

DATE OF ISSUE : FEBRUARY 27, 19
DATE OF TEST : FEBRUARY 27, 19

CHARGE NO.	ROLL NO.	DIMENSIONS (mm)	WEIGHT (kg)	TENSILE TEST GL- 8" Y. P. KSI Min 32.0	T. S. 60.0 -72.0	EL (%) Min 21	REDUC. OF AREA (%) Min	HARDNESS TEST Max	BEND 180° R- 0.5	CHARPY IMPACT TEST 1	2	3	Ave.	REMARKS
1-[illegible]	1.1[illegible]	9.[illegible] × 3,070 × 11,2[illegible]5 (.[illegible]-C[illegible]-1)	2,5[illegible]2	47.6	67.0	29			GOOD					

CHEMICAL ANALYSIS (%)

CHARGE NO.	C ×100	Si ×100	Mn ×100	P ×1000	S ×1000	Cu ×100	Ni ×100	Cr ×100	Mo ×1000	V ×1000			Ceq ×100
	Max. 21	15 -[illegible]	65 -1[illegible]	Max. 35	Max. [illegible]	Max.							
[illegible]	11	24	113	15	8								

HEAT TREATMENT
Plate:
Normalized...1652°F for 20 minutes

DOCUMENT REVIEW
[illegible] 4 77 I.S.V.

We hereby certify that the material described herein has been made in accordance with the rules of the ~~xxxxxxx~~ ASME SA516GR.60.

CHIEF INSPECTOR

FILE NO. : JF071-22/48(A)
CONTRACT NO. : 2139-20089
SUPPLIER : YAMAMOTO

MIZUSHIMA KAWASAKI DORI
KURASHIKI JAPAN

KAWASAKI STEEL CORPORATION MIZUSHIMA WORKS

INSPECTION CERTIFICATE

NAME OF CLIENT : I.H.I. YOKOHAMA NO.3 WORKS.

PROJECT : PITTSBURGH-DES MOINES STEEL CO.
SEABROOK UNIT 1 & 2
WORK NO. : 5501-382
SPEC.NO. : 382E500001

MATERIAL : ASME SA516GR.60NT

DATE OF ISSUE : MARCH 25, 1975
DATE OF TEST : MARCH 25, 1975

CHARGE NO.	ROLL NO.	DIMENSIONS (mm)	WEIGHT (kg)	TENSILE TEST (GL- 8") Y. P. (KSI) Min. 32.0	T. S. ( ) 60.0 -72.0	El. (%) Min. 21	REDUC. OF AREA (%) Min.	HARDNESS TEST ( ) Max.	BEND 180° R- 0.5t	CHARPY IMPACT TEST Notch : mmV Ave. Min. At ℃ ( ) 1	2	3	Ave.	REMARKS
1-3551	2D291A	9.60 x 3,070 x 11,205 (SB-C2-9-1)	2,592	46.5	68.6	32			GOOD					

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CHEMICAL ANALYSIS (%)

CHARGE NO.		C ×100	Si ×100	Mn ×100	P ×1000	S ×1000	Cu ×100	Ni ×100	Cr ×100	Mo ×1000	V ×1000			Ceq ×100
		Max. 21	15 -30	85 -120	Max. 35	Max. 40	Max.							
1-3551		11	23	106	14	7								

HEAT TREATMENT :

Plate :
Normalized...1652°F for 20 minutes

IHI QC
M. Maeda

We hereby certify that the material described herein has been made in accordance with the rules of the contract. ASME SA516GR.60N.

T. Ekoro
CHIEF INSPECTOR

0373

MIZUSHIMA KAWASAKI DORI
KURASHIKI JAPAN

KAWASAKI STEEL CORPORATION MIZUSHIMA WORKS

INSPECTION CERTIFICATE

FILE NO. : [illegible]
CONTRACT NO. : 2139-30009
SUPPLIER : YAMAMOTO

NAME OF CLIENT : I.H.I. YOKOHAMA NO.3 WORKS.

PROJECT : PITTSBURGH-DES MOINES STEEL CO.
SEABROOK UNIT 1 & 2
WORK NO. : 5501-383
[illegible] NO. : [illegible]0001

MATERIAL : ASME SA516GR.60NT

DATE OF ISSUE : MARCH 25, 1975
DATE OF TEST : MARCH 25, 1975

CHARGE NO.	ROLL NO.	DIMENSIONS (mm)	WEIGHT (kg)	TENSILE TEST GL- 8" Y. P. (KSI) Min 32.0	T. S. (KSI) 60.0 -72.0	El. (%) Min 21	REDUC. OF AREA (%) Min.	HARDNESS TEST ( ) Max.	BEND 180° R- 0.5 t	CHARPY IMPACT TEST Notch : mmV Ave. Min. At ℃ ( ) 1	2	3	Ave.	REMARK
1-3601	29288A	9.60 x 3,070 x 11,205 (SB-C2-9-1)	2,592	47.9	66.8	32			GOOD					

Austenite Grain Size.....No.8(Min.No.5)
Ultrasonic Test..........Acceptable
Dimensions Inspection....Acceptable
Surface Inspection.......Acceptable

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[illegible] 4'77 [illegible]

CHEMICAL ANALYSIS (%)

CHARGE NO.		C × 100	Si × 100	Mn × 100	P × 1000	S × 1000	Cu × 100	Ni × 100	Cr × 100	Mo × 1000	V × 1000	Sol.Al ×1000		Ceq × 100
		Max. 21	15 -30	85 -120	Max. 35	Max. 40	Max.					Max. 40		Max. 34
1-3601		10	23	111	18	8						22		30

HEAT TREATMENT :

Plate:
Normalized......1652°F for 20 minut

I.H.I. QC
M. Nishiki

We hereby certify that the material described herein has been made in accordance with the rules of the [illegible]
ASME SECTION III NC-2000.

T. Shono
CHIEF INSPECTOR

MIZUSHIMA KAWASAKI DORI
KURASHIKI JAPAN

KAWASAKI STEEL CORPORATION MIZUSHIMA WORKS

INSPECTION CERTIFICATE

FILE NO. : JP071-28,29/40(A)
CONTRACT NO. : 2139-20089
SUPPLIER : YAMAMOTO

NAME OF CLIENT : I.H.I. YOKOHAMA NO.3 WORKS.

PROJECT : PITTSBURGH-DES MOINES STEEL CO.
SEABROOK UNIT 1 & 2
WORK NO.: 5501-382
SPEC.NO.: 383E500001

MATERIAL : ASME SA516GR.60MT

DATE OF ISSUE : MARCH 25, 1975
DATE OF TEST : MARCH 25, 1975

CHARGE NO.	ROLL NO.	DIMENSIONS (mm)	WEIGHT (kg)	TENSILE TEST GL-8" Y. P. KSI Min. 32.0	T. S. ( ) 60.0 -72.0	El. (%) Min. 21	REDUC. OF AREA (%) Min.	HARDNESS TEST ( ) Max.	BEND 180° R=0.5 t	CHARPY IMPACT TEST Notch: mmV Ave. Min. At °C ( ) 1	2	3	Ave.	REMARKS
1-3706	2F885A	9.60 x 3,070 x 11,205 (SB-C2-9-1)	2,592	46.9	67.1	31			GOOD					
	B	9.60 x 3,070 x 11,205 (SB-C2-9-1)	2,592											

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By 83 4 77 I.S.V.

CHEMICAL ANALYSIS (%)

CHARGE NO.		C ×100	Si ×100	Mn ×100	P ×1000	S ×1000	Cu ×100	Ni ×100	Cr ×100	Mo ×1000	V ×1000			Ceq ×100
		Max. 21	15 -30	85 -120	Max. 35	Max. 40	Max.							
1-3706		11	23	108	15	8								

HEAT TREATMENT :

Plate :
Normalized...1652°F for 20 minutes.

141.@C
M. Miyoshi

We hereby certify that the material described herein has been made in accordance with the rules of the ~~contract~~ ASME SA516GR.60.

T. Shoue
CHIEF INSPECTOR

0380

FILE NO. : 3F071-23/48K(A)

CONTRACT NO. : 2139-20089

SUPPLIER : YAMAMOTO

MIZUSHIMA KAWASAKI DORI
KURASHIKI JAPAN

KAW[illegible]AKI STEEL CORPORATION MIZUSHIMA [illegible]RKS

INSPECTION CERTIFICATE

NAME OF CLIENT : I.H.I. YOKOHAMA NO.3 WORKS.

PROJECT : PITTSBURGH-DES MOINES STEEL CO.
SEABROOK UNIT 1 & 2

WORK NO. : 5501-382

SPEC. NO. : 382E500001

MATERIAL : ASME SA516GR.60MT

DATE OF ISSUE : MARCH 25, 1975

DATE OF TEST : MARCH 25, 1975

CHARGE NO.	ROLL NO.	DIMENSIONS (mm)	WEIGHT (kg)	TENSILE TEST (GL- 8") Y. P. (KSI) Min. 32.0	T. S. (KSI) 60.0 -72.0	El. (%) Min. 21	REDUC. OF AREA (%) Min.	HARDNESS TEST ( ) Max.	BEND 180° R- 0.5 t	CHARPY IMPACT TEST Notch : mmV Ave. Min. At ℃ ( ) 1	2	3	Ave.	REMARKS
3-5372	2BA05A	9.60 x 3,070 x 11,205 (SB-C2-9-1)	2,592	46.5	65.3	32			GOOD					

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U.E.&C.

CHEMICAL ANALYSIS (%)

CHARGE NO.		C × 100	Si × 100	Mn × 100	P × 1000	S × 1000	Cu × 100	Ni × 100	Cr × 100	Mo × 1000	V × 1000			Ceq × 100
		Max. 21	15 -30	85 -120	Max. 35	Max. 40	Max.							
3-5372		11	23	110	11	6								

HEAT TREATMENT :

Plate :

Normalized...1652°F for 20 minutes.

M. Iai

KAWATECO

14 I 81

M. Nishi

We hereby certify that the material described herein has been made in accordance with the rules of the ASME SA516GR.60.

T. Ohmae

CHIEF INSPECTOR

FILE NO. JP071-11.12/48(B)

CONTRACT NO. 2139-20009

SUPPLIER YAMAMOTO

MIZUSHIMA KAWASAKI DORI
KURASHIKI JAPAN

KAW KI STEEL CORPORATION MIZUSHIMA RKS

INSPECTION CERTIFICATE

NAME OF CLIENT : I.H.I. YOKOHAMA NO.3 WORKS.

PROJECT : PITTSBURGH-DES MOINES STEEL CO. SEABROOK UNIT 1 & 2

MATERIAL : ASME SA516GR.60MT

WORK NO. : 5501-382

SPEC.NO. : 382E500001

DATE OF ISSUE : MARCH 25, 1975

DATE OF TEST : MARCH 25, 1975

CHARGE NO.	ROLL NO.	DIMENSIONS (mm)	WEIGHT (kg)	TENSILE TEST GL-8" Y. P. KSI Min. 32.0	T. S. 60.0-72.0	El. (%) Min. 21	REDUC. OF AREA (%) Min.	HARDNESS TEST ( ) Max.	BEND 180° R-0.5 t	CHARPY IMPACT TEST Notch : mmV Ave. Min. At ℃ 1	2	3	Ave.	REMARK
3-6323	2C358A	9.60 x 3,070 x 11,205 (SB-C2-9-1)	2,592	48.5	66.4	32			GOOD					
	B	9.60 x 3,070 x 11,205 (SB-C2-9-1)	2,592											DOCUMENT REVIEW

Austenite Grain Size..........No.8(Min.No.5)
Ultrasonic Test...............Acceptable
Dimensions Inspection.........Acceptable
Surface Inspection............Acceptable

CHEMICAL ANALYSIS (%)

CHARGE NO.		C × 100	Si × 100	Mn × 100	P × 1000	S × 1000	Cu × 100	Ni × 100	Cr × 100	Mo × 1000	V × 1000	Sol.Al x1000	Ceq × 100
		Max. 21	15-30	85-130	Max. 35	Max. 40	Max.					Max. 40	Max. 34
3-6323		11	24	110	17	8						27	31

HEAT TREATMENT :

Plate:
Normalized.......1652°F for 20 min.

141.Q1

M. Nishi

We hereby certify that the material described herein has been made in accordance with the rules of the ~~contract~~ ASME SECTION III CC-2000.

T. Otsuo

CHIEF INSPECTOR

MIZUSHIMA KAWASAKI DORI
KURASHIKI JAPAN

KAWASAKI STEEL CORPORATION MIZUSHIMA WORKS

INSPECTION CERTIFICATE

FILE NO.: [illegible]
CONTRACT NO.: [illegible]-17300
SUPPLIER: [illegible]

NAME OF CLIENT: I.H.I. YOKOHAMA NO.3 WORKS.

[illegible] STEEL CO.
[illegible] 1 & 2
[illegible]
[illegible]

MATERIAL: ASME SA516GR.60NF

DATE OF ISSUE: JANUARY 30, 19[illegible]

DATE OF TEST: JANUARY 30, 19[illegible]

CHARGE NO.	ROLL NO.	DIMENSIONS (mm)	WEIGHT (kg)	TENSILE TEST Y.P. (KSI) Min 32.0	T.S. ( ) 60.0-72.0	El. (%) Min 21	REDUC. OF AREA (%) Min.	HARDNESS TEST ( ) Max.	BEND 180° R=0.5 t	CHARPY IMPACT TEST 1	2	3	Ave.	REMARK
[illegible]	173[illegible]	9.60 x 3,070 x 11,205 (SB-C3-9-1)	2,592	46.2	66.6	30			GOOD					
	"	9.60 x 3,070 x 11,205 (SB-C3-9-1)	2,592											

Austenite Grain Size..................No.8(Min.No.5)
Ultrasonic Test......................Acceptable
Dimensions Inspection................Acceptable
Surface Inspection...................Acceptable

DOCUMENT REVIEWED

CHEMICAL ANALYSIS (%)

CHARGE NO.		C ×100	Si ×100	Mn ×100	P ×1000	S ×1000	Cu ×100	Ni ×100	Cr ×100	Mo ×1000	V ×1000	Sol.Al ×1000		Ceq ×100
		Max. 21	15-30	85-120	Max. 35	Max. [illegible]	Max.					Max. 40		Max. 34
[illegible]		10	22	104	13	8						26		29

HEAT TREATMENT:

Plate:

Normalized.........1652°F 58' 20min

We hereby certify that the material described herein has been made in accordance with the rules of the ASME SECTION III CC-[illegible].

CHIEF INSPECTOR

MIZUSHIMA KAWASAKI DORI
KURASHIKI JAPAN

KAWASAKI STEEL CORPORATION MIZUSHIMA WORKS

INSPECTION CERTIFICATE

FILE NO.: [illegible]-11,12/50
CONTRACT NO.: 8139-17500
SUPPLIER: YAMAMOTO

NAME OF CLIENT: I.H.I. YOKOHAMA NO. 3 WORKS.
PROJECT: [illegible]-DES MOINES STEEL CO. [illegible] UNIT 1 & 2
WORK NO.: 5301-362
SPEC. NO.: 362.500001

MATERIAL: ASME SA516GR.60MT

DATE OF ISSUE: JANUARY 30, 1975
DATE OF TEST: JANUARY 30, 1975

CHARGE NO.	ROLL NO.	DIMENSIONS (mm)	WEIGHT (kg)	TENSILE TEST (GL- 8") Y.P. (KSI) Min 32.0	T.S. 60.0-72.0	El. (%) Min 21	REDUC. OF AREA (%) Min.	HARDNESS TEST ( ) Max.	BEND 180° R-0.5 t	CHARPY IMPACT TEST Notch: mmV Ave. Min. At ( ) 1	2	3	Ave.	REMARK
2-0897	2Y441A	9.60 x 3,070 x 11,205 (SB-C2-9-1)	2,592	45.8	66.0	25			GOOD					
	B	9.60 x 3,070 x 11,205 (SB-C2-9-1)	2,592											

Austenite Grain Size....No.8(Min.No.5)
Ultrasonic Test.........Acceptable
Dimensions Inspection...Acceptable
Surface Inspection......Acceptable

DOCUMENT REVIEWED BY 4/77 I.S.I.

CHEMICAL ANALYSIS (%)

CHARGE NO.		C ×100	Si ×100	Mn ×100	P ×1000	S ×1000	Cu ×100	Ni ×100	Cr ×100	Mo ×1000	V ×1000	Sol.Al ×1000		Ceq ×100
		Max. 21	15-39	85-120	Max. 35	Max. 40	Max.					Max. 40		Max. 34
2-0897		11	23	110	19	7						26		31

HEAT TREATMENT:
Plate :
Normalized.....1650°F for 20 min

We hereby certify that the material described herein has been made in accordance with the rules of the ~~[illegible]~~ ASME SECTION III CC-2000.

[signature]

[signature]
CHIEF INSPECTOR

RAI 17

Indicate the correlation between containment failure sequences and the containment failure modes.

RESPONSE 17

Three terms are used to describe the failure behaviour of the Seabrook containment. These are:

- o Containment Failure Mode: The term containment failure mode is used to describe the structural member of the containment system which fails as a result of an overpressure condition in the containment. For example wall hoop failure or feedwater penetration flued head failure are containment failure modes. Table 11.3-1 of the SSPSA (PLG-0300) lists the different containment failure modes.

- o Containment Failure Type: The term containment failure type is used to discretize the range of leak areas which are predicted for the different containment failures modes. Three containment failure types have been defined, namely Types A, B and C as defined in Section 11.3.1 of the SSPSA.

- o Containment Failure Sequence: The term containment failure sequence is used to describe the containment failure associated with each release category. For example release category S3 is associated with late containment overpressure failure sequences. In the SSPSA two additional distinctions were made to characterize a release category, namely whether the containment spray system worked and whether the containment floor and reactor cavity were dry (vaporization release) or wet. The release categories are described in Section 11.6.4 and in Tables 11.6-1 and 2. Since all risk dominant accident sequences involved dry containment conditions without spray, the latter two distrinctions are dropped in the RMEPS study and in the WASH-1400 sensitivity study. All release categories were than based on dry containment conditions without spray and the containment failure sequence was the only remaining distinction for the release categories.

The correlation between containment failure modes and containment failure type is given in Table 11.3-1 of the SSPSA. One containment failure type is assigned to each containment failure mode. The correlation between the containment failure sequences (or release categories in the RMEPS and WASH-1400 Sensitivity Study) and the containment failure mode is described below.

- o Release category S1 model an early gross containment failure sequence. It is equivalent to PWR-1 in WASH-1400. It is always treated as a gross (Type C) containment failure.

- o Release category S2 models containment failure sequences with an early increased leak rate. It models a Type A containment failure occuring at the time of vesselbreach. From a containment failure pressure point of view, the S2 release category is more conservative continues until a late overpressure failure mode occurs as in release category S3.

- Release category S3 represents a late overpressure failure sequence. It is modeled as a linearly increasing leak rate beginning at the time when the first type A failure mode is predicted to occur and building up to a full type A failure mode (6 square inches) at the time when a type B or C failure mode is predicted. This is followed by a type B or C failure type at that time. However, because all S3 accident sequences had calculated containment failure times longer than 24 hours they were always treated as Type C failures. The lowest containment failure time estimated in the uncertainty analysis was 22 hours and the type C failure assumption was retained even for this case.

- Release category S4 represesnets basemat melt-through failure sequences. Because of the uncertainties in the release fractions for basemat melt-through at a rock foundation site, all basemat melt-through failures were conservatively modeled as an S3 release category.

- Release category S5 represents accident sequences where the containment remains intact. A continuous release at a rate of 0.1 v/o per day was conservatively simulated as an equivalent instantaneous release at the time of vessel breach.

- Release category S6 represents accident sequences where the containment is ot isolated from the beginning. It is modeled as the largest containment penetration which is allowed by technical specification to be temporarily open during normal operation and which communicates directly between the containment atmosphere and the environment. This is the 8 inch diameter online purge penetration with a flow area of 50 square inches. Thus in size it corresponds to a pre-existing Type B containment failure.

- Release category S7 represents containment bypass accident sequences. It is modled as an RHR pump seal failure with a combined leak area for both RHR pumps of 2.6 square inches. The frequency of the traditional V-sequence failure mode (RHR pipe rupture) is included in release category S1.

In summary, it is noted that the analysis did not depend on the distinction between a type B or a type C overpressure containment failure, except that the inclusion of type B failures shifted the containment failure probability distribution to lower pressures compared to the case where the type B failures are not considered.

RAI 18

Provide the basis for concluding that the sight glasses in the hatches will not fail under high containment temperature and pressure conditions.

ANSWER 18

The sight glass in the personnel hatch was tested by its supplier, Owen Corning Co., under the following conditions:

Pressure = 150 psig
Temperature = 550°F

In addition the pressure was cycled from 0 psig to 150 psig ten times at a constant temperature of 550°F.

The Owens Corning data sheet is attached.

We are currently persuing discussion with Corning Glass to determine if any testing has been done above these valves.

RAI 19

Document the effect that the recent update in seismic fragilities will have on the conclusions of the PSA results.

RESPONSE 19

Seismic sequences dominate release categories S2 and S6 in the Risk Management and Emergency Planning Study (RMEPS). The response to question #23 discusses the principal contribution to early health risk and explains how these release categories contribute to early health risk. The following explains how the seismic fragility update is expected to change the frequency of S2 and S6; it is expected that the effect on the frequency of all other release categories will be insignificant. A complete requantification will be included in the probabilistic safety assesment (PSA) update now in progress and planned for completion in 1987.

In the complete seismic risk analysis, a point estimate analysis is first performed using the plant event trees that are quantified for several discrete values of ground acceleration. From the point estimate results, dominant sequences initiated by seismic events are identified; then, these sequences are reanalyzed using a computer code called SEIS4. This code is described in the SSPSA Section 4 and 9. In SEIS4, the seismicity curves and fragility curves are appropriately combined and uncertainties in these curves are propagated to obtain uncertainty distributions on the final result, which is either a core melt or plant damage state frequency contribution. In the following approximate analysis, the point estimate step is bypassed, so some assumptions are made about dominant sequences. Hence, these results are only rough approximations and should only be used for order-of-magnitude estimates. Again, a complete reanalysis of seismic events is currently in progress and is planned for completion in 1987.

1.1 RELEASE CATEGORY S2

This release category is dominated by earthquake and transient initiating events. These sequences can be simply represented as

$$OG(DT + DG + SSPS) \qquad (1)$$

where

- OG = Offsite Power Fragility
- DT = Diesel Generator Day Tank Fragility
- DG = Diesel Generator Fragility
- SSPS = Solid State Protection System (SSPS) Fragility (actually 120V AC power panel required for SSPS success)

and only seismic unavailabilities are included.

Also, earthquake and large loss of coolant accident (LOCA) initiating events provide a small contribution and can be represented as

$$LL*OG*(DT + DG + SSPS) \qquad (2)$$

where

LL = Large LOCA Fragility

Equation (1) was quantified with the SEIS4 computer code and resulted in the following annual core melt frequency:

Mean $= 2.84 \times 10^{-5}$

Variance $= 2.24 \times 10^{-9}$

Based on the fragility update, SSPS and DT can be dropped from the model, based on significantly higher capacities. However, a relay chatter fragility at a relatively lower capacity has been identified in the 4,160V switchgear. This chatter could have a negative effect; e.g., trip out the diesels. Until the consequences of this chatter are evaluated, it is assumed that the chatter fails both diesels. Therefore, Equation (1) can be changed as follows:

OG*(chatter + DG) (3)

where

Chatter = Relay Chatter Fragility (4,160V switchgear)

Quantifying equations (3) for annual core melt frequency with SEIS4 results in

Mean $= 1.8 \times 10^{-5}$

Variance $= 9.58 \times 10^{-10}$

Comparing the quantification of Equations (1) and (3) shows a slight reduction (less than a factor of 2) in frequency. However, this assumes the chatter fails the diesels without recovery. An ongoing relay chatter review will determine whether this particular chatter is a real concern. In addition, this review will determine whether there are any other relay chatters that should be considered in the model.

1.2 RELEASE CATEGORY S6

This release category is dominated by earthquake and transient initiating events. These sequences can be simply represented as

NOG*SSPS (4)

where

NOG = Offsite Power Available (negation of OG - fragility)

As described above under release category S2, the solid state protection system can be dropped from the model. Therefore, the simple model in Equation (4) would go to zero. To actually determine the new S6 frequency, the whole plant model needs to be requantified and unraveled to obtain new dominant sequences and frequencies. However, the trend is a reduced frequency unless the ongoing relay chatter review identifies new sequences.

RAI 20

Assess the impact on risk of assuming that the containment capability corresponds to the pressure which produces 1% strain in the containment wall.

RESPONSE 20

A sensitivity analysis on the risk results in the SSPSA, the RMEPS and the WASH-1400 Sensitivity Study was performed to determine whether any of the basic conclusions with respect to risk or emergency planning requirements would change substantially if the containment was postulated to fail at a deformation strain of 1 percent. It is emphasized that there is no basis in experimental data or analysis to support a 1 percent strain failure condition as being a reasonable definition of failure or that containment failure at this condition is any more likely than what is implied by Figure 11.3-14 in the SSPSA or Figure 4-7 in the RMEPS.

Since the failure definition is arbitrary, there is no meaning in assessing uncertainty in this failure condition and it has been postulated to occur with certainty when the containment pressure reaches that pressure value where the calculated deformation strain reaches 1 percent according to Figure 4-2 in Appendix H.1 of the SSPSA. This corresponds to a maximum radial displacement of the containment wall of 8.4 inches and it occurs at a pressure of 175 psig or 190 psia. This corresponds to the low temperature wet containment condition in Figure 11.3-14 of the SSPSA or Figure 4-7 in the RMEPS. Since in the analysis no distinction is made between type B and type C containment failures (see answer to question 17), the composite wet containment probability distribution (solid curve) is stepped from the calculated value of 0.2 at 190 psia to 1.0 (guaranteed failure). Note this implies that the containment failure analysis in the SSPSA predicted a 20 percent probability for containment failure at the pressure corresponding to the 1 percent deformation strain. For dry containment conditions (dotted curve), the elevated temperature conditions reduce the strength of the innermost steel layers and the 1 percent deformation strains are reached at a pressure of 158 psig or 173 psia. An containment failures which in the original containment failure analysis were predicted at pressures below these 1 percent strain pressures were retained without modifications. The modified containment failure distributions of the 1 percent strain failure sensitivity analysis are shown in the attached Figure 20-1.

Next, the impact of this change on the calculated risk and on the conclusions with respect to emergency planning were examined. An impact would result in one of two ways. Containment event tree split fractions would change where these were determined from the containment failure pressure probability distribution. Secondly, the timing and release magnitude of late overpressure failure release categories (S2 and S3) would change due to earlier containment failure times. However, changes would only result in cases which depended on containment failure pressures above 173 psia for dry cases.

The impact on release categories was examined first. New probability distributions for the time of containment failure were generated for dry containment sequences. These are shown in Figure 20-2. These curves do not exhibit a steep change because the uncertainty in the containment pressure versus time relationship calculated by the MARCH and COCOCLASS9 codes is still a valid consideration. The net effect of these changes is to reduce the time of release for release category S3B in the RMEPS from 89 hours to 70 hours and to reduce the release duration of release category S2B-3 from 56 hours to 37 hours. No changes result in either the conservative source terms in RMEPS or in the source terms for the WASH-1400 Sensitivity Study, because for all these source terms, the release timing was assessed as occurring before the time when the pressure in the containment reaches the 1 percent strain level. The release fraction factor for particulate radionuclides is shown in Figure 11.6-3 of the SSPSA. It is shown that no significant change occurs between 70 and 90 hours. Therefore, there would be no change in the release fractions as a result of the above noted change in release timing. Furthermore, in both cases (S3B and S2B-3), the warning times are still much longer than required for complete evacuation. It is thus concluded that no change whatsoever can be identified in any of the consequences calculated either in the SSPSA or in RMEPS or in the WASH-1400 sensitivity study as a result of postulating containment failure at a 1 percent deformation strain.

Lastly, the impact on containment split fractions was examined in Section 11.7 of the SSPSA. In no case is a split fraction dependent on a containment failure pressure in excess of the 1 percent strain value for dry condition (173 psia). The split fractions for two top events in the containment event tree are affected. The split fraction for top event 10B on Table 11.7-8 in the SSPSA shifts slightly to increase the probability of late overpressure failure and correspondingly decrease the probability of basemat melt through. However, as explained in the answer to question 17, all basemat melt through cases are conservatively treated as late overpressure failures and this change therefore has no impact on the results. Secondly, the split fractions for top event 12A in Table 11.7-8 in the SSPSA would shift significantly to increase the probability of type C containment failure and decrease the probability of type B containment failure. However, since in the analysis all late over pressure failures were treated as type C failures, this change also has no effect on any of the results.

Overall, it is concluded that the assumption of containment failure at a pressure corresponding to 1 percent deformation strain has no discernible effect on any of the results and conclusions documented in either the SSPSA or the RMEPS or in the WASH-1400 Sensitivity Study. This conclusion can be traced to three distinct reasons: (1) no early pressure transients reach a magnitude of 190 psia, (2) the reduction in release timing for late overpressure is insignificant with respect to warning times and release fractions, and (3) conservative analysis assumptions in the containment event tree quantification absorb any effect which would otherwise be visible in the release category frequencies.

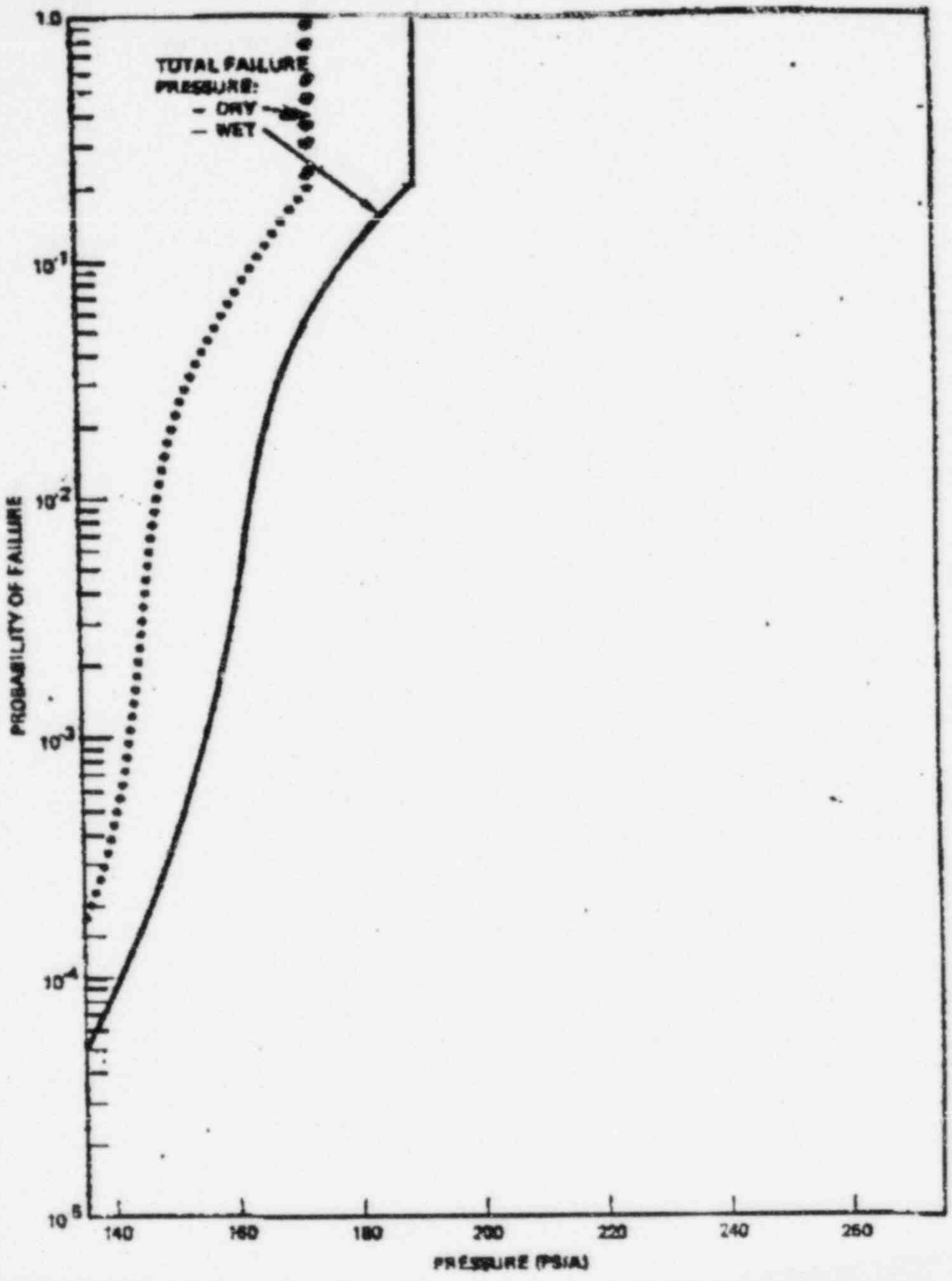


FIGURE 20-1. COMPOSITE CONTAINMENT FAILURE PROBABILITY DISTRIBUTIONS FOR THE 1% STRAIN CONTAINMENT FAILURE SENSITIVITY CASE

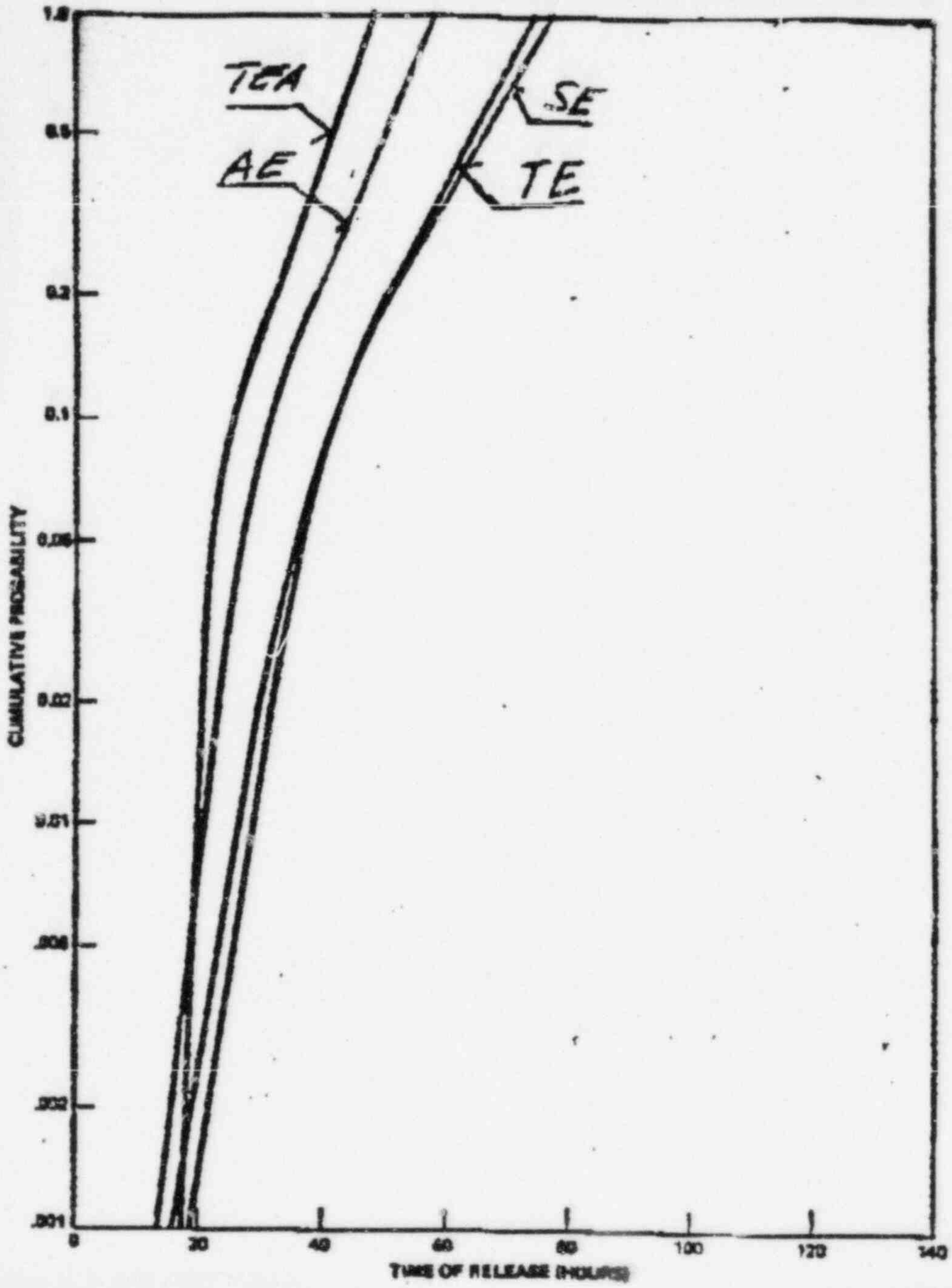


FIGURE 20-2. CUMULATIVE PROBABILITY DISTRIBUTIONS FOR LATE OVERPRESSURE FAILURE TIME IN DRY SEQUENCES - 1% STRAIN CONTAINMENT FAILURE SENSITIVITY CASE

RAI 21

What is the impact on risk from accidents during shutdown and refueling when the containment function may not be available?

RESPONSE 21

The purpose of this technical note is to address the risk from accident sequences that could potentially initiate during plant shutdown at Seabrook Station. Specifically, this note is intended to answer a question posed by the NRC staff during their review of the Risk Management and Emergency Planning Study (Reference 1), the companion sensitivity syudy (Reference 2), and Question Number 21 (Reference 3).

All work performed to date to identify and to assess the risk of potential accidents at Seabrook station has concerned itself primarily with scenarios that could initiate at or near full power operation. In the original full scope PSA (Reference 4), the coverage of accident sequences in terms of initiating events, the possibilities for system success and failure states, and the treatment of dependent events met or exceeded those of other published PSAs. This coverage was certainly greater than was possible during the seventies when the Reactor Safety Study was performed. A judgment normally made in a PSA, and made in the SSPSA, is that the level of risk associated with accidents that could initiate during full power operation, however small, is substantially greater than that associated with accidents that occur during shutdown. There are many reasons to support this judgment including the fact that at full power there is a greater level of RCS stored energy, after-heat level and inventory of radionuclides than the case with plant shutdown. There is also generally more time available to recover from adverse situations during shutdown.

Several years after the SSPSA was completed a research project was performed for the Electric Power Research Institute in which the risk of accidents at the Zion nuclear plant during plant shutdown and RHR system operation was assessed (Reference 5). The only risk parameter quantified in this study was core melt frequency. The results in comparison with the results of the Zion plant PSA (Reference 6) for power operation events that had been completed previously by the same PSA team are as follows.

Description	Core Damage Frequency	
	Mean	Median
Cold Shutdown (Reference 5)	1.8×10^{-5}	2.6×10^{-6}
Power Operations (Reference 6)	6.7×10^{-5}	5.0×10^{-5}

Hence, the core melt frequency from cold shutdown events at Zion is less likely but more uncertain than that from power operations. The Zion cold shutdown study did not address consequences of these events; it only addressed the frequency of core damage events.

The Zion cold shutdown study examined plant shutdown and startup procedures in detail to identify a wide spectrum of potential accident sequences that could originate and develop during plant shutdown. It also made use of an in-depth review of in-plant records and information that covered 10 refueling outages, 24 maintenance outages, and some 27,888 hours of RHR system operations. Several person-years of effort went into the Zion investigation.

It is of interest in this note to address the risk from plant shutdown events at Seabrook Station, which like the Zion plant, is a four-loop Westinghouse PWR with a large dry containment. In the brief time available, it is not possible to complete the kind of in-depth examination that was described in Reference 5. On the other hand, for the purpose of addressing the implications on emergency planning, it is not sufficient to measure risk simply in terms of core damage frequency. With this perspective in mind, the objectives of this response are to:

- Provide an order of magnitude estimate of the frequency of core damage events that could initiate at Seabrook Station during plant shutdown.

- Estimate the frequency of the above events that result in containment bypass, containment high leakage, or containment intact end states.

- Account for important specific and unique features of the Seabrook plant hardware and procedures.

- Provide a suitable allowance for uncertainties associated with a preliminary level of analysis through the appropriate use of conservative assumptions.

- Provide for a reasonable level of accountability of operating experience with events that have occurred in similar plants during plant shutdown.

APPROACH

The approach taken to address shutdown loss of cooling events at Seabrook Station was first to review the Zion study (Reference 5), to compare the design and operational features of Zion and Seabrook, and to identify key differences important to the determination of shutdown cooling risk. Based on this review and the key differences that were identified, a determination was made of the extent to which all or part of the NSAC-84 results for Zion could be applied to Seabrook. In cases where Seabrook specific features indicate a reduced level of risk, appropriate corrections were made to the Zion results. Finally, a quantification was made of sequences that could occur at Seabrook Station at a higher frequency than that assessed for Zion. In summary, the risk of shutdown cooling events at Seabrook Station was evaluated as follows:

Seabrook Risk = Zion Risk per NSAC-84
- Portion of Zion Risk Not Applicable to Seabrook
+ Portion of Seabrook Risk Not Applicable to Zion

In other words, there are some design and operational features common to Zion and Seabrook and some unique to each plant. The enhanced features of Seabrook were accounted for by reducing the risk contribution of selected dominant sequences in the Zion results. This resulted in a reduction of the core damage frequency evaluated in NSAC-84. Then, the enhanced features of Zion were accounted for by adding to those results a separate Seabrook specific analysis of accident sequences that were not important in the Zion results because of its unique enhanced features.

The above process resulted in a balanced and unbiased albeit conservative assessment for Seabrook Station that was especially designed to make maximum and appropriate use of the Zion results for core damage frequency. Then, all the resultant core damage sequences were evaluated to determine the frequeny of three types of core damage release states: core damage with intact containment, sma;; bypass, and large bypass. Finally bounding estimates were made of the contributions of shutdown loss of cooling events to the 200-rem dose versus distance curves in References 1 and 2.

COMPARSION OF ZION AND SEABROOK DESIGN FEATURES

The ability to respond ti this question quickly facilitated by the fact that key plant and systems analysts of the Zion PSA team played major roles on the Seabrook PSA team. The design features of the respective plants were compared from two perspectives. First, the major differences between the two plants were noted based in our general understanding of the plants, systems, components, and PSA results. Second, the 34 dominant accident sequences for Zion shutdown cooling between these systems and the frontline systems, such as the RHR system. For one thing, there are ways to utilize equipment on Unit 2 for Unit 1 and vice versa at Zion that are not possible at Seabrook. These differences stem from the fact that modern design criteria, to which Seabrook was designed and Zion was not, call for a strict physical separation between redundant trains of safety-related systems and greatly reduce the opportunities for lining up cross-train pump and heat exchanger combination. In othere words, there are more success paths in the older plants such as Zion. Ironically, the introduction of these more restrictive design criteria in Seabrook produces a relative advantage for Zion in this regard. Therefore, we would expect to see a higher contribution from sequences involvong cross-train combinations of electric power, service water, component cooling water, and RHR systems at Seabrook, relative to Zion.

3. Other Plant Differences.

The remaining plant differences that were identified could be significant in the determination of the risk of power operation events, but are not found to be significant with respect to shutdown cooling risk. These differences include those in the containment heat removal systems (different configurations of containment spray and fan cooler systems), use of solid state versus relay technology in the safeguards actuation system at Seabrook and the ability to utilize Unit 2 equipment for Unit 1 and vice versa at Zion. There is a high degree of similarity between Zion and Seabrook in the procedures that govern shutdown operation. Of the differences in this area, there are distinct advantages to Seabrook (e.g. some of the local manual valve operations at Zion are performed remote manually from the control room at Seabrook.

UTILIZATION OF NSAC-84 RESULTS FOR SEABROOK

Following the design and procedures review and comparison, the dominant sequences from Table 6-1 in NSAC-84 were reviewed for applicability to Seabrook Station. The following conclusions were reached.

- Because of the similarity between the plants and the procedures, the dominant sequences from Table 6-1 are generally applicable to Seabrook.

- The NSAC-84 sequences would be expected to occur at the same frequency at Seabrook, except for those sequences involving inadvertent closure of RHR suction path MOVs and those involving combinations of support system faults and RHR train failures.

The sequences involving suction path MOV closures would occur at a lower frequency at Seabrook because Seabrook has a separate suction path for each pump. The frequency of valve closures was calculated as part of Top Event RM in NSAC-84. This top event asks whether RHR cooling is maintained during maintenance and refueling outages. The cause table for this event is shown in Table 1 (adapted from Table 5-5 of NSAC-84). Also shown in the table is a correction factor that shows the effect of two drop lines at Seabrook in lowering the frequency of "hardware failures" and "human errors." The derivation of the correction factors is explained below.

For spurious valve closure to cause a loss of RHR cooling at Seabrook Station, it is necessary to postulate either a common cause event involving one valve in each suction path, or a coincidence of a single valve closure and maintenance being performed on the other RHR train (these could also be maintenance in a support system of the other RHR train, but these sequences are separately accounted for below). The correction factor for this cause of RM is given by

$$\beta_{MOV} + (1-\beta_{MOV})\ (.5)\ Q_{RHRM} = .072$$

where

β_{MOV} = MOV Common Cause Parameter = .043 from SSPSA Section 6

Q_{RHRM} = Maintenance Unavailability of a Single RHR Pump Train During Shutdown

= 6.1×10^{-2} Based on Zion Data in NSAC-84

The factor of .5 is the chance that the maintenance is being done in one of two specific trains.

The correction factor for errors in inverter switching is given by

$$.5\ Q_{RHRM} = .031$$

The result of the above corrections to this top event is a reduction in the failure frequency to a factor of .145. This factor was applied to applicable sequences in Table 6-1 and the following results were obtained:

Results	Core Melt Frequency	
	Mean	Median
NSAC-84 Results for Cold Shutdown	1.8×10^{-5}	2.6×10^{-6}
Results Corrected for two RHR Suction Paths	7.6×10^{-6}	1.1×10^{-6}*

*Estimated as source factor reduction as calculated for mean results.

Hence, because of the dominance of the valve closure events in the NSAC-84 results, the effect of having two suction paths is a reduction of core damage frequency of the NSAC-84 sequences at Seabrook by a factor of about 2.

ANALYSIS OF SEABROOK SUPPORT STATE SEQUENCES

Because of differences in the support system interfaces with the RHR system and because these particular differences are unfavorable for Seabrook, separate event tree analyses were performed to cover these events for Seabrook. The following initially events were selected for this analysis.

Designator	Initiating Event
LOSP	Loss of Offsite Power
L1RH	Loss of One RHR Train
L1PC	Loss of One PCC Train
L2PC	Loss of Both PCC Trains
L1SW	Loss of One Service Water Train
L2SW	Loss of Both Service Water Trains

As shown in Figure 1, these initiating events were first analyzed in support system event trees whose sequences result in one of five different plant states. The plant states denote the number of RHR trains and safety grade AC power trains rendered unavailable by the combination of the initiating event and support system failures. These states together with the sequences borrowed from NSAC-84 as corrected for Seabrook were then fed into a frontline system event tree, which considers additional events needed to resolve the end states of the event sequences in terms of release categories. This main line event tree is based in the event sequence diagram in Figure 2. In this analysis, the NSAC-84 sequences were assigned to support state R2E0 (loss of both trains of RHR with both trains of AC power available).

The event tree quantifications for L1RH, L1PC, L1SW, and LOSP are shown in Figures 3, 4, 5, and 6, respectively. The quantifications were based on the SSPSA and RMEPS results for the support systems and initiators, except for maintenance unavailability. Train B of all systems was assumed to be unavailable for maintenance with a conservative value of unavailability of 0.1. This more than accounts for the higher chance of maintenance during plant shutdown. The initiating events L2PC and L2SW are assigned directly to support state R2E0 because of a very small chance of electric power failure with no loss of offsite power. The results of the analysis up to the point of support state are presented in Table 2, which is organized into three types of events: Type 1 is events with one RHR train unavailable (R1E0, R1E1); Type 2, with two RHR trains unavailable (R2E0, R2E1, and R2E2), and Type 3 is the set of NSAC-84 sequences. When the sequences are combined according to support state, the following results are obtained.

Support State	Mean Frequency (events per reactor-year)
R1E0	1.7×10^{-1}
R1E1	3.3×10^{-5}
R2E0	2.0×10^{-3}
R2E1	2.0×10^{-7}
R2E2	3.8×10^{-7}

RXEY = Sequence with X RHR trains and Y electric power train unavailable.

The event sequence diagram in Figure 2 defines the possible progression for Type 1, 2, and 3 support state sequences. For Type 1 and

2 sequences, consideration is given to operator recovery to prevent core melt. On the other hand, such consideration is not made for Type 3 because such actions are already considered in NSAC-84. Next, the RO event question whether the RCS pressure boundary is open initially; i.e., vessel head or steam generator manway cover is removed. For RO closed sequences, the ESD tracks the possible developement of interfacing LOCA conditions either through check valve closures or RHR system repressurization via Top Events CV, RV, and MC. For pressure boundary open sequences or all other nonbypass sequences, consideration is given to whether large and small penetrations are initially open and, when open, whether or not operator actions to secure these penetrations are successful. Since the containment sprays are not tracked in the ESD for simplicity, successfully isolated sequences could result in either a containment intact (S5) or delayed overpressurization (S3). Those with large or small bypass sequences are assigned to S6 and S2, respectively.

EVENT TREE QUANTIFICATION

With reference to Figure 1, the event trees were quantified in two stages. First the support system event trees were quantified for each initiating event resulting in the quantification of the unconditional frequencies of 5 different RHR support states. RIEO, RlEl, R2EO, RZEI and R2EZ (where RXEY is the state in whcih X RHR trains and and Y trains of safety grade AC electric power are rendered unavailable. Then, the main line event tree was quantified 6 times, one for each RHR state and a separate quantification for the sequences borrowed from the NSAC-84 results. The derivation of the event tree split fractions for each event tree quantification.

Support System Event Trees (Figure 3, 4, 5 and 6)

The support system event trees were quantified for the following initiating events.

L1RH - loss of 1 RHR train
L1PC - loss of 1 PCC train
L2PC - loss of both PCC trains
L1SW - loss of 1 service water train
L2WS - loss of both service water trains
LOSP - loss of offsite power

L1RH

The frequency of loss of 1 RHR train during shutdown was estimated using the following model.

$$\phi L1RH = 2\lambda_{RHR}\ t_{RHR}$$

where ϕ_{L1RH} = frequency of the initiating event (events per reactor year)

λ_{RHR} = failure rate of 1 RHR train (dominated by the RHR pump)

t_{RHR} = the number of hours per year in shutdown

Note the mission of MOV closure events in the above is by design, these events are included in the "type 3 events" borrowed from NSAC-84 and corrected for Seabrook having 2 RHR suction paths from the RCS.

The time on RHR, t_{RHR}, is estimated using zion experience, whcih is viewed as a conservative assumption for Seabrook. The reason for this view is that the zion experience is worse than average for PWRs and does reflect the generally higher availability factors of the Yankee system of plants (Maine Yankee, Vermont Yankee, Yankee Nuclear Power Station). In the first 16 reactor-years of experience of zion 1 and 2 there were 12 refueling outages of average duration 1,992 hrs and 3.05 maintenance outages per unit-year of average duration 488 hrs.

$$t_{RHR} = \frac{12 \text{ refuelings}}{16 \text{ reactor-years}} + 3.05 \text{ maintence events/reactor-year}$$

$$\times\ 488 \text{ hrs/outage} = 2982 \text{ hrs/year}$$

Hence, in the first 16 reactor-years at zion the plant was shut down about 34% of the time. We conservatively assume the same value for the plant lifetime at Seabrook.

The support system event tree quantification for L1RH is shown in Figure 3. In this event tree, it is assumed that the plant is initially being cooled with RHR train A and train B is in the standby. Becuase of the strict train-wise dependencies at Seabrrok Station, critical operation of RHR train a precludes unavailability of service water and PCC trains A, since both are needed to operate RHR train A. In normal power operation, the unavailability of single service water and PCC trains is very low- no greater than 10^{-3} to 10^{-2} per train. However, during plant shutdown, the unavailability due to maintenance is generally higher. For example, at sion the plant specific data shows single train maintenance unavailabilities of RHR, SW and PCC the range of .03 to .06. It is conservatively assumed in the analysis that <u>all</u> safety grade non-operating subsystem have shutdown maintenance unavailabilities of 10^{-1}. This is greater than any shutdown maintence unavailabilities observed in the data.

L1PC

The loss of one PCC train initiating event is analyzed in Figure 4. The initiating event frequency is estimated with the following model.

$$\phi_{L1PC} = 2\lambda_{pc}\ t_{RHR} \left[\lambda_s + \lambda_{pc}\ \tau\right]$$

where

λ_{pc} = failure rate of 1 PCC pump = 3.4(.5) per hour from PLG-0300 Section 6.

t_{RHR} = (.34) (8760) based on L1RH analysis

λ_s = failure rate to start of standby PCC pump = 2.4×10^{-3} from PLG-0300 Section 6.

τ = mean time to repair the initcally running pump = 24 hours per PLG-0300 Section 6.

Hence $\phi_{L1PC} = 6.3 \times 10^{-4}$/reactor-year

The L1PC event tree is quantified using information from the RMEPS and SSPSA for service water train A and the same train B maintenance assumptions as with the L1RH event.

L1SW

The configuration and failure rate of the SW system are comparable to the SW system, i.e., each train has an operating and a standby pump. The event tree for L1SW is quantified in Figure 5. The initiating event frequency is the same as that for L1PC. Note that this analysis includes the tunnel SW system only. The SW cooling tower system is considered in the subsequent recovery analysis.

LOSP

The loss of offsite power event tree is quantified in Figure 6. The initiating event frequency is estimated using the following model:

$$\phi_{LOSP} = \lambda_{LOSP}\, t_{RHR}\, EPR$$

where

LOSP = frequency of LOSP = .135 events/year (SSPSA Section 6)

t_{RHR} = time on RHR = .35 (see L1RH above)

EPR = frequency of non-recovery of LOSP before core damage = .01 (assumed)

The above assessment for EPR can be compared with EPR-1 in the SSPSA, a value of .03 for full power operation. The.01 value is viewed as conservative in comparison with EPR-1 since the time constants for core recovery are much longer during plant shutdown.

The event tree split fraction for LOSP in Figure 6 are based on the results of the SSPSA and RMEPS for train A and 10% maintenance unavailability for train B used for all shutdown loss of cooling events.

L2PC and L2SW

The initiating event frequency for loss of both trains of PCC and loss of both trains of service water are estimated with the following model.

$$\phi_{2PC} = \phi_{L2PC} \frac{t_{RHR}}{t_{po}} \qquad \phi_{L2SW} = \frac{\phi_{L2SW}}{t_{po}}$$

where

ϕ = frequency of same event during power operation from SSPSA

t_{po} = hours of power operation assumed in SSPSA (8760)

Top Event RH

For the L1RH and other R1A0 support states sequences, it is assumed that train A is initially used to provide RHR and train B is in standby. Hence, it is the train A subsystem that is involved in the initiating event. For these conditions, the failure to provide continued RHR cooling is estimated from the following model.

$$\emptyset_{RH} = \emptyset_m + \lambda_{ps} + \lambda_{PR} \tau_p$$

where

$\emptyset m$ = RHR pumps train maintenance unavailability during shutdown
= 6×10^{-2} from Zion data in NSAC-84

λps = standby pump failure to start vote
= 3.3(-3) from SSPSA Section 6

λpr = running pump failure to run vote
= 3.4×10^{-5}/hour from SSPSA Section 6

τ_{0p} = meantime to repair the initially failed pump = 21 hours from SSPSA Section 6

$\emptyset_{RH}$ = 6(-2) + 3.3(-3) + 3.4(-5)(21) = 6.4(-2)

For R1A1, the same model is used except the standby pump must run longer to cover the repair time of a diesel generator - assumed to be one week. Hence, for R2E0, R2E1, and R2E2, $\emptyset_{RH}$ = 1

0_{RH} = 6(-2) + 3.3(-3) + 3.4(-5)(168) = 6.9(-2)

Top Event OM

R1E0 represents the most ideal conditions and minimum stress levels of these considered for OM for these conditions, OM is estimated from:

OM = DE1 = SC = BF

where

DE1 = operators fail to recognize that RCS heat removal should be restored after running RCS pump stops.

SC = operators fail to align and restart a core cooling system

BF = operators fail to provide long term makeup to chargining system

Using appropriate values from Table 5-6 of NSAC-80, the following quantification is made:

OM = 1.0(-5) + 5.0(-4) + 1.0(-5) = 5.2(-4)

As shown in Table 3, higher values are used for the remaining support states to reflect different and progressively greater stress and comparison levels going through the sequence R1E0, R2E0, R1E1, R2E1, R2E2. For type 3 events, OM = 1 to avoid double counting recovery already considered in NSAC-84.

Top Event RO

The fraction of time the pressure boundary is open is keyed to the time assumed for RHR shutdown cooling. For consistency, since Zion data was used to quantify the latter, it must be used to quantify the former. From Table 3-4 of NSAC-84, the total time the RCS is opened during maintenance outages is 5,014 hours. From table 3-1, the RCS open time during refueling outages is above 3,000 hours. Hence, overall the 31,687 hours of the Zion outage experience, the RCS was opened (8014/31,687) = .25 of the time.

Top Event CV

The frequency of CV is quantified per our response to question 47.

$$\phi CV = 5.5 \times 10^{-4}$$

Top Event RV

This event is quantified in RMEPS; it is estimated using:

$$\phi RV = 2\lambda RV = 4.8(-5)$$

where λRV is the failure rate of each RHR relief value = 2.4(-5) from SSPSA Section 6

Top Event MC

For both trains of AC power available, the top event is estimated using the following model.

$$MC = 2(\lambda MOV^2 + B_{MOV}\lambda MOV) + 4(\lambda CV^2 + B_{CV}\lambda CV)$$

where

λMOV = failure rate (fail to close on demand) = for MOVs - 4.3(-3) per SSPSA

λCV = failure rate (fail to close on demand) for check valves = 5.5(-4) per question 75 response.

B_{MOV}; B_{CV} = Beta factors for each type of valve = .1

hence

$$\phi MC = 2[(4.3 \times 10^{-3})^2 + (4.3 \times 10^{-3})(.1)] + 4[(5.5 \times 10^{-4})^2 + (5.5 \times 10^{-4})(.1)] = 1.1(-3)$$

For one AC power bus available (R1E1 and R2E1) there is only one MOV on each suction path potentially available. For these states

$$\phi MC = 2\lambda MOV + 4(\lambda CV\ B_{CV}) = 8.8(-3)$$

For two AC power basas unavailable $\phi MC = 1$

Top Event LI

This event question whether any large penetrations are open initially. Large is defined as a total equivalent single opening of greater than 3" in diamter. Examples of such penetrations are equipment hatch, personnel hatch, and containment purge penetrations. These penetrations maybe opened during shutdown unless fuel is being moved.

The chance that large penetrations are opened is highly dependent on ht reason for the shutdown. If the reason is refueling, steam generator manintenance or othere maintenance on reactor coolant system somponents, it is likely that large penetrations such as the equipment hatch will open. If on the other hand, the outage occurs due to need to repair or maintain equipment outside the containment, (e.g. turbine generator related maintenance) there would not be a compelling reason to open up large penetrations in the containment.

To reflect the above considerations, LI in assessed as a function of the status of event RD. If RO is true (reactor coolant system is opened), it is assumed that LI is true (large penetrations are open) 90% of the time. If RO is not true (reactor coolant system is closed), it is assumed that LI is true on 10% of the time. Note that at Zion, of the 8,014 hours during shutdown that the RCS was opened, the fuel was being shuffled for 1600 hours (roughly 160 hrs per refueling outage). Hence 80 percent of the time that the RCS was opened, it would have been permitted by tech specs to have the equipment hatch open.

Top Event OL

Given a large penetration is opened initially, the event questions whether the operators successfully close the penetrations before a potential release situation could develop. The probability of successful recovery is assessed as dependent on the RHR support state, i.e. the combination of the initiating event and the response of the plant support systems. At different support states then would be different levels of stress and confusion to inhibit operator recovery actions.

To provide an indication of the amount of time available to close the equipment hatch or other large penetrations, the time to core damage, taken as the time to uncover the core was estimated for the following cases:

Cases	Time to core uncovery (hr)
1. Reactor vessel head open with water level at hot leg nozzle midplane	
A. Loss of cooling at 2 days after shutdown	0.8
B. Loss of cooling at 30 days after shutdown	2.6
2. RCS filled at pressure $\leq$ 425 psig with	
A. Loss of cooling at 1 day after shutdown	5.4
B. Loss of cooling at 10 days after shutdown	14
C. Loss of cooling at 30 days after shutdown	22
3. Water at refueling level with	
A. Loss of cooling at 5 days after shutdown	72
B. Loss of cooling at 30 days after shutdown	162

It is not known how quickly the equipment hatch can be secured. Our current information is it would take several hours to attach, and up to 8-12 hours to secure all the bolts and establish a tight seal. Just how quickly this process can be accelerated is uncertain. To address this uncertainty, a base case and a bounding case are performed. In the base case, it is assumed that the mean time to close the hatch is 4 hours. In the bounding analysis, it is assumed that the hatch remains open with a probability of 1.

From the Zion data in NSAC-84, there were 10 reflueing outages and 24 forced maintenance outages resulting in an average outage duration of about 39 days. Of the entire 31,687 hours of outages, roughly 1% of the time the RCS was drained, 5% of the time refueling was taking place, 5% of the time the plant was not on RHR, and in most of the remaining 89% of the time, the reactor system was filled on RHR. Based on the above recovery time, and assuming a 4-hour hatch recovery time, it is seen that with the RCS drained to the hot leg nozzle midplanes, the chances of successful hatch recovery are not very high. While under all other conditions, the chances are high. Therefore, the base case and low stress levels, a value of .01 is used for failure to isolate large penetrations. For degraded RHR states, this valve is increased to correspond with higher stress levels, as indicated in Table 3. In the bounding case, a failure frequency of 1 is assured for all states.

Top Events SI and OS

It is conservatively assumed that small penetrations are open 90% of the time and the chances of recovery are assessed at levels compoarable to those for OL, even though all small penetrations can be isolated quickly.

Results

The results of this preliminary analysis of shutdown loss of cooling events are shown in Table 4 for the base case assumptions on event OL. To bound the consequences of these events, accident sequences were assigned to the existing PSA release categories, even though the release fractions for shutdown events would be expected to be considerably lower than those calculated for power operation events. Based on what is believed to be a very conservative set of assumptions in this base case, the impact of shutdown events is assessed to result in no greater than a 14% increase in core melt frequency, and an 18% increase in category S6 frequency. For the bounding case of no credit for event OL, the frequency of S6 would increase to about 5×10^{-6}/year.

The impact of these boumding estimates of shutdown events on the dose vs distance curves for 50 rem and 200 rem whole body gamma doses are shown in Figures 13 and 14 for the base case OL and bounding case OL assumptions, respectively. Our best current statement of risk levels is represented by Figure 13. As seen from this figure, the addition of shutdown events impacts the right tails of these curves, but the combined results at 1 mile are still less than the NUREG-0396 valves at 10 miles. Even with no credit for equipment hatch recovery as assumed Figure 14, the combined shutdown and power operation results fall below the NUREG-0396 10 mile levels at less than 2 miles. Hence, even a very conservative analysis of these events does not impact the conclusions of the sensitivity study. It is expected that a more detailed investigation of these events would result in much lower levels of risk than either set of results presented here.

REFERENCES

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

2. Pickard, Lowe and Garrick, Inc., "Seabrook Station Emergency Planning Sensitivity Study," prepared for Public Service Company of New Hampshire, New Hampshire Yankee Division, PLG-0465, April 1986.

3. Letter from S. M. Long, U.S. Nuclear Regulatory Commission Staff, to R. J. Harrison, Public Service Company of New Hampshire, "Request for Additional Information...," Docket Nos. 50-443 and 50-444.

4. 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.

5. Bley, D. C., and J. W. Stetkar, "Zion Nuclear Plant Residual Heat Removal PRA," EPRI/NSAC Report NSAC-84, July 1985.

6. Pickard, Lowe and Garrick, Inc., Westinghouse Electric Corporation, and Fauske & Associates, Inc., "Zion Probabilistic Safety Study," prepared for Commonwealth Edison Company, September 1981.

Table 1. CORRECTION OF NSAC-84 RESULTS FOR RHR LOSS TO ACCOUNT FOR 2 SUCTION PATHS

Failure Cause	Mean Value	Dominant Contributor	Seabrook Correction Factor	Revised Failure Frequency
Hardware Failures	6.08-2	Spurious Closure of RH8701 or RH8702	+ .072	4.38-3
Maintenance	7.37-3	Running RHR Pump Fails with Standby Pump Out for Maintenance	x 1	7.37-3
Human Errors	6.00-2	Errors During TSS 15.6.36 or Inverter Switching (RH8701 or RH8702 close)	x .031	1.86-3
Support System Failures	2.94-6	Component Cooling Water Heat Exchanger Failures	x 1	2.94-6
Dependent Component Failures	6.03-3	RHR Pumps Fail During Operation	x 1	6.03-3
Total	1.34-1			1.96-2

TABLE 2. CLASSIFICATION OF SUPPORT MODEL ACCIDENT SEQUENCES FOR MAIN LINE MODEL QUANTIFICATION CASES

Sequence	Frequency	Impact Vector				Event Classification
		AC Power		RHR		
		A	B	A	B	
Type 1 Events - One RHR Train Made Unavailable						
L1RH	1.7-1			X		R1E0
L1PC*PR	5.0-5			X		R1E0
LOSP*GBM*D1R	2.0-5		X		X	R1E1
LOSP*GA1*D1R	1.3-5	X		X		R1E1
L1SW*SR	5.1-6			X		R1E0
LOSP*PBM*PR	3.4-6				X	R1E0
LOSP*WBM*SR	3.8-7		X		X	R1E1
LOSP*PA2*PR	8.9-8			X		R1E0
LOSP*WA3*SR	5.6-8	X		X		R1E1
L1PC*WA3*SR	1.8-9			X		R1E0
Type 2 Events - Two RHR Trains Made Unavailable						
L1RH*PBM*PR	1.8-3			X	X	R2E0
L1RH*WBM*SR	2.0-4			X	X	R2E0
L1PC*PBM*PR	5.5-6			X	X	R2E0
L1SW*WBM*SR	6.3-7			X	X	R2E0
L1SW*PBM*SR	5.7-7			X	X	R2E0
LOSP*GA1*GBM*D2R	3.5-7	X	X	X	X	R2E2
LOSP*GA1*PBM*D1R*PR	1.4-7	X		X	X	R2E1
LOSP*PA2*GBM*D1R	6.0-8		X	X	X	R2E1
LOSP*GA1*WBM*D1R*SR	1.5-8	X	X	X	X	R2E2
LOSP*PA2*PBM*PR	9.8-9			X	X	R2E0
LOSP*WA3*WBM*SR	6.9-9	X	X	X	X	R2E2
LOSP*WA3*PBM*SR	6.2-9	X		X	X	R2E1
LOSP*WA3*GBM*D1R*SR	3.9-9	X	X	X	X	R2E2
LOSP*PA2*WBM*SR	1.1-9		X	X	X	R2E1
L1PC*WA3*WBM*SR	2.2-10			X	X	R2E0
L1PC*WA3*PBM*SR	2.0-10			X	X	R2E0
Type 3 Events - From NSAC-84	7.6-6			X	X	R2E0

NOTE: Exponential notation is indicated in abbreviated form; i.e., 1.7-1 = 1.7×10^{-1}.

TABLE 3. SUMMARY OF SPLIT FRACTIONS FOR MAIN LINE EVENT TREE

Main Line Event Tree Top Event	Event Tree Quantification Cases					
	Type 1		Type 2			Type 3 (NSAC-84)
	R1E0	R1E1	R2E0	R2E1	R2E2	
$\overline{RH}$	6.4-2	6.9-2	1	1	1	1
$\overline{OM}$	5.2-4	1.0-2	2.0-3	1.0-1	1	1
$\overline{RO}$	.75	.75	.75	.75	.75	.75
$\overline{CV}$	5.5-4	5.5-4	5.5-4	5.5-4	5.5-4	5.5-4
$\overline{RV}$	4.8-5	4.8-5	4.8-5	4.8-5	4.8-5	4.8-5
$\overline{MC}$	1.1-3	8.8-3	1.1-3	8.8-3	1	1.1-3
$\overline{LT}$|RO	.90	.90	.90	.90	.90	.90
$\overline{LT}$|$\overline{RO}$	.10	.10	.10	.10	.10	.10
$\overline{OL}$	1.0-2	3.0-2	1.0-2	3.0-2	.10	1.0-2
$\overline{ST}$	.90	.90	.90	.90	.90	.90
$\overline{OS}$	1.0-2	.10	1.0-2	.10	1	1.0-2

X = Event Success

$\overline{X}$ = Event Failure

NOTE: Exponential notation is indicated in abbreviated form; i.e., 6.4-2 = 6.4 x 10^{-2}.

TABLE 4. KEY RESULTS OF SHUTDOWN SEQUENCES FOR SEABROOK STATION

Event Tree	Release Category		
	S5 or S3	S2	S6
R1E0	5.6-6	5.0-8	1.9-8
R1E1	1.9-8	1.7-9	2.0-10
R2E0	4.0-6	3.5-8	1.4-8
R2E1	1.8-8	1.8-9	1.9-10
R2E2	3.2-8	3.3-7	1.2-8
Type 3	7.5-6	6.8-8	2.6-8
Total for Shutdown Events	1.7-5	4.9-7	7.1-8
Total for Power Operation Events	1.1-4	2.0-5	3.2-7
Percent Increase with Shutdown Events	13.3	2.4	18.2

NOTE: Exponential notation is indicated in abbreviated form; i.e., 5.6-6 = 5.6×10^{-6}.

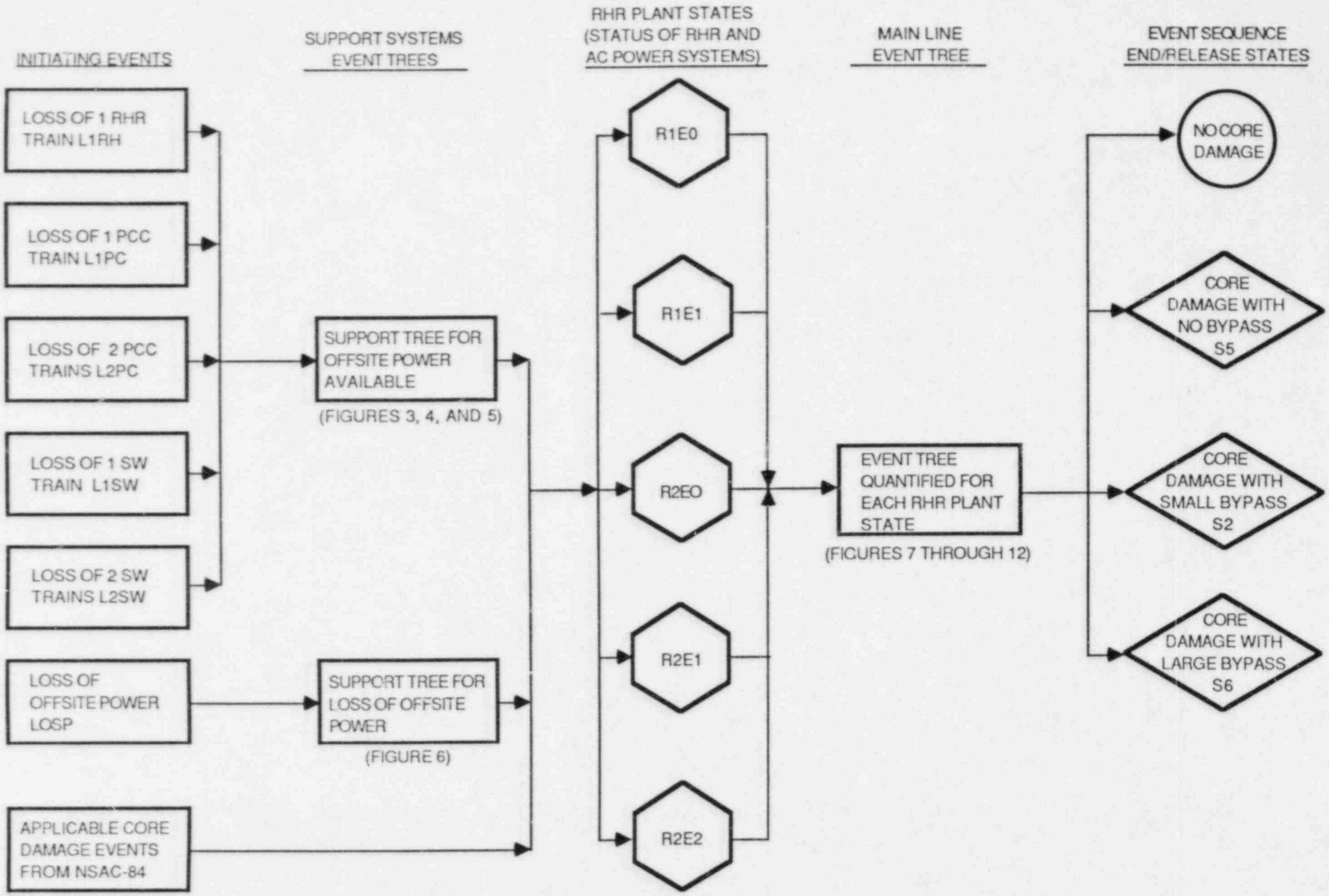


FIGURE 1. ACCIDENT SEQUENCE EVENT TREE MODEL FOR SEABROOK STATION SHUTDOWN COOLING EVENTS

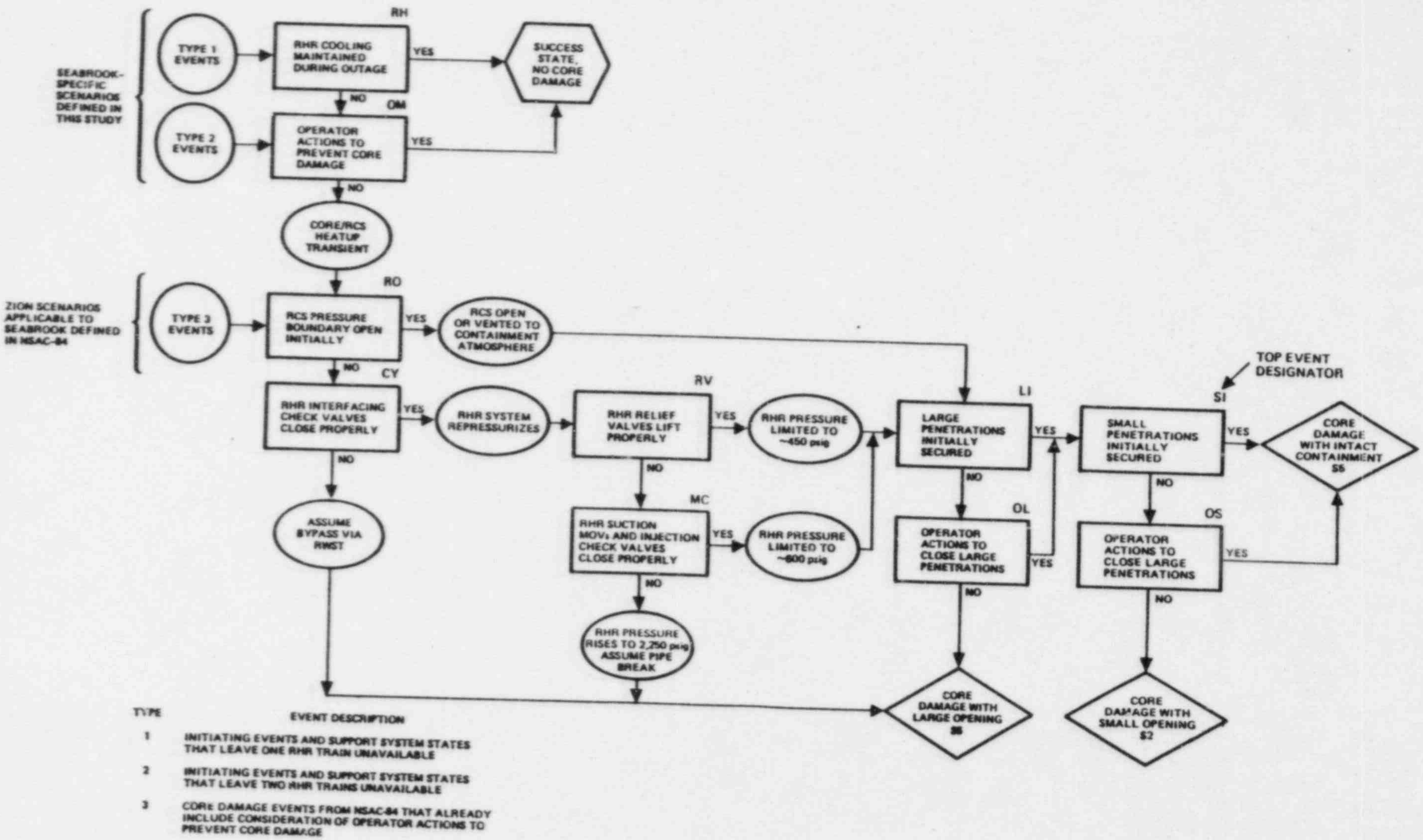
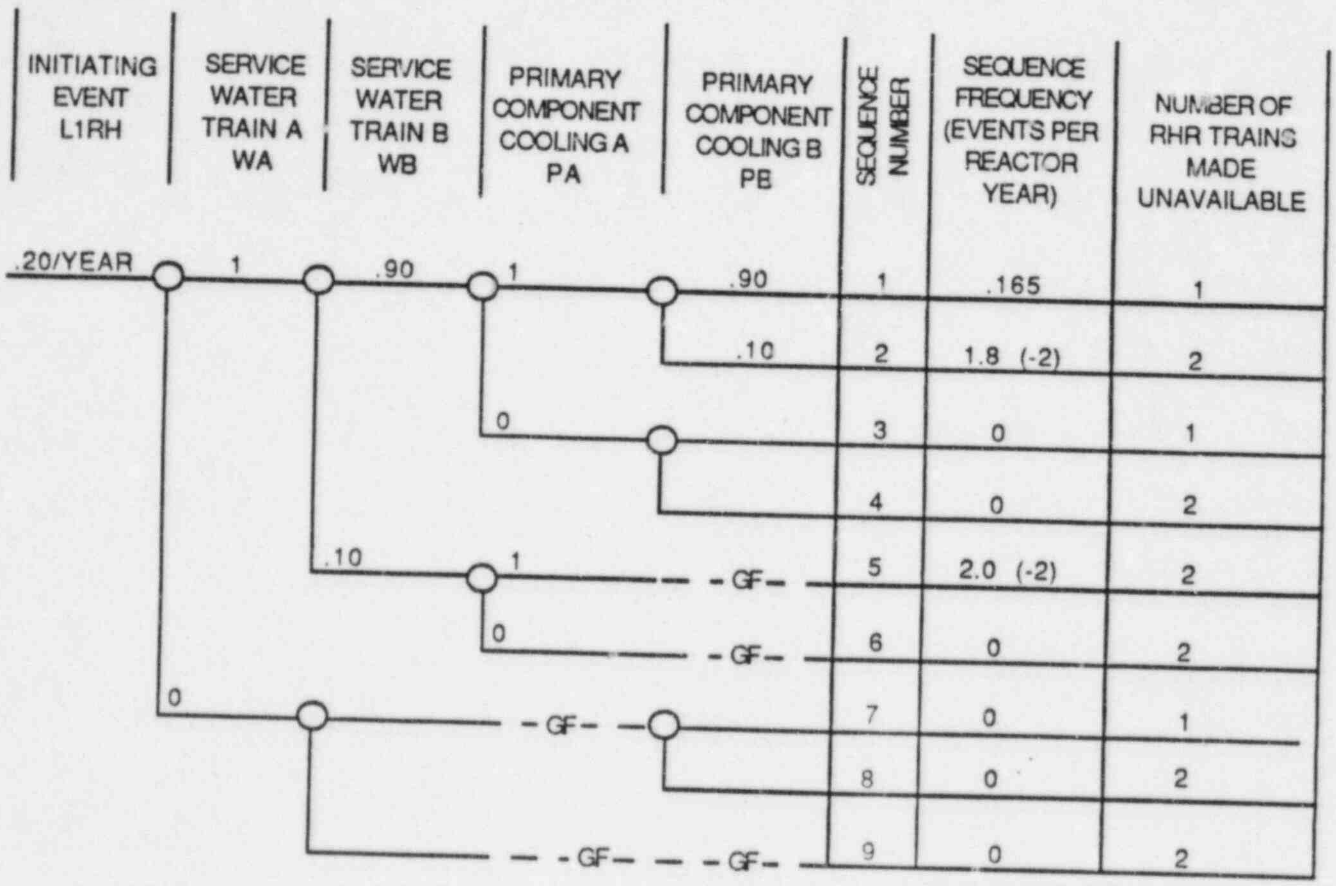


FIGURE 2. EVENT SEQUENCE DIAGRAM FOR CORE DAMAGE SEQUENCES AT SEABROOK INITIATED DURING PLANT SHUTDOWN



GF = GUARANTEED FAILURES

FIGURE 3. SUPPORT SYSTEM EVENT TREE QUANTIFICATION FOR LOSS OF ONE RHR TRAIN (L1RH)

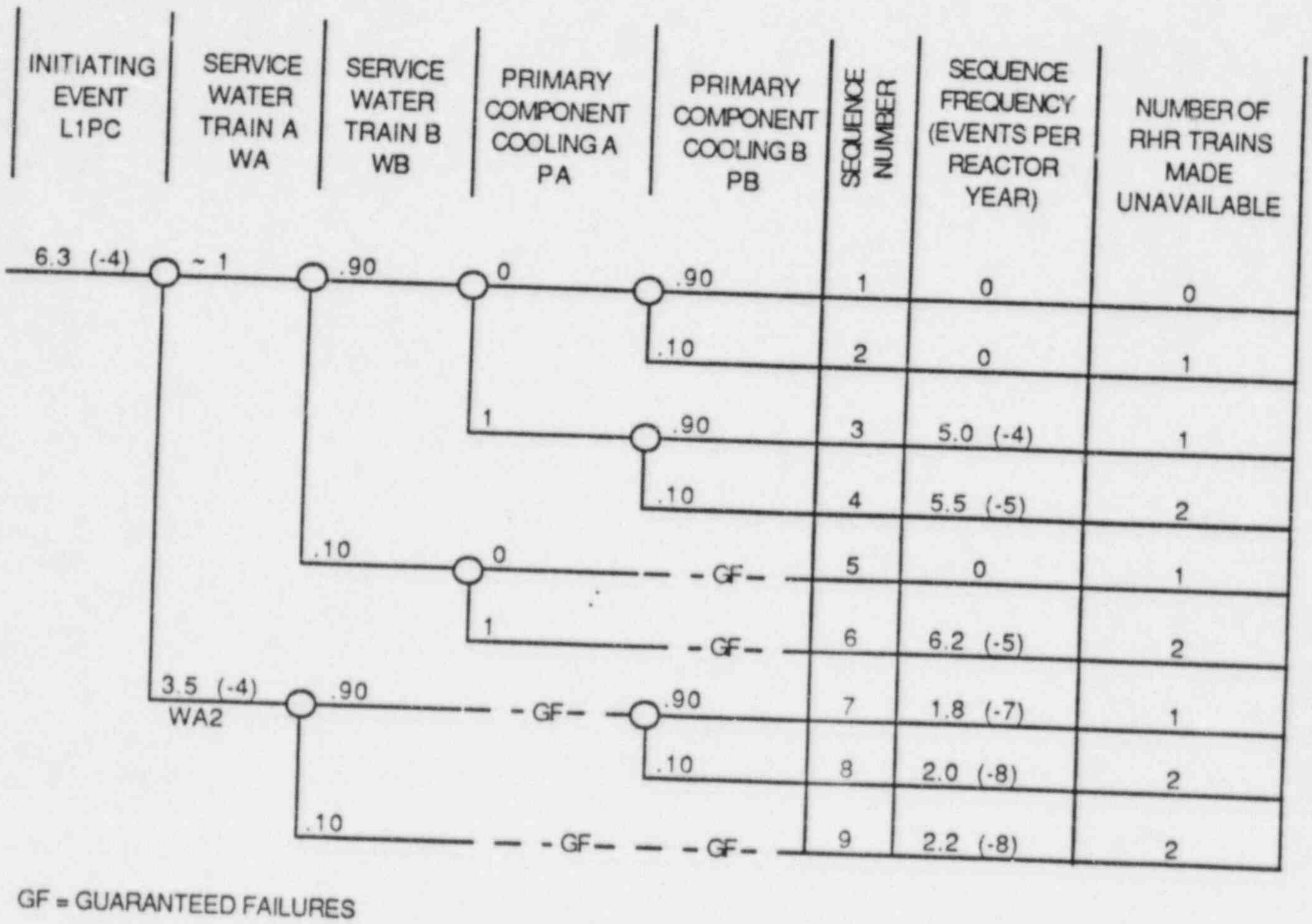


FIGURE 4. SUPPORT SYSTEM EVENT TREE QUANTIFICATION FOR LOSS OF ONE PCC TRAIN (L1PC)

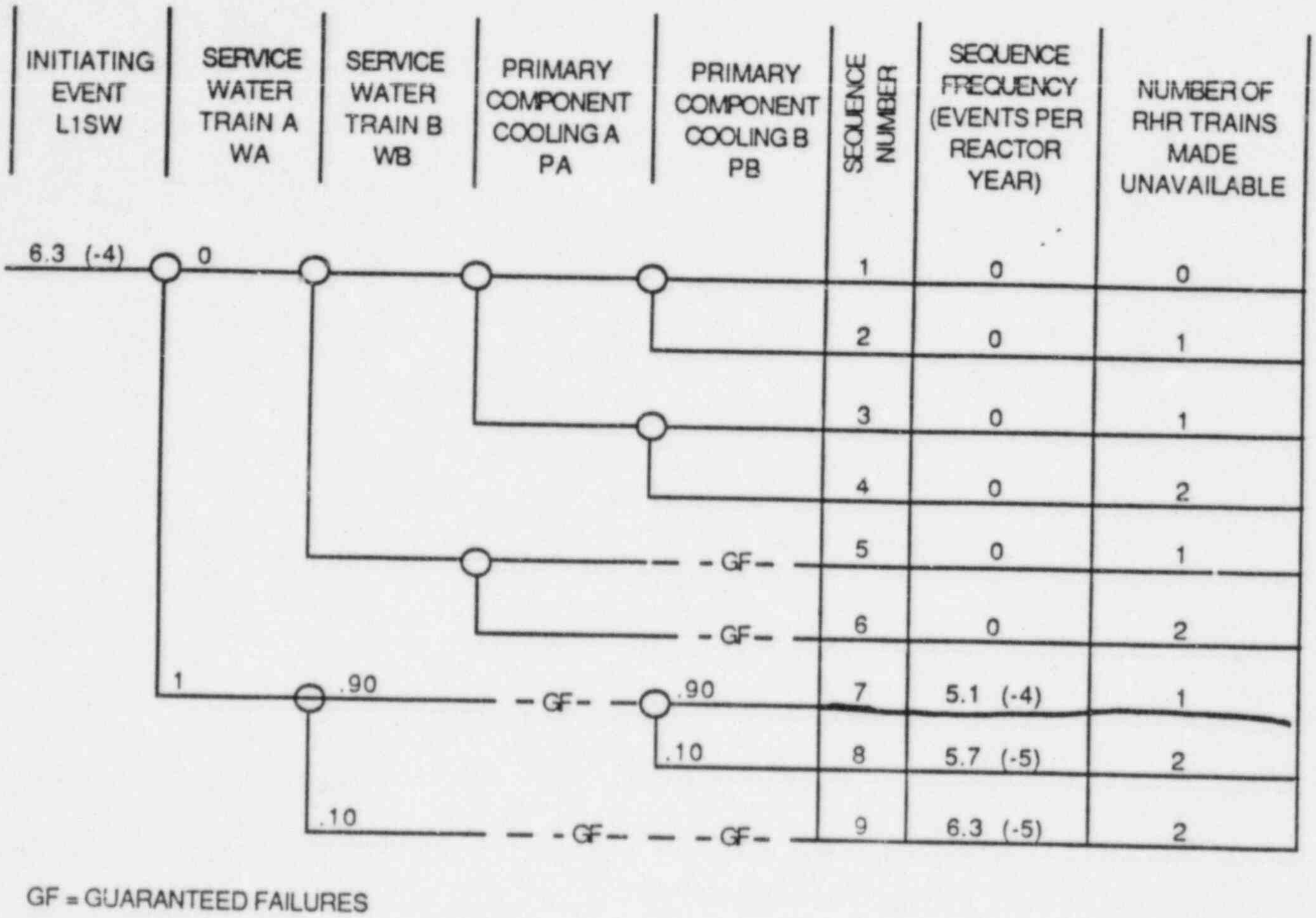
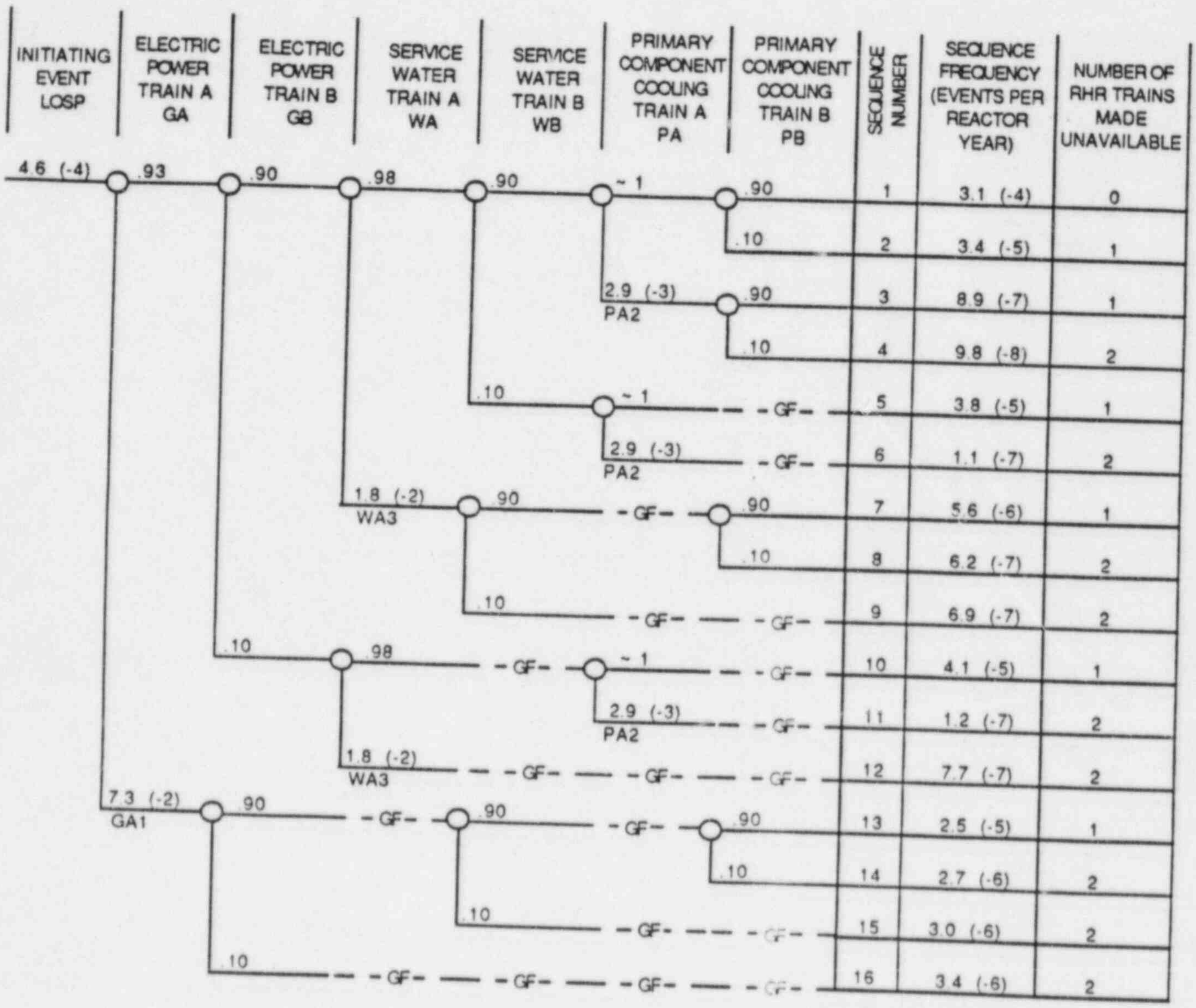


FIGURE 5. SUPPORT SYSTEM EVENT TREE QUANTIFICATION FOR LOSS OF ONE SERVICE WATER TRAIN (L1SW)



LOSP = LOSS OF OFFSITE POWER WITH NO RECOVERY BEFORE CORE DAMAGE
GF = GUARANTEED FAILURE

FIGURE 6. SUPPORT SYSTEM EVENT TREE QUANTIFICATION FOR LOSS OF OFFSITE POWER (LOSP)

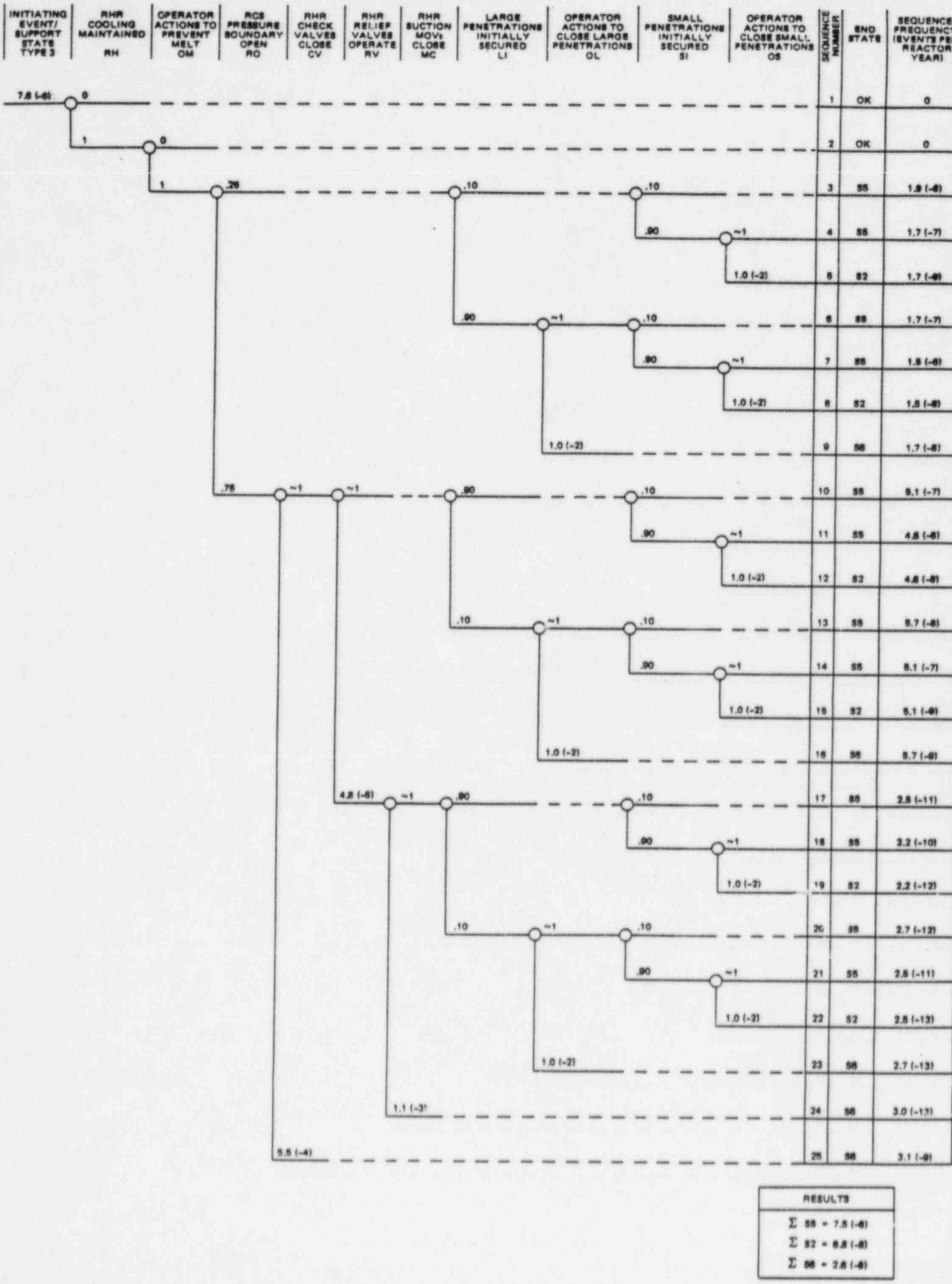


FIGURE 12. EVENT TREE QUANTIFICATION FOR TYPE 3 EVENTS - EVENTS TAKEN FROM NSAC-84 THAT ARE APPLICABLE TO SEABROOK STATION

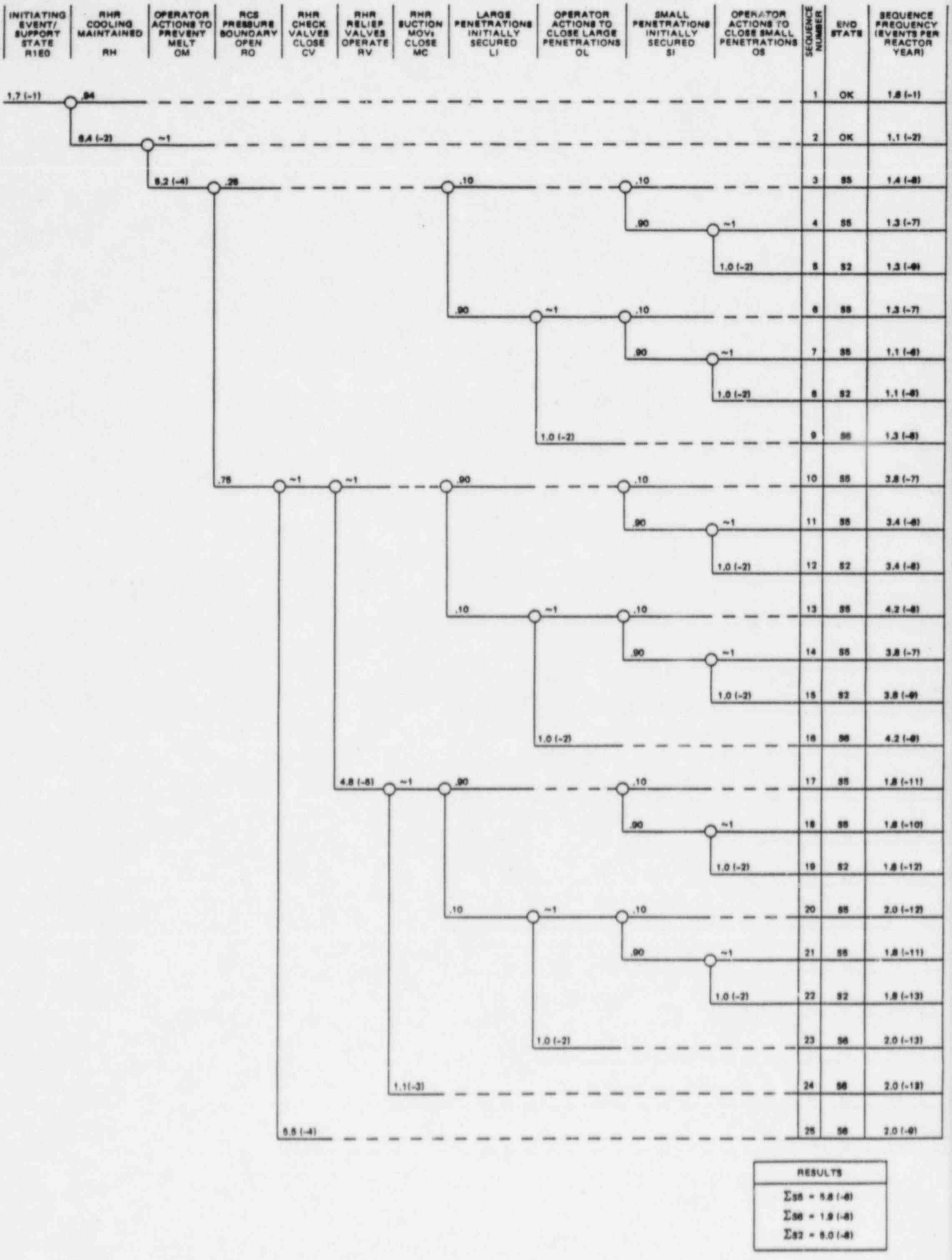
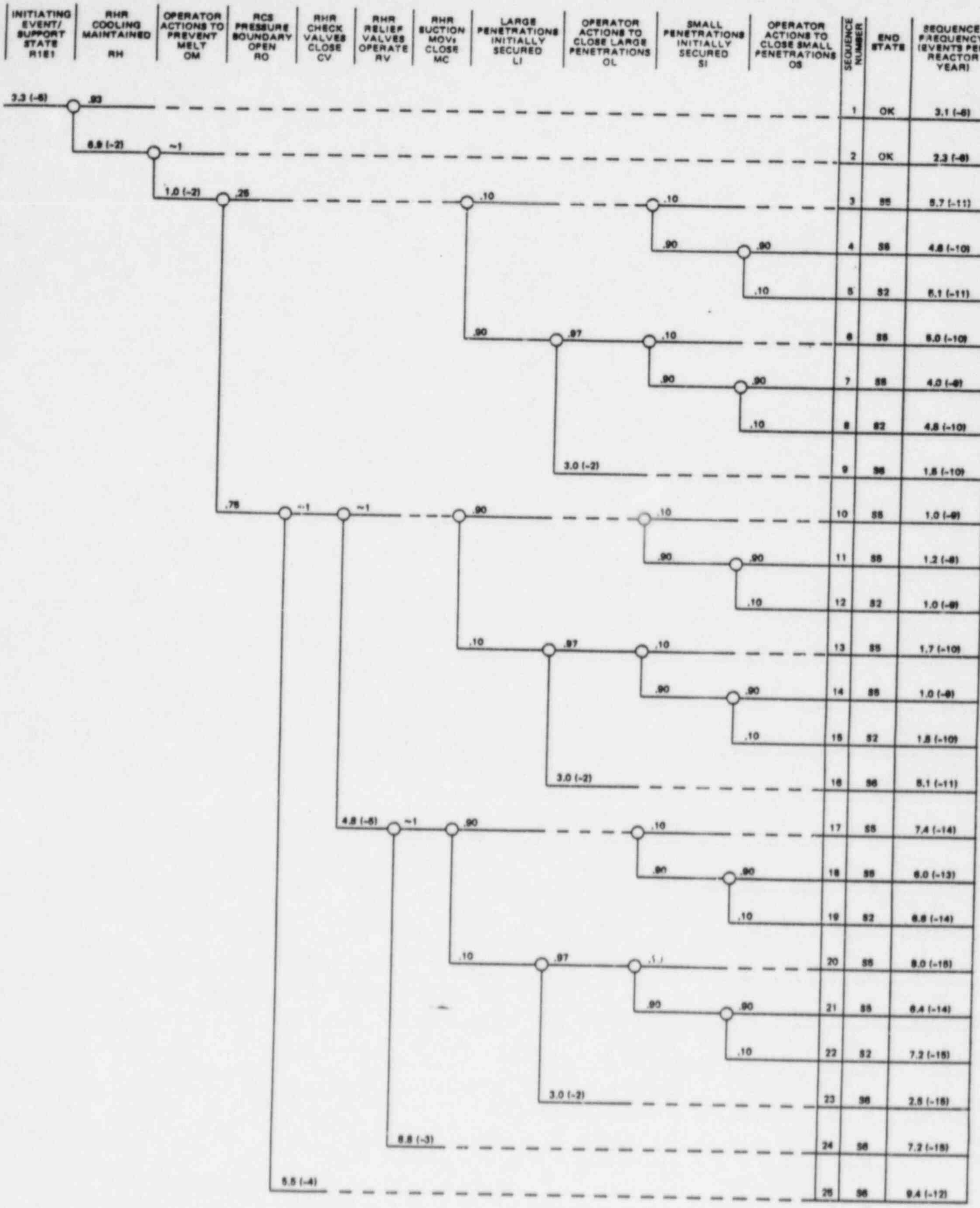


FIGURE 7. EVENT TREE QUANTIFICATION FOR R1E0 - EVENTS THAT FAIL ONE RHR TRAIN AND NO TRAINS OF SAFETY GRADE AC POWER



RESULTS
Σ S5 = 1.9 (-8)
Σ S2 = 1.7 (-9)
Σ S6 = 2.0 (-10)

FIGURE 8. EVENT TREE QUANTIFICATION FOR R1E1 - EVENTS THAT FAIL ONE RHR TRAIN AND ONE TRAIN OF SAFETY GRADE AC POWER

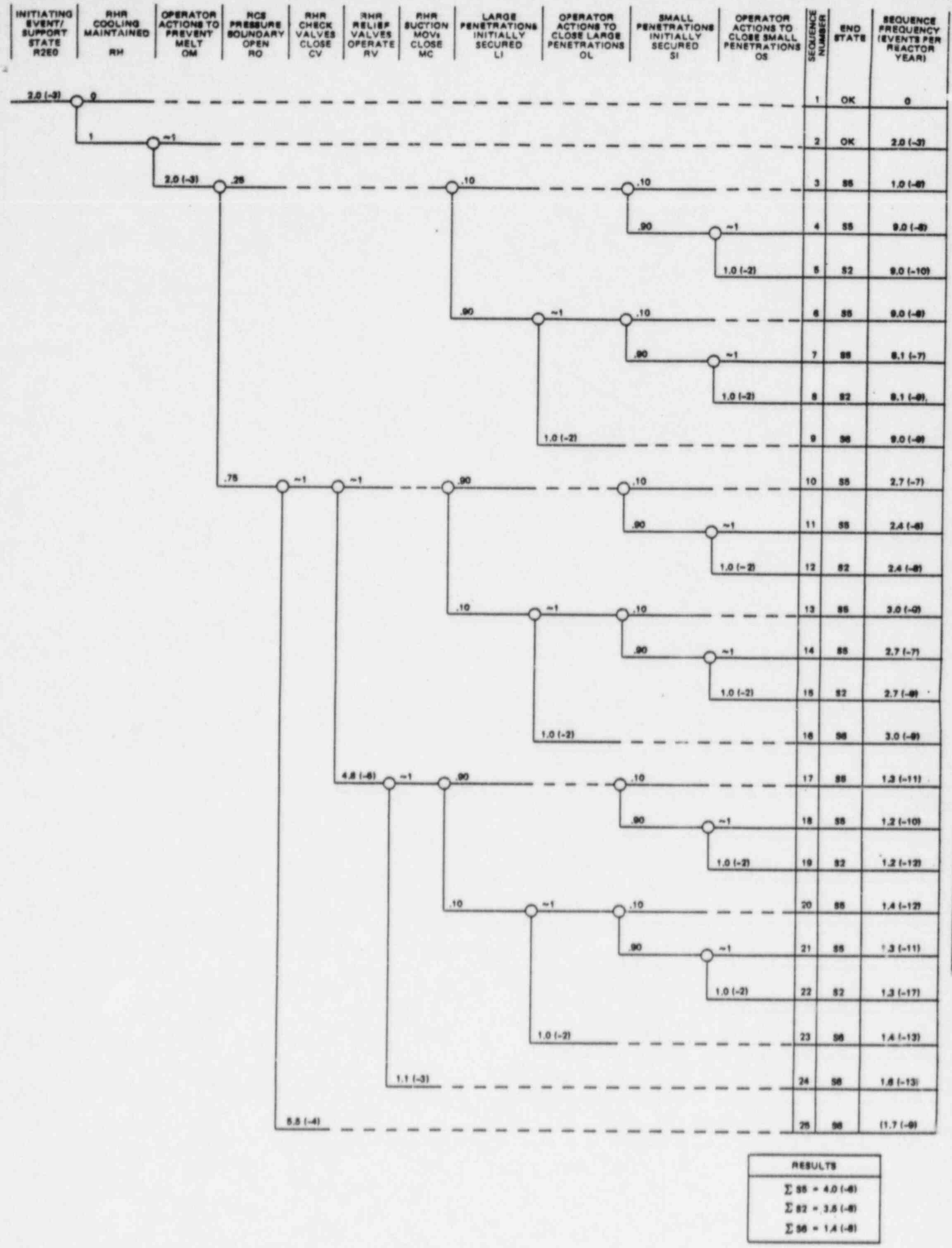
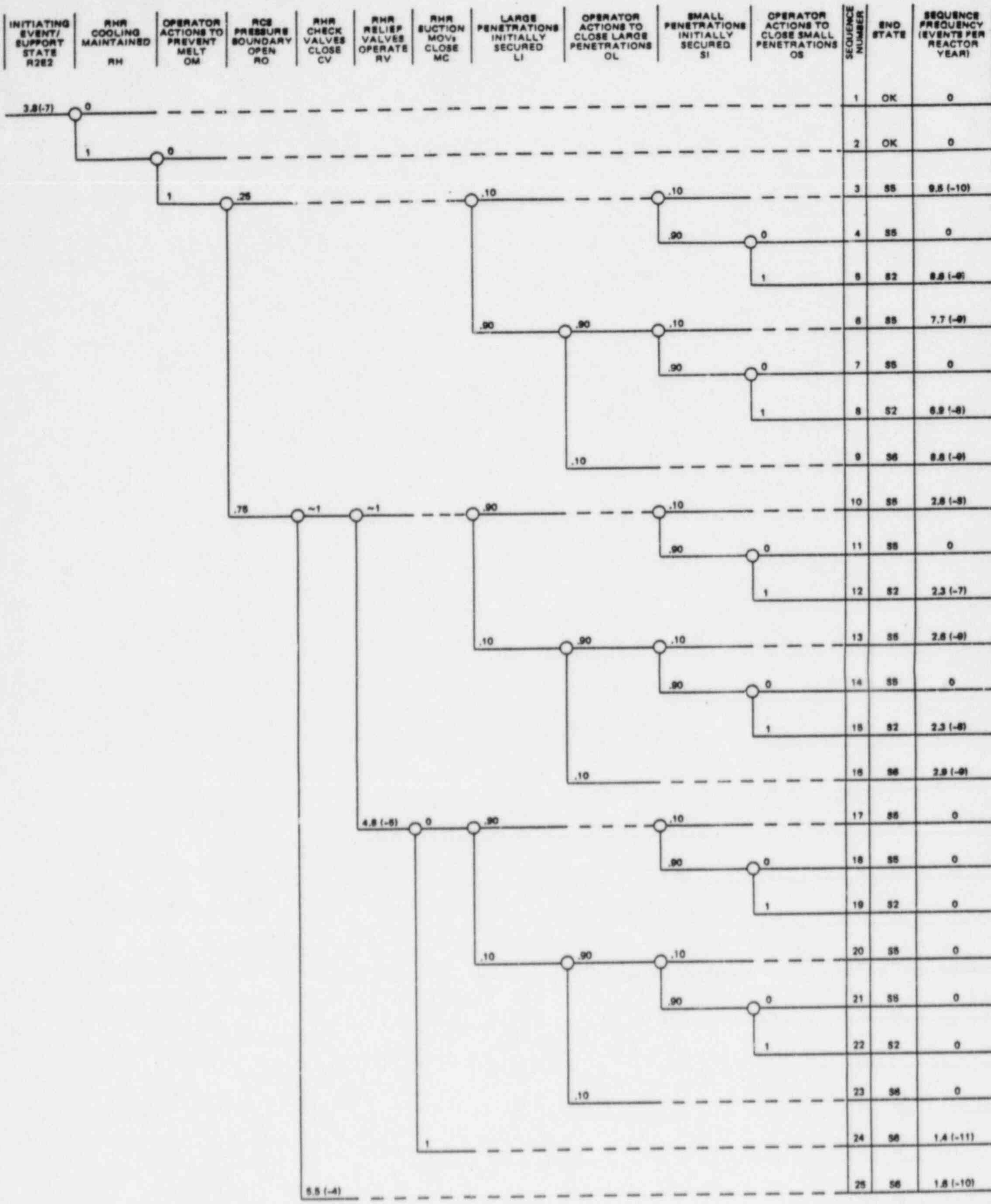


FIGURE 9. EVENT TREE QUANTIFICATION FOR R2EO - EVENTS THAT FAIL TWO RHR TRAINS AND NO TRAINS OF SAFETY GRADE AC POWER



RESULTS
ΣS5 = 3.7 (-8)
ΣS2 = 3.3 (-7)
ΣS6 = 1.2 (-8)

FIGURE 11. EVENT TREE QUANTIFICATION FOR R2E2 - EVENTS THAT FAIL TWO TRAINS OF RHR AND TWO TRAINS OF SAFETY GRADE AC POWER

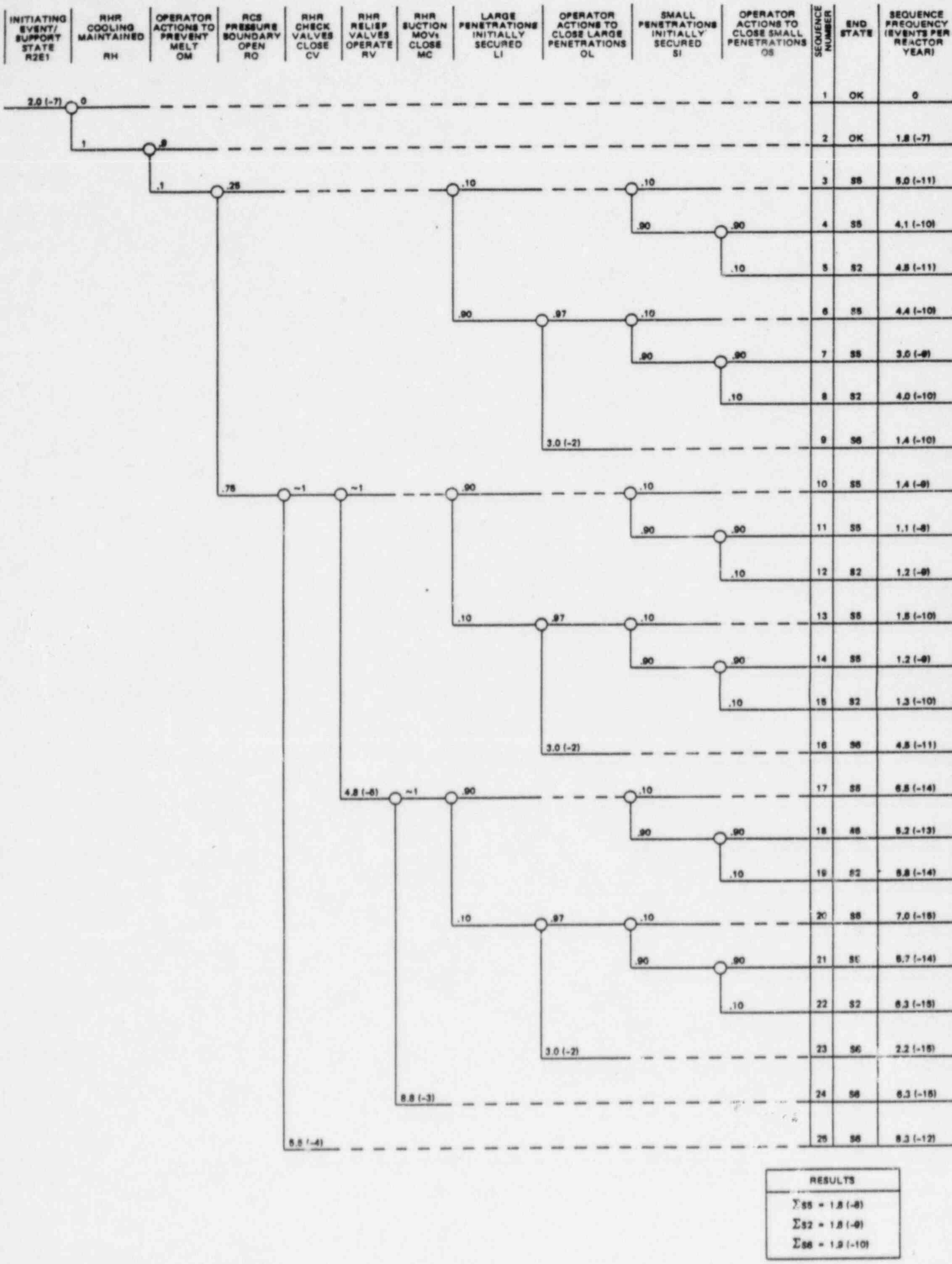
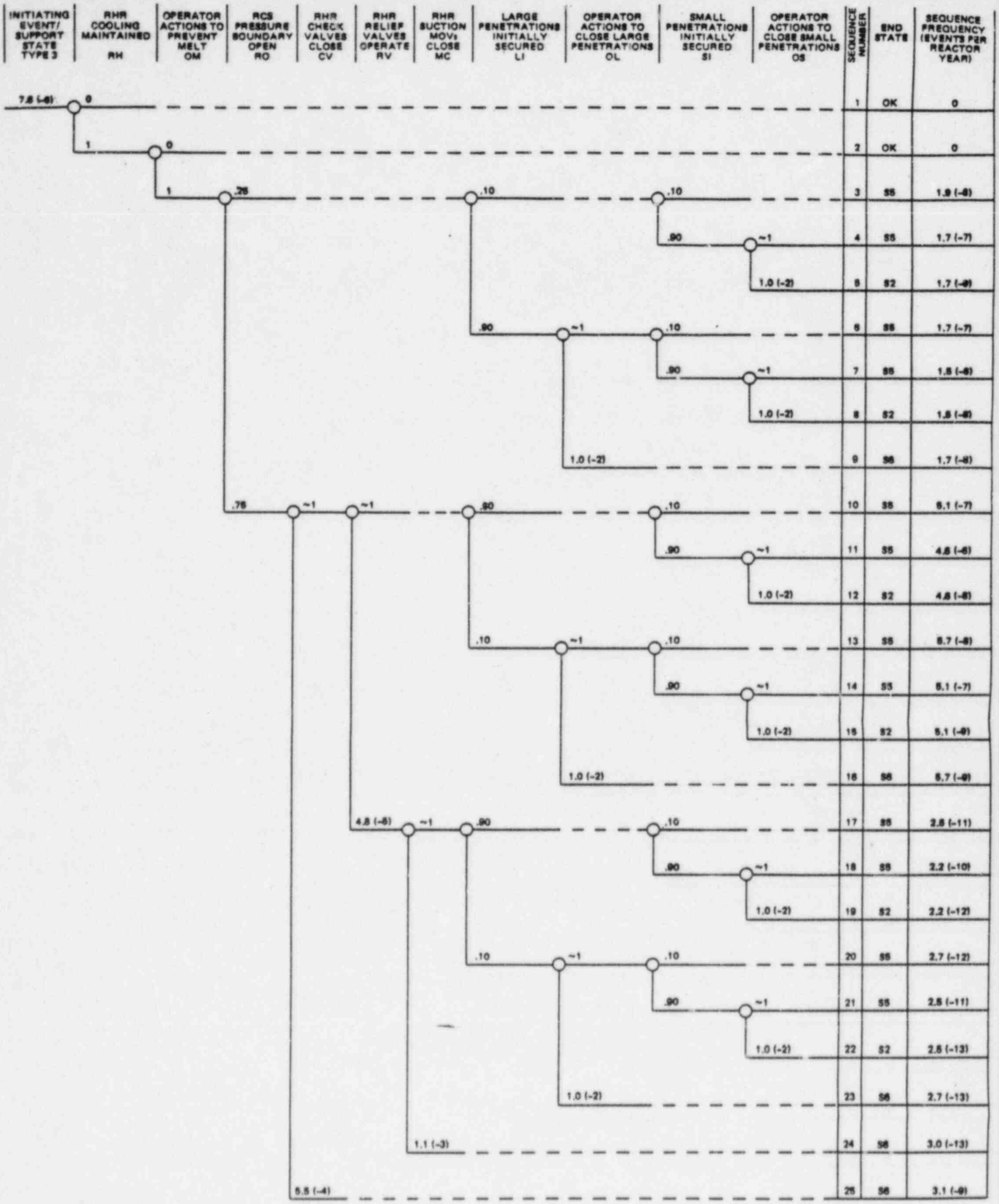


FIGURE 10. EVENT TREE QUANTIFICATION FOR R2E1 - EVENTS THAT FAIL TWO RHR TRAINS AND ONE TRAIN OF SAFETY GRADE AC POWER



RESULTS
Σ S5 = 7.5 (-6)
Σ S2 = 6.8 (-8)
Σ S6 = 2.8 (-8)

FIGURE 12. EVENT TREE QUANTIFICATION FOR TYPE 3 EVENTS - EVENTS TAKEN FROM NSAC-84 THAT ARE APPLICABLE TO SEABROOK STATION

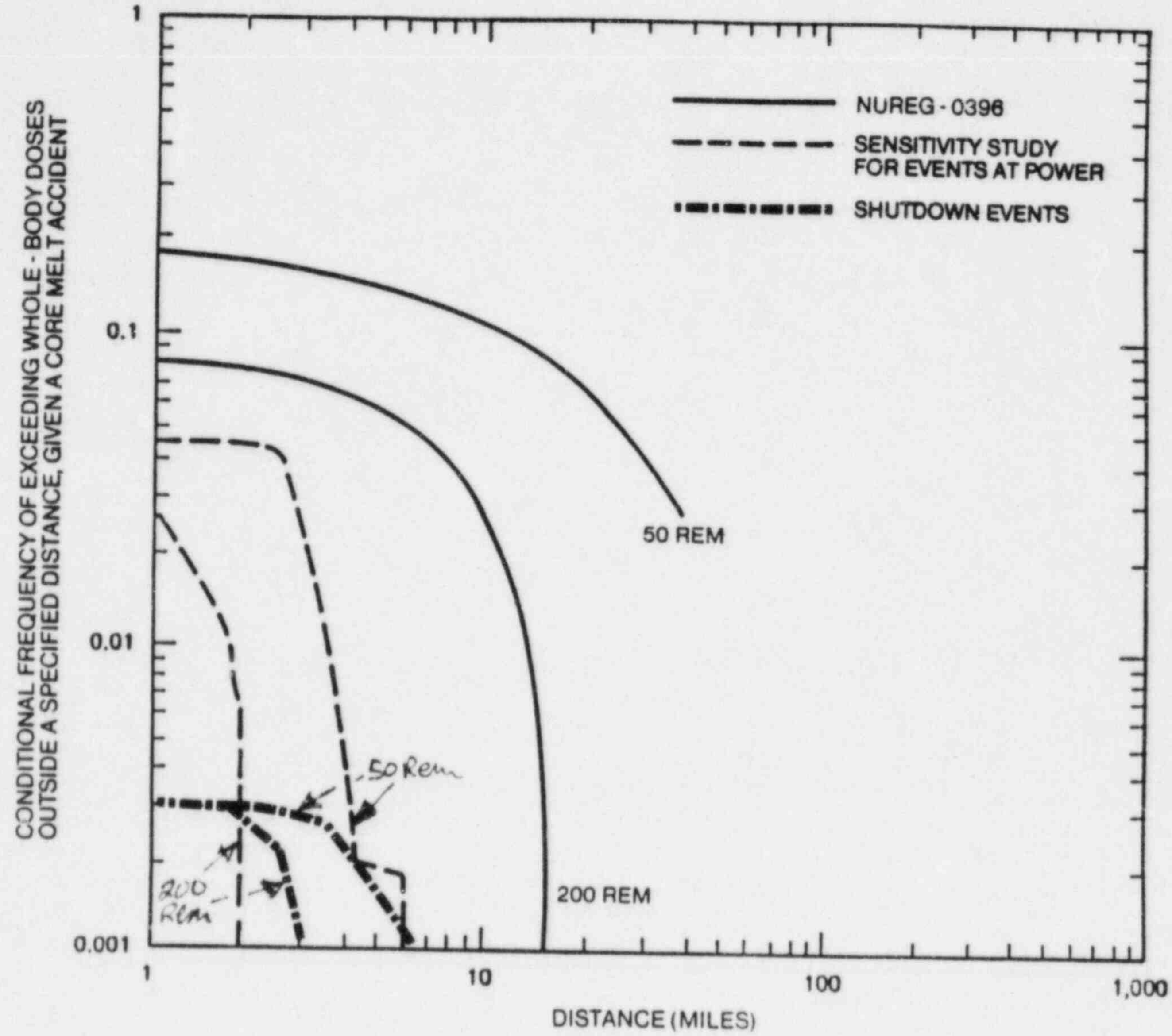


Figure 13 COMPARISON OF 200 REM AND 50 REM DOSE VERSUS DISTANCE CURVES WITH CONTRIBUTIONS FROM SHUTDOWN EVENTS

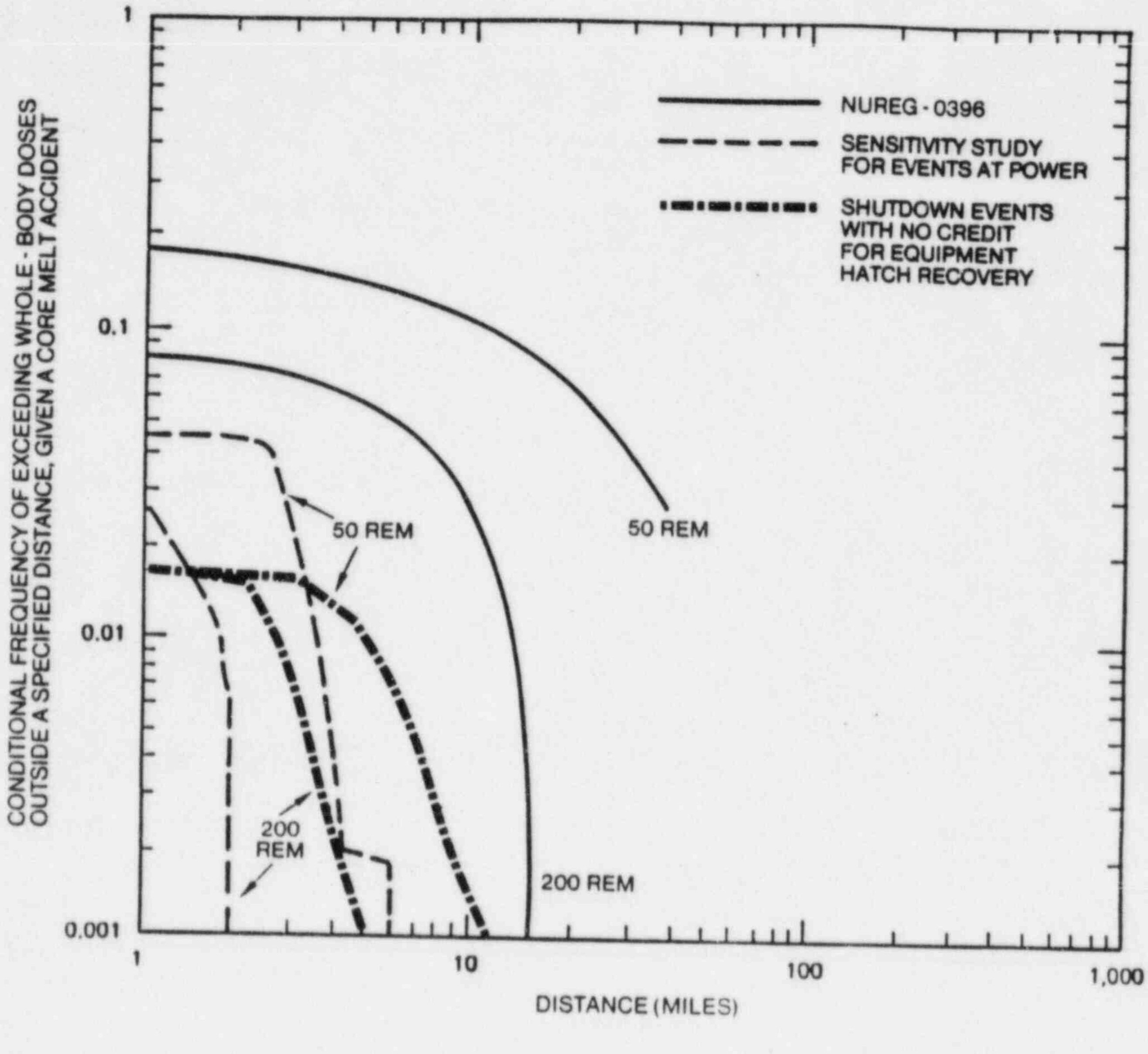


Figure 14 COMPARISON OF 200 REM AND 50 REM DOSE VERSUS DISTANCE CURVES FOR CONSERVATIVE ASSUMPTION OF NO CREDIT FOR OPERATOR RECOVERY OF OPEN EQUIPMENT HATCH

RAI 22

It is the staff's understanding that preexisting violations of containment integrity were "included" in the PSA by assuming the average effect was to raise the containment leak rate to the design basis value of 0.1%/day.

a. Compare this assumption with the containment integrity violation data presented in NUREG/CR 4220.

b. What contributions would these containment integrity violation data make to the probabilities for each of the release categories (assume the S5W category is redistributed over all the appropriate categories by the conditional probabilities of preexisting leakage paths of the size appropriate for each category).

RESPONSE 22

As discussed in Section 11.6.3 of the SSPSA (PLG-0300), release category S5 represents accident sequences where the containment remains intact (does not fail). All containment failures and bypasses, including failure to isolate the containment are included in the other release categories (S1, S2, S3, S4, and S6). Preexisting containment leakage is quantified in S5 by assuming an effective average leak rate equivalent to the containment design leak rate of 0.1% per day. As shown in Section 11.6.4.1, this is a reasonably good approximation when compared to Weinstein's Containment Integrity availability work (reference 11.6-18 of PLG-0300 - also referenced in NUREG/CR-4220). This leakage is also imposed on category S3 but is increased to reflect the predicted higher containment pressures. Weinstein's work found PWR containment integrity to be available 97.3% of the time and an average leak rate of 31 times design basis the rest of the time.

a. NUREG/CR-4220 suggests that PWR containment integrity availability could be as low as 71% with leakage in the range of 1 to 10 times allowable 29% of the time. The probability of larger leaks (28 square inches) is estimated to be in the range of 0.001 to 0.01 with a point estimate of .005. Section 9 of NUREG/CR-4220 states that these are upper bound estimates of containment unavailability. Section 6.1 indicates that the 0.29 unavailability is based on 215 events in 740 reactor years where type B and C leak testing results exceeded 60% of allowable leakage; a leakage duration of one year is simply assumed for each event. Section 6.2 indicates that the .01 to .001 chance of a 28 square inch leak is based on 4 events in 740 reactor years ranging from "small drilled holes" to an open six inch valve; again a one year duration is simply assumed with little, if any, basis. There is insufficient information presented in this document to assess the applicability to Seabrook of each event.

We believe it is highly conservative to assume that 0.29 events per year will occur at Seabrook in which 60% of allowable leakage is exceeded to a level one to ten times allowable leakage and that such conditions will exist 29% of the time. It is also highly conservative to assume that 0.005 events per year ranging "from small drilled holes to an open six inch valve" represents a 0.01 to 0.001 chance of a 28 square inch leak. The reasons why we believe the application of these assumptions to Seabrook to be conservative are as follows. First, it seems extreme to assume that evidence of exceeding Technical Specification limits with LERS will mean that the extent of the leakage is ten times the limit. Also, events involving mispositioning of manual valves are subjected to monthly surveillance testing at Seabrook, therefore, one year is an inappropriate fault duration for Seabrook. Many of the possibilities for mispositioned valves will be covered by automatic containment isolation actuation, whose failures are included in the PSA.

With regard to the four large leakage events noted in sections 4.1.7 and 6.2 of the NUREG, the information provided is insufficient to compare directly with Seabrook; however, it may be that none of these events are applicable to Seabrook (especially for durations of one year) for the following reasons.

1. The containment purge valves at Seabrook are leak tested every six months or less; their position is checked monthly; they are actuated valves which receive containment isolation signals (failure of containment isolation signals/valves is included in the risk models).

2. Seabrook's containment is three to four feet thick; it is difficult to imagine that a hole could inadvertantly be drilled through it, never mind go undetected.

3. At Seabrook, valves, flanges, penetrations, airlocks, etc., are leak tested and position checked after any maintenance activities on them.

4. Manual isolation valves outside containment are position checked every month. All isolation valve positions are checked before return to power operation and at least once per 18 months.

The upper bound frequency of pre-existing leakage derived in NUREG/CR-4220 is greater than that included in the Seabrook PSA; however, they are overly conservative estimates which are inappropriate for use in the Seabrook risk models.

b. Regardless of the above, the effects of applying the NUREG/CR-4220 preexisting leakage estimates to Seabrook was evaluated; it was found that even these upper bound estimates would not result in any of the emergency planning risk criteria to be exceeded.

The effect of an assumed small preexisting leakeage, one to ten times allowable, was evaluated as follows. Assume a preexisting leak of ten times design leak rate 100% of the time. This would increase the source terms for S5 and the first part of S3 by a factor of ten. Since the source terms for the first 24 hours of S3 are greater than S5, S3 will be evaluated. (For the first 24 hours, S3 and S5 are the same leak size but the driving containment pressure is higher for S3.) Using Tables 4-14 of PLG-0432 and 4-4 of PLG-0465, it can be seen that the first 24 hours of the release for S2C clearly envelopes ten times the first 24 hours of S3W in terms of source terms; release timing is also conservatively enveloped. Table D-1 of PLG-0432 shows that S2C is an insignificant contributor to early fatality risk and, in fact, S2C has zero consequences (early fatalities). Table 1, which follows, is a partial reproduction of the no evacuation case risk summary table for the EPZ Sensitivity Study (PLG-0465) which was provided in the response to question 23; however, the S3 and S5 releases have been replaced by S2C. (The S2C source term represents 50 to 100 times the Seabrook maximum design leakage rate and assumes it exists 100% of the time. In other words, Table 1 represents a 100% chance of five to ten times the preexisting "Technical Specification Violations" leakage predicted to occur in the NUREG 30% of the time.) As Table 1 shows, this extremely conservative case still shows zero early fatalities and a small contribution to early injuries.

NUREG/CR-4220 estimates an upper bound of 5E-3 for "large leakage" which is conservatively assumed to be a six inch valve (or 28 square inch hole). We can conservatively bound the risk contribution of this by assigning this frequency to release category S6W.

In Figure 1, we plot the upperbound effects of both small and large leaks in the 200- and 50-rem dose versus distance curves based on the assignment of small leaks to S2-CM and the large leaks to S6W according to the NUREG-4220 probabilities. While we don't agree that these results are reasonable for Seabrook, they show that the NUREG-0396 results at ten miles do not occur at Seabrook at one mile or less. Hence, a conservative interpretation of the NUREG/CR-4220 experience with pre-existing holes has no impact on the conclusions of the sensitivity study.

TABLE 1

FILENAME: PSHSPZ25

SEABROOK EPZ CONSEQUENCE TABULATION   W-1400 SOURCE TERM SENSITIVITY STUDY   SOURCE TERM SEC-M SUBSTITUTED FOR S3W AND S5HAT TO SIMULATE LARGER LEAK RATE

RUN	RUN ID	RELEASE CATEGORY	EVAC DIST	FREQUENCY	PROBABILITY P.C.	PROBABILITY IRAC	EARLY FATALITY	EARLY INJURY	THYROID CANCERS	OTHER CANCERS	TOTAL CANCERS	RUN	REL.CAT	ABSOLUTE RISK FATALITY	ABSOLUTE RISK INJURY	ABSOLUTE RISK CANCER	PERCENTAGE RISK FATALITY	PERCENTAGE RISK INJURY	PERCENTAGE RISK CANCER
NO EVACUATION CASES			W-1400 SOURCE TERMS				----- CONSEQUENCES -------							--------- MEAN RISK ---------			----- PERCENT OF RISK ------		
447	S1W-0	S1W	0	4.0E-09	1	1	745.6	1028.7	4474.2	716.4	940.1	447	S1W	3.0E-06	4.1E-06	3.8E-06	0.1	.0	.0
433	S2W-0	S2W	0	2.1E-05	1	1	122.1	554.7	1115.1	643.3	699.1	433	S2W	2.6E-03	1.2E-02	1.5E-02	84.2	76.8	5.1
139	S2C-M-0	S3W	0	1.4E-04	1	1	0.0	9.6	576.9	1055.9	1084.7	139	S3W	0.0E+00	1.3E-03	1.5E-01	0.0	8.9	53.0
432	S6W-0	S6W	0	6.5E-07	1	1	734.3	1700.1	2681.2	1151.4	1265.5	432	S6W	4.8E-04	1.1E-03	6.4E-04	15.7	7.3	0.2
434	S7W-0	S7W	0	6.3E-08	1	1	0.0	0.4	10.2	13.7	14.2	434	S7W	0.0E+00	2.4E-08	8.9E-07	0.0	.0	.0
139	SEC-M-0	S5HAT	0	1.1E-04	1	1	0.0	9.6	575.9	1033.9	1064.7	139	S5HAT	0.0E+00	1.1E-03	1.2E-01	0.0	7.0	41.6
TOTAL				2.7E-04								TOTAL		3.0E-03	1.5E-02	2.9E-01	100.0	100.0	100.0

IN THE ABOVE SPREAD SHEET FOR THE W-1400 SENSITIVITY STUDY THE CONSEQUENCES OF SOURCE TERM SEC-M-0 FROM THE RMEPS CALCULATIONS WAS SUBSTITUTED FOR THE CONSEQUENCES OF SOURCE TERMS S3W AND S5HAT. THIS SIMULATES A LEAK RATE WHICH IS 50 TO 100 TIMES LARGER THAN THE CONTAINMENT DESIGN BASIS LEAK RATE. EVEN WITH THIS LARGE INCREASE THERE IS NO CHANGE IN THE EARLY FATALITY RISK. THERE IS ABOUT A 15 % INCREASE IN THE EARLY INJURY RISK, HOWEVER THIS IS A SMALL INCREASE COMPARED TO THE INCREASE IN LEAK RATE. THE EARLY INJURY RISK IS STILL CLEARLY DOMINATED BY RELEASE CATEGORY S2W. THIS WOULD MEAN THAT THE IMPACT ON THE 50 REM CURVE AND THE 200 REM CURVE WOULD BE SMALL.

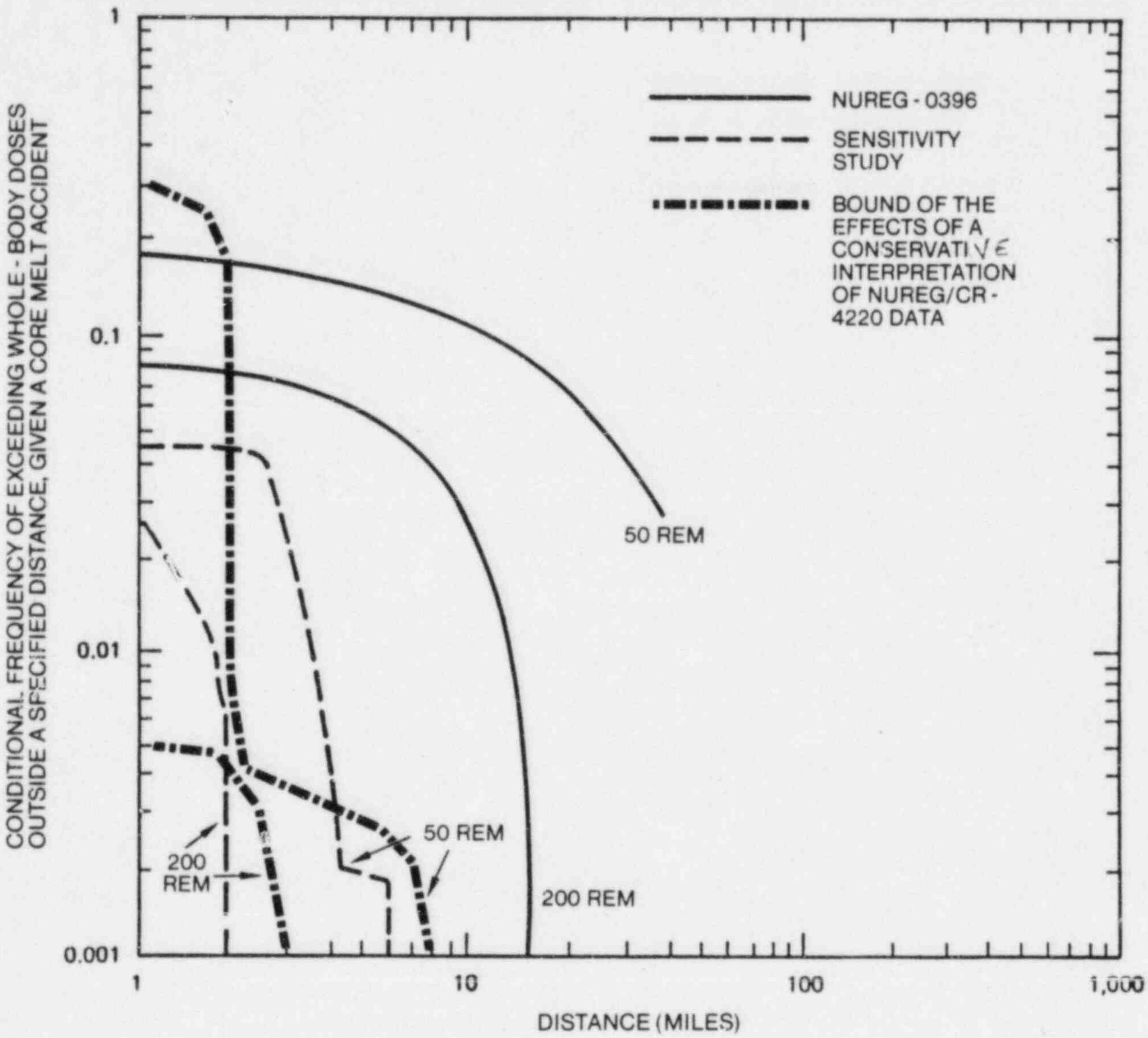


FIGURE 1 BOUNDING THE CONSERVATIVE INTERPRETATION OF NUREG/CR - 4220 DATA
PRE-EXISTING CONTAINMENT LEAKAGE

RAI 23

a. Provide a narrative description that quantitatively delineates the dominant contributors to the dose probability vs distance curves and the early fatality probability curves. The dominant release categories should be specified and the dominant accident sequences contributing to each of these release categories should be specified. The probability of occurence of each release category should be stated. These data should be provided for the current study and for the original PSA results. Changes between the two studies should be attributed to specific differences in the analysis.

b. Provide a set of early fatality conditional probability curves for each release category, assuming evacuation distances of 1 mile and 2 miles.

c. Provide the conditional mean risk of early fatality for each of the curves provided in b.

RESPONSE 23

The following describes the principal contributors to early health risk at Seabrook Station, as determined in the original probabilistic safety assessment (PSA) in 1983 (PLG-0300), in the PSA updates of 1985 (PLG-0432), and in the sensitivity study (PLG-0465). The risk measures of interest here are the early fatality risk curves and the frequency of exceedance of dose and distance curves for the whole body dose of 200 rem. The three (3) parts of this question are addressed collectively.

1. SSPSA RESULTS (1983)

1.1 EARLY FATALITY RISK CURVES

The only results available for early health risk in the Seabrook Station Probabilistic Safety Assessment (SSPSA) (PLG-0300) assume a 10-mile evacuation zone. Significant and costly analysis would be required to produce these results assuming evacuation distances of 1 and 2 miles. The contributors to risk can be expressed in a number of different ways. Alternative ways to group accident sequences are to group the individual sequences by initiating event, by plant damage state, and by release category. A graphic display of how sequences grouped by release category contribute to the mean risk of early fatalities in the original PSA is shown in Figure 1, which is taken from Figure 13.2-1a in PLG-0300. As seen in this figure, release category S6 (large isolation failure) makes a small contribution, and all other categories make negligible contributions that are at frequency levels below 10^{-9} per reactor year. Note that the mean risk curve, whose contributions are being discussed here, is the mean of a family of curves that characterize uncertainty in the risk estimate. This family is shown in Figure 2, which is reproduced from Figure 13.1-5a in PLG-0300. The fact that the mean curve falls well outside the median (.50) risk curve indicates large uncertainties. These uncertainties are due to uncertainties in estimating the accident frequencies, source terms, and site model parameters. The conditional risk curves for each release category can be found in Figure 3.2-2A of the SSPSA. (See page 13.2-78 of PLG-0300.)

A tabular representation of the information in Figure 1 is provided in Table 1, which is adopted from Table 13.2-7a in PLG-0300. This table shows that more than 99% of the mean risk curve comes from release category S6. Most of the remaining contribution comes from S2. Only in the extreme right hand tail, at frequencies below 10^{-9} per reactor year, does another category appear, S1 (early containment rupture due to steam explosion, early overpressure, or external missile).

The next step in breaking down the SSPSA risk contributors is to examine the contribution of sequences grouped by initiating event. Because nearly all of the early health risk comes from S6 and a small contribution from, S2, it is more efficient to confine our search to these release categories. The initiating events that make significant contributions to S6 and S2 are provided in the table below, which was adapted from Table 13.2-5a in PLG-0300. As can be seen, release category S6, which indicates early fatality risk, is, in turn, dominated by the interfacing system loss of coolant accident (LOCA) (V-sequence).

Initiating Event	Percent Contribution	
	S6	S2
V, Interfacing LOCA	76.	0
ET, Seismic Transient	22.	95
EL, Seismic LOCA	2.	5
Others	< 1.	< 1
Total	100.	100.

In a similar fashion, accident sequences can be grouped with respect to plant damage states (sometimes referred to as a bin). The following is the plant damage state breakdown of the S6 and S2 release categories.

Plant Damage State*	Percent Contribution	
	S6	S2
1F	78	0
1FP	0	5
3F	21	0
3FP	0	35
7F	1	0
7FP	0	60
Others	< 1	< 1
Total	100	100

In comparing the previous two tables, note that 100% of the interfacing systems LOCAs were modeled as 1F states. Hence, of the 24% of the seismic contribution to S6, 21% terminated in plant damage state 3F, 2% in 1F, and 1% in 7F. Hence, most of the overall risk contribution comes from the interfacing system LOCA initiator and plant damage state 1F. Essentially, all the remainder are seismically initiated sequences ending in plant damage state 3F. Note that all the FP states, and have more than 99% of the release category S2 and are dominated by similar sequences with the same release path and characteristics, namely Station Blackout with a RCP seal LOCA and failed open seal return line isolation valves. The result is high early leakage and delayed overpressurization of the containment.

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* See Table 1-2 in the Seabrook Station Risk Management and Emergency Planning Study (RMEPS) (PLG-0432) for definitions. Numbers denote containment and reactor coolant system conditions at time of reactor vessel melt-through; letters denote status of containment systems and leak paths. F states are isolation failures or bypasses more than 3 inches in diameter; FP states are isolation failures less than 3 inches in diameter.

The final step in breaking down the early health risk is to examine specific accident sequences. In the SSPSA, an accident sequence is a single path that can be traced through the plant event trees from the point of entry (the initiating event) to the point of termination (the plant damage state). As with other PRAs, the interfacing system LOCA was analyzed as a single sequence. That is, the event was analyzed as an initiating event and assigned directly to the most severe plant damage state considered in the study and denoted as 1F. This reflects the conservative assumption that multiple failures of the interfacing valves automatically result in a core melt and early large containment bypass. All other initiating events were modeled through the plant event trees, which include more than 4.5 billion sequences counting all the initiating events, the plant damage states, and paths connecting them through the plant event trees. Therefore, the above contributions from the V-sequence are from a single sequence, whereas the seismic contributions come from many sequences.

The nature of the specific sequences initiated by seismic events is next described. Of the seismically induced transients that make up 22% of the frequency of release category S6, the single sequence having the greatest frequency makes up only about one-fourth of this contribution and was analyzed as follows.

Event	Frequency
Earthquake Occurs (.3g)	1.1×10^{-4}/year
Offsite Power Does Not Fail	.35
Solid State Protection Fails	.041
Charging Pumps Fail	.88
Containment Is Initially in Purge Mode	.10
Emergency Core Cooling System, Containment Isolation, and Containment Sprays Fail [dependent failures resulting from loss of solid state protection system (SSPS)]	1.0
Other Equipment Does Not Fail	.86
Total	1.2×10^{-7}/year

The remaining three-fourths of the seismic sequences in S6 are made up from a large number of sequences, some involving loss of offsite power and others involving failures of other equipment.

In a similar fashion, the seismic contributors to release category S2 are also spread over many sequences. The single most frequent sequence in this category is a seismically induced loss of offsite power and failure of both diesels due to either seismic causes or independent causes. This sequence appears several times in the scenario identification tables (in Section 13.2 of PLG-0300) once for each discrete range of ground acceleration. The total frequency of this sequence summed over all values of ground acceleration is 6.9×10^{-6} per reactor year, or about 40% of the total release category frequency.

In summary, the early health risk curves in the SSPSA, which were only performed for a 10-mile evacuation zone, were dominated by the interfacing LOCA sequence (about 76% contribution to the mean exceedance frequency in the risk curves). Most of the remaining contributions come from seismically induced sequences with release paths either through the purge lines (the S6 sequences) or through the reactor coolant pump seal return line (the S2 sequences).

1.2 DOSE VERSUS DISTANCE CURVES

The 200 rem and 50 rem dose versus distance curves that correspond with the SSPSA results are compared with the RMEPS and sensitivity study in Figure 3. The SSPSA curves are dominated by release categories S2 and S6.

2. PSA UPDATE RESULTS (RISK MANAGEMENT AND EMERGENCY PLANNING STUDY, 1985)

2.1 EARLY FATALITY RISK CURVES

In the RMEPS update of the Seabrook PSA, the following changes were made that had an impact on the risk levels and the ordering of the risk contributors.

Plant Model Changes

- Item 1. The single sequence interfacing LOCA model was replaced by a two-event tree model, one for suction side and one for injection side residual heat removal/reactor coolant system (RHR/RCS) interfacing valve ruptures. This led to a reduction in the frequency of plant damage state 1F and the addition of three new plant damage states (1FV, 1FPV, and 7 FPV).

  Two plant damage states (1FPV and 7FPV) were added to model new scenarios with a submerged RHR pump seal bypass. This, in turn, led to the introduction of a new release category, S7, which takes credit for decontamination and scrubbing in the source term determination. Plant damage state 1FV contains interfacing LOCAs resulting from unsubmerged piping failures.

- Item 2. A conservatism in the treatment of certain seismically initiated sequences in release category S6 (plant damage states 1F, 3F, and 7F) was eliminated. In the updated results, credit was taken for loss of instrument air to the air-operated valves (AOVs) in the purge lines on loss of offsite power; hence, a high probability of purge isolation valve closures in these instances. This resulted in a shift in some of the frequency of release category S6 to S2 because, when the large purge valves are assumed to close, there remain small open lines with motor-operated valves that fail in these same sequences. There still remain some seismic sequences in S6 with the purge isolation failure. Those that remain either involve a no loss of offsite power condition or mechanical failure of the purge valves.

- Item 3. In support of the effort to optimize plant technical specifications (PLG-0431), the PSA systems modeled were revised to incorporate revision to the technical specifications and a more complete treatment of common cause failures. This led to many minor changes to individual sequence frequencies, with the most significant change being an increase to the unavailability of the primary component cooling system. This led to a slight increase in core melt frequency from 2.3×10^{-4} to 2.7×10^{-4}, most of which occurred in plant damage state 8D.

- Item 4. The updated results take credit for recovery of certain containment systems (principally the containment building spray) during core melt scenarios initiated by loss of offsite power and involving a station blackout. Both the original and updated results took credit for recovery of electric power prior to and in prevention of core melt. This new recovery action results in a small shift in frequency from release category S3 (gradual containment overpressure) to S5 (containment intact). This change does not appreciably affect the results of the RMEPS or sensitivity studies since neither S3 nor S5 contributes to early health risk with at least 1 mile of evacuation.

Containment Model Changes

- Item 5. Uncertainties in source terms were reasssessed for all release categories with the net effect of a reduction in the mean source terms for all categories.

- Item 6. A new release category and three new plant states for interfacing system LOCA scenarios were added.

- Item 7. Interfacing system LOCAs resulting in unsubmerged RHR piping failures (plant state 1FV) were reassigned from release category S6 to S1 (small conservative effect).

Site Model Changes

- Item 8. Site model uncertainties were reassessed (minor effect).

- Item 9. The evacuation distance and sheltering assumptions were varied.

- Item 10. The Unit 2 construction workers were eliminated from the population distribution.

Although all the above changes contribute in some way to differences in the updated results, the ones that had the most significant impact on early health risk are items 1, 5, 6, and 9. A more quantitative picture of the significance of each change is provided below.

The results in the RMEPS update for early fatality risk are presented in Tables 2, 3, and 4 for evacuation cases of no evacuation, 1-mile evacuation, and 2-mile evacuation, respectively. These are the comparison tables for Table 1 and the original SSPSA results. The conditional risk curves for each release category and evacuation distance can be found in Appendix C of PLG-0432. There are two kinds of differences exhibited in the new tables. One is that the risk levels (exceedance frequency values) are lower although less evacuation is assumed, and, as expected, the levels decrease as the evacuation zone is increased from 0 to 2 miles. The other difference is that several new release categories, in addition to S6 and S2, appear as making significant contributions: S3, S7, and S1. Release category S3 contributes only under the assumption of no evacuation. This result is viewed as purely academic because the time of release for S3 is some 89 hours after the initiating event, during which even ad hoc protective actions would be affective.

For 1 or 2-mile evacuation, release categories S2, S6, S7, and S1 are significant. The contribution of S2 only appears in the low consequence, relatively high frequency portions of the risk curve. In comparison of these results with Table 1, the shift in the ranking of contributors is due to the following.

1. The frequency of S6 in RMEPS is lower because of the deletion of the interfacing LOCA and some seismically initiated sequences with station blackout.

2. The frequency of S2 increased slightly from the same seismic sequence noted in 1.

3. Some of the old V-sequence frequency formerly categorized in S6 is now in S7. While source terms in S7 are lower than S6, they are still great enough for potentially fatal doses.

4. The frequency of S1 increased due to the addition of the pipe break type V-sequences formerly categorized in S6 and to a smaller extent by a reassessment of some turbine missile scenarios that was done since the RMEPS.

The contributions of plant damage states and initiating events to all updated release categories are shown in Tables 5 and 6, respectively. Tables 7 and 8 define the codes used for initiating events. About 78% of the scenarios in category S1 are pipe break type interfacing LOCAs that are assigned to plant damage state 1FV. The remaining scenarios in S1 include aircraft and turbine missile scenarios that fail the containment in plant damage states 1FA, 2FA, and 6FA and a wide spectrum of transient and LOCA scenarios with containment failure due to reactor vessel steam explosions. The contributors to category S2 are the same as those in the SSPSA; namely, seismically induced station blackout with a failed open small penetration. Release category S6 is now dominated by seismically induced accident sequences with no loss of offsite power and failure of the SSPS system with an assumed containment purge in progress. No credit for operator recovery of any system or component is taken for any seismic sequence, including those that now dominate S2 and S6.

Release category S7 is composed wholly of new interfacing LOCA scenarios in which the RHR piping remains intact and the bypass occurs via a degraded and submerged RHR pump seal. In assessing the uncertainty on the source term for S7, a 10% probability was assigned to the possibilitly that the leak path would not be submerged. From the information provided in RMEPS, it is clear that there would be no contribution to early health risk from S7 if only best estimate (submerged) source terms had been used. Similarly, had the conservative source terms not been used for the remaining release categories, the risk levels calculated in RMEPS would have been much lower than they were. In fact, on the basis of using the best estimate source terms only, release category S1 is the only category that produced any potential for 200-rem doses and, hence, any potential for early fatalities.

2.2 DOSE VERSUS DISTANCE CURVE

In the RMEPS results, there was found to be very little potential for 200-rem doses, even close to the site. As seen in Figure 2-9 of RMEPS

(PLG-0432), the frequency of exceedance scale had to be extended from .001 to .0001 to pick up the mean risk of exceeding the 200-rem dose shown on the curve. The median curve for the 200-rem dose was off-scale. The contributions to the mean risk at various distances are indicated in the table below.

Release Category	Percent Contribution to 200-Rem Exceedance Frequency (Figure 2-9 in PLG-0432)		
	1 mile	1.5 miles	2 miles
S1	1	3	6
S2	13	34	0
S3	67	3	0
S5	0	0	0
S6	17	50	73
S7	2	9	20

By comparing these results with those in Table 2, it is seen that the 200-rem risk has the same set of release category contributors as the early fatality risk curve for no evacuation. Category S3 dominates at 1 mile. Again, this result is largely academic. It is difficult to envision, even if no emergency plans existed, that any individual would be in a position to receive a large dose more than 3 days after the initiating event. At 2 miles, the 200-rem curve is dominated by S6, with smaller contributions by S7 and S1. Hence, the overall picture of the risk contributors is the same for the early fatality risk curves and the 200-rem dose versus distance curves.

2.3 CONDITIONAL MEAN RISK

Appendix D of PLG-0432 provides mean risk summary tables for early fatality risk as well as cancer risk.

3. SENSITIVITY STUDY UPDATE

In the Seabrook Station Emergency Planning Sensitivity Study (PLG-0465, 1986), there were no changes made to the plant model. Source terms were revised to reflect the Reactor Safety Study (WASH-1400) source term methodology; i.e., were calculated using the CORRAL computer program for the Seabrook plant configuration. These CORRAL source terms had been developed during the original SSPSA. CRACIT computer program runs were made using best estimate (median) modeling assumptions, and median accident frequencies were used for consistency with NUREG-0396 and WASH-1400. However, unlike NUREG-0396 and WASH-1400, the full treatment of dependent and external events in the Seabrook results was left unchanged.

3.1 EARLY FATILITY RISK CURVES

The early fatality risk curves for 0, 1 and 2-mile evacuations are plotted in Figure 2-1 of PLG-0465. The conditional early fatality risk curves for each release category are found in Appendix B of this report. The contributors by release category are shown in the tables below for no evacuation, 1-mile evacuation, and 2-mile evacuation.

RESULTS FOR NO EVACUATION

Release Category	Percent Contribution to Early Fatality Risk Curve		
	1 Fatality	100 Fatalities	1,000 Fatalities
S1	< 1	< 1	< 1
S2	100	99	99
S6	< 1	1	1
Others	< 1	< 1	< 1
Total	100	100	100

RESULTS FOR 1-MILE EVACUATION

Release Category	Percent Contribution to Early Fatality Risk Curve		
	1 Fatality	100 Fatalities	1,000 Fatalities
S1	< 1	< 1	< 4
S2	99	95	0
S6	1	5	96
Others	< 1	< 1	< 1
Total	100	100	100

RESULTS FOR 2-MILE EVACUATION			
Release Category	Percent Contribution to Early Fatality Risk Curve		
	1 Fatality	100 Fatalities	1,000 Fatalities
S1	2	2	*
S2	0	0	*
S6	98	96	8
Others	< 1	< 1	*
Total	100	100	100

* Results below 10^{-9} per reactor year not shown.

By comparing these results with the RMEPS results, it can be seen that one chief difference is that S2 now has a more dominating impact than it did in RMEPS, especially for the 0 and 1-mile evacuation cases. Category S6 dominates the low frequency tail of the 1-mile curve and completely dominates the 2-mile results. The other chief difference is that categories S3 and S7 no longer make a significant contribution to early health risk and the percent of contribution of S1 is reduced somewhat. These differences stem from the fact that application of the WASH-1400 source term methodology did not have uniform impact on all the source terms. The application of this methodology appears to have increased the S2 source term more than the others. In addition, the RMEPS results for early health risk are heavily influenced by the conservative source terms used in that study. For category S7, the conservative RMEPS source term assumed no credit for a flooded RHR vault, while such credit was taken in the sensitivity study to make the analysis consistent with WASH-1400. In WASH-1400, credit was taken for suppression pool scrubbing in some boiling water scenarios.

In Figure 3, the 200-rem and 50-rem dose versus distance curves are compared between NUREG-0396, the Sensitivity Study, RMEPS, and the original SSPSA. The 200-rem curves for the latter two studies are off scale. As can be seen from this figure, the Sensitivity Study results fully bound the RMEPS and SSPSA results.

3.2 DOSE VERSUS DISTANCE CURVES

The 200-rem dose versus distance curve is fully dominated by release category S2, with very small contributions from S6 and S1. The contributions of S6 and S1 occur below the level of conditional core melt frequency at which the curves are cut off in NUREG-0396 (.001).

3.3 CONDITIONAL MEAN RISK

Table 9 provides a mean risk summary table for the sensitivity study results. Column number 5 provides the information requested in Part C of this question. The corresponding information for RMEPS can be found in Appendix D of RMEPS. There are no results in the SSPSA that assume either a 1-mile or 2-mile evacuation. The results presented in Table 9 confirm that release categories S2W and S6W completely dominate the risk at Seabrook Station if the WASH-1400 source term methodology is used to define source terms. The only additional small contribution to risk is made by release category S1W for the early fatality risk with a 2-mile evacuation distance.

Figures 4 and 5 show the decrease in the early fatality risk as a function of evacuation distance, comparing the mean risk results from the RMEPS study (PLG-0432) and the EPZ sensitivity study (PLG-0465), which used WASH-1400 based source terms. Figure 4 compares the risk reduction for the two cases on an absolute basis, and, in Figure 5, the risk reduction is normalized to the no-evacuation case for each study. The results indicate that, for the WASH-1400 source terms (PLG-0465), the acute fatality risk decreases even more rapidly in the first 2 miles than for the RMEPS baseline case. Furthermore, the risk for all WASH-1400 source term cases remains below the safety goal risk.

TABLE 1. CONTRIBUTIONS OF RELEASE CATEGORIES TO RISK OF EARLY FATALITIES AS CALCULATED IN SSPSA

	Number of Early Fatalities (percent contribution of release category)				
	1	10	100	1,000	10,000
	S6 (98.98) S2 (0.92) Others (< .1)	S6 (98.8) S2 (1.10) Others (< .2)	S6 (99.4) S2 (0.52) Others (< .1)	S6 (99.4) S2 (0.49) Others (< .1)	S6 (99.5) S1 (0.5) Others (0)
Frequency of Exceedance	4.60-7	3.87-7	3.14-7	1.78-7	6.26-10

NOTE: Exponential notation is indicated in abbreviated form; i.e., 4.60-7 = 4.60 x 10^{-7}.

TABLE 2. CONTRIBUTIONS OF RELEASE CATEGORIES TO RISK OF EARLY FATALITIES BASED ON RMEPS UPDATE - NO EVACUATION CASE (E0)

	Number of Early Fatalities (percent contribution of release category)				
	1	10	100	1,000	10,000
	S2 (40.1) S3 (35.4) S6 (18.3) Others (< 7)	S3 (62.7) S6 (27.6) S7 (6.7) Others (< 3)	S3 (50.0) S6 (32.7) S7 (13.3) Others (< 4)	S6 (67.6) S7 (24.9) S1 (7.5) Others (= 0.0)	S7 (98.5) S1 (1.5) Others (= 0.0)
Updated Frequency of Exceedance	1.40-7	7.91-8	2.98-8	4.41-9	2.53-11
SSPSA Results	4.60-7	3.87-7	3.14-7	1.78-7	6.26-10

NOTE: Exponential notation is indicated in abbreviated form; i.e., 1.40-7 = 1.40 x 10^{-7}.

TABLE 3. CONTRIBUTIONS OF RELEASE CATEGORIES TO RISK OF EARLY FATALITIES BASED ON RMEPS UPDATE - 1-MILE EVACUATION ZONE (E1)

	Number of Early Fatalities (percent contribution of release category)				
	1	10	100	1,000	10,000
	S2 (71.7) S6 (18.6) S7 (7.4) Others (< 3)	S6 (65.5) S7 (26.4) S1 (8.1) Others (= 0.0)	S6 (50.7) S7 (41.2) S1 (8.1) Others (= 0.0)	S7 (62.4) S6 (28.3) S1 (9.3) Others (= 0)	S7 (98.5) S1 (1.5) Others (= 0)
Updated Frequency of Exceedance	7.44-8	1.64-8	8.25-9	1.59-9	2.53-11
SSPSA Results	4.60-7	3.87-7	3.14-7	1.78-7	6.26-10

NOTE: Exponential notation is indicated in abbreviated form; i.e., 7.44-8 = 7.44×10^{-8}.

TABLE 4. CONTRIBUTIONS OF RELEASE CATEGORIES TO RISK OF EARLY FATALITIES BASED ON RMEPS UPDATE - 2-MILE EVACUATION ZONE (E2)

	Number of Early Fatalities (percent contribution of release category)				
	1	10	100	1,000	10,000
	S2 (46.0) S7 (43.3) S1 (10.7) Others (= 0)	S7 (55.2) S6 (33.3) S1 (11.5) Others (= 0)	S7 (65.3) S6 (23.9) S1 (10.8) Others (= 0)	S6 (72.9) S7 (19.1) S1 (8.0) Others (= 0)	
Updated Frequency of Exceedance	4.96-9	3.07-9	1.17-9	2.28-10	0.0
SSPSA Results	4.60-7	3.87-7	3.14-7	1.78-7	6.26-10

NOTE: Exponential notation is indicated in abbreviated form; i.e., 4.96-9 = 4.96×10^{-9}.

TABLE 5. CONTRIBUTIONS OF PLANT DAMAGE STATES TO RELEASE CATEGORIES MAKING MAJOR CONTRIBUTIONS TO RISK OF CORE MELT FREQUENCY - RMEPS UPDATE RESULTS

	Major Risk and Core Melt Frequency Contributing Release Categories (percent contribution of plant damage states to release categories)					
	S1	S2	S3	S5	S6	S7
	1FV (77.5) 1FA (7.3) 8A (5.0) 2FA (3.7) 6FA (1.3) Others (5.2)	7FP (49.5) 3FP (43.8) 1FP (6.6) Others (< 1)	8D (72.0) FD (14.6) 3D (11.3) Others (< 3)	8A (82.1) 4A (15.0) 2A (1.6) Others (< 2)	3F (92.5) 7F (7.3) Others (< 1)	1FPV (69.5) 7FPV (31.5) Others (0.0)
Release Category Frequency	6.00-9	2.02-5	1.43-4	1.17-4	3.00-7	3.93-8

NOTE: Exponential notation is indicated in abbreviated form; i.e., 6.00-9 = 6.00×10^{-9}.

TABLE 6. CONTRIBUTIONS OF ACCIDENT SEQUENCES GROUPED BY INITIATING EVENT TO PLANT DAMAGE STATES WITH MAJOR RISK OR CORE MELT FREQUENCY CONTRIBUTIONS - RMEPS UPDATE RESULTS

	Plant Damage States (percent contribution by initiating event)												
	1FV	1FPV	1FA	2FA	7FPV	3D	3F	3FP	4A	7D	7FP	8A	8D
	VI (58.1)	VS (100)	TMLL (86.0)	TMLL (86.0)	VS (76.3)	ALOMF (49.1)	E.3A (56.3)	E.7T (55.3)	IMSIV (23.7)	LOSP (39.2)	E.7T (31.4)	LOSP (53.0)	RT (21.0)
	VS (41.9)		APC (14.0)	APC (14.0)	VI (23.7)	LOSP (14.8)	E.7A (16.0)	E.5T (14.5)	LIDC (23.1)	EXTLP (29.5)	E.4T (18.3)	SLOCA (18.4)	TLPCC (17.0)
						E.3A (8.9)	E.4A (8.9)	E1.0T (10.3)	LOSP (12.1)	EXTAC (11.5)	E.5T (17.6)	RT (7.1)	PLMFW (16.8)
						E.7T (6.0)	E.5A (6.1)	E.4T ([illegible])	ATT (11.3)	SGTR (4.1)	E.3T (14.3)	SLBI (6.8)	TT (13.2)
						E.7A (3.1)	E1.0A (4.5)	E.7A (5.2)	ALOMF (7.1)	TLMFW (4.1)	E.2T (12.9)	TT (4.2)	EXFW (9.2)
								E1.0A (3.5)	E.4T (3.7)	EXTCR (2.3)		EXFW (2.9)	TLSW (6.0)
									E.5T (3.0)			PLMFW (2.8)	LOSP (4.8)
									E.7T (2.8)				LOPF (3.7)
									E.3T (2.6)				SLOCA (3.7)
									SLOCA (2.0)				
						Others < 20	Others < 9	Others < 5	Others < 8	Others < 10	Others < 6	Others < 5	Others < 5
Plant Damage State Frequency (events per year)	4.65-9	2.73-8	4.36-10	2.21-10	1.20-8	1.70-5	2.77-7	8.84-6	1.77-5	2.20-5	1.00-5	9.66-5	1.03-4

NOTES:

1. Initiating event codes are defined in Tables 7 and 8.
2. Exponential notation is indicated in abbreviated form; i.e., 4.65-9 = 4.65×10^{-9}.

TABLE 7. BINNING OF INITIATING EVENTS THAT HAVE IDENTICAL IMPACTS

New Initiating Events		Binned SSPSA Initiating Events	
Title	Frequency (events/year)	Title	Frequency (events/year)
EXTAC	2.70-6	FSRAC	5.19-7
		FCRAC	2.10-6
		FL2SG	8.50-8
EXTLP	1.20-3	FTBLP	6.00-4
		FLLP	3.20-4
		TCTL	2.76-4
EXTCR	5.43-7	TMCR	3.98-7
		MCR	5.80-9
		ACR	1.39-7
TLPCC	1.82-5	LPCC	1.39-6
		FSRCC	3.60-6
		FCRCC	9.00-6
		FPCC	4.20-6
		TMPCC	1.27-8
		MPCC	5.46-9
TLSW	6.22-6	LOSW	2.52-6
		FCRSW	2.10-6
		FLSW	1.60-6
TLCV	4.18-1	LCV	4.18-1
		TMLCV	8.30-5

NOTE: Exponential notation is indicated in abbreviated form; i.e., 2.70-6 = 2.70×10^{-6}.

TABLE 8. INITIATING EVENT CATEGORIES SELECTED FOR QUANTIFICATION OF THE SEABROOK STATION RISK MODEL FOR THE SSPSA

Sheet 1 of 2

Group	Initiating Event Categories Selected for Separate Quantification	Code Designator
• Loss of Coolant Inventory	1. Excessive LOCA	ELOCA
	2. Large LOCA	LLOCA
	3. Medium LOCA	MLOCA
	4. Small LOCA	SLOCA(a)
	5. Interfacing Systems LOCA	V
	6. Steam Generator Tube Rupture	SGTR(a)
• General Transients	7. Reactor Trip	RT
	8. Turbine Trip	TT(b)
	9. Total Loss of Main Feedwater	TLMFW(c)
	10. Partial Loss of Main Feedwater	PLMFW(c)
	11. Excessive Feedwater Flow	EXFW(b)
	12. Loss of Condenser Vacuum	LCV(b)
	13. Closure of One Main Steam Isolation Valve (MSIV)	1MSIV(b)
	14. Closure of All MSIVs	AMSIV
	15. Core Power Excursion	CPEXC
	16. Loss of Primary Flow	LOPF(b)
	17. Steam Line Break Inside Containment	SLBI
	18. Steam Line Break Outside Containment	SLBO
	19. Main Steam Relief Valve Opening	MSRV
	20. Inadvertent Safety Injection	SI
• Common Cause Initiating Events		
- Support System Faults	21. Loss of Offsite Power	LOSP(d)
	22. Loss of One DC Bus	L1DC
	23. Total Loss of Service Water	LOSW
	24. Total Loss of Component Cooling Water	LPCC
- Seismic Events	25. 0.7g Seismic LOCA	E.7L
	26. 1.0g Seismic LOCA	E1.0L
	27. 0.2g Seismic Loss of Offsite Power	E.2T(e)
	28. 0.3g Seismic Loss of Offsite Power	E.3T(e)
	29. 0.4g Seismic Loss of Offsite Power	E.4T(e)

a. Transient without scram scenarios are represented by a separate code, ASLOC.
b. Transient without scram scenarios are represented by a separate code, ATT.
c. Transient without scram scenarios are represented by a separate code, ALOMF.
d. Transient without scram scenarios are represented by a separate code, ALOSP.
e. Transient without scram scenarios are represented by a separate code, ExA, x = .2, .3, .4, .5, .7, 1.0.

TABLE 8 (continued)

Sheet 2 of 2

Group	Initiating Event Categories Selected for Separate Quantification	Code Designator
	30. 0.5g Seismic Loss of Offsite Power	E.5T(e)
	31. 0.7g Seismic Loss of Offsite Power	E.7T(e)
	32. 1.0g Seismic Loss of Offsite Power	E1.0T(e)
- Fires	33. Cable Spreading Room - PCC Loss	FSRCC
	34. Cable Spreading Room - AC Power Loss	FSRAC
	35. Control Room - PCC Loss	FCRCC
	36. Control Room - Service Water Loss	FCRSW
	37. Control Room - AC Power Loss	FCRAC
	38. Electrical Tunnel 1	FET1
	39. ELectrical Tunnel 3	FET3
	40. PCC Area	FPCC
	41. Turbine Building - Loss of Offsite Power	FTBLP
- Turbine Missile	42. Steam Line Break	TMSLB
	43. Large LOCA	TMLL
	44. Loss of Condenser Vacuum	TMLCV
	45. Control Room Impact	TMCR
	46. Condensate Storage Tank Impact	TMCST
	47. Loss of PCC	TMPCC
- Tornado Missile	48. Loss of Offsite Power and One Diesel Generator	MELF
	49. Loss of PCC	MPCC
	50. Control Room Impact	MCR
- Aircraft Crash	51. Containment Impact	APC
	52. Control Room Impact	ACR
	53. Primary Auxiliary Building Impact	APAB
- Flooding	54. Loss of Offsite Power	FLLP
	55. Loss of Offsite Power and One Switchgear Room	FL1SG
	56. Loss of Offsite Power and Two Switchgear Rooms	FL2SG
	57. Loss of Offsite Power and Service Water Pumps	FLSW
- Others	58. Truck Crash into Transmission Lines	TCTL

a. Transient without scram scenarios are represented by a separate code, ASLOC.
b. Transient without scram scenarios are represented by a separate code, ATT.
c. Transient without scram scenarios are represented by a separate code, ALOMF.
d. Transient without scram scenarios are represented by a separate code, ALOSP.
e. Transient without scram scenarios are represented by a separate code, ExA, x = .2, .3, .4, .5, .7, 1.0.

TABLE 9. RISK SUMMARY AND CONTRIBUTION FROM EACH RELEASE CATEGORY FOR THE SEABROOK STATION EMERGENCY PLANNING SENSITIVITY STUDY (PLG-0465) FOR 0, 1, AND 2-MILE EVACUATION CASES

RUN	RELEASE CATEGORY	EVAC DIST	FREQUENCY	EARLY FATALITY	TOTAL CANCERS	ABSOLUTE RISK FATALITY	ABSOLUTE RISK CANCER	PERCENTAGE RISK FATALITY	PERCENTAGE RISK CANCER
NO EVACUATION CASES				--- CONSEQUENCES ---		---- MEAN RISK ----		- PERCENT OF RISK -	
447	S1W	0	4.0E-09	745.6	940.1	3.0E-06	3.8E-06	0.1	.0
453	S2W	0	2.1E-05	122.1	699.1	2.6E-03	1.5E-02	84.2	93.9
455	S3W	0	1.4E-04	0.0	0.7	0.0E+00	9.3E-05	0.0	0.6
452	S6W	0	6.5E-07	734.3	1285.5	4.8E-04	8.4E-04	15.7	5.3
454	S7W	0	6.3E-08	0.0	14.2	0.0E+00	8.9E-07	0.0	.0
	S5HAT	0	1.1E-04	0.0	0.1	0.0E+00	1.5E-05	0.0	0.1
TOTAL			2.7E-04			3.0E-03	1.6E-02	100.0	100.0
ONE MILE EVACUATION CASES				--- CONSEQUENCES ---		---- MEAN RISK ----		- PERCENT OF RISK -	
447	S1W	1	4.0E-09	745.6	940.1	3.0E-06	3.8E-06	0.5	.0
456	S2W	1	2.1E-05	9.4	659.8	2.0E-04	1.4E-02	35.7	93.7
455	S3W	1	1.4E-04	0.0	0.7	0.0E+00	9.3E-05	0.0	0.6
457	S6W	1	6.5E-07	541.9	1264.9	3.5E-04	8.2E-04	63.8	5.6
454	S7W	1	6.3E-08	0.0	14.2	0.0E+00	8.9E-07	0.0	.0
	S5HAT	1	1.1E-04	0.0	0.1	0.0E+00	1.5E-05	0.0	0.1
TOTAL			2.7E-04			5.5E-04	1.5E-02	100.0	100.0
TWO MILE EVACUATION CASES				--- CONSEQUENCES ---		---- MEAN RISK ----		- PERCENT OF RISK -	
447	S1W	2	4.0E-09	745.6	940.1	3.0E-06	3.8E-06	6.8	.0
459	S2W	2	2.1E-05	0.0	440.4	0.0E+00	9.2E-03	0.0	92.5
455	S3W	2	1.4E-04	0.0	0.7	0.0E+00	9.3E-05	0.0	0.9
458	S6W	2	6.5E-07	63.3	986.5	4.1E-05	6.4E-04	93.2	6.4
454	S7W	2	6.3E-08	0.0	14.2	0.0E+00	8.9E-07	0.0	.0
	S5HAT	0	1.1E-04	0.0	0.1	0.0E+00	1.5E-05	0.0	0.1
TOTAL			2.7E-04			4.4E-05	1.0E-02	100.0	100.0

TOTAL RISK BY EVACUATION DISTANCE

	ABSOLUTE RISK FATALITY	ABSOLUTE RISK CANCER	PERCENTAGE RISK FATALITY	PERCENTAGE RISK CANCER
NO EVAC	3.0E-03	1.6E-02	100.0	100.0
1 MI EVAC	5.5E-04	1.5E-02	18.1	94.6
2 MI EVAC	4.4E-05	1.0E-02	1.4	64.0

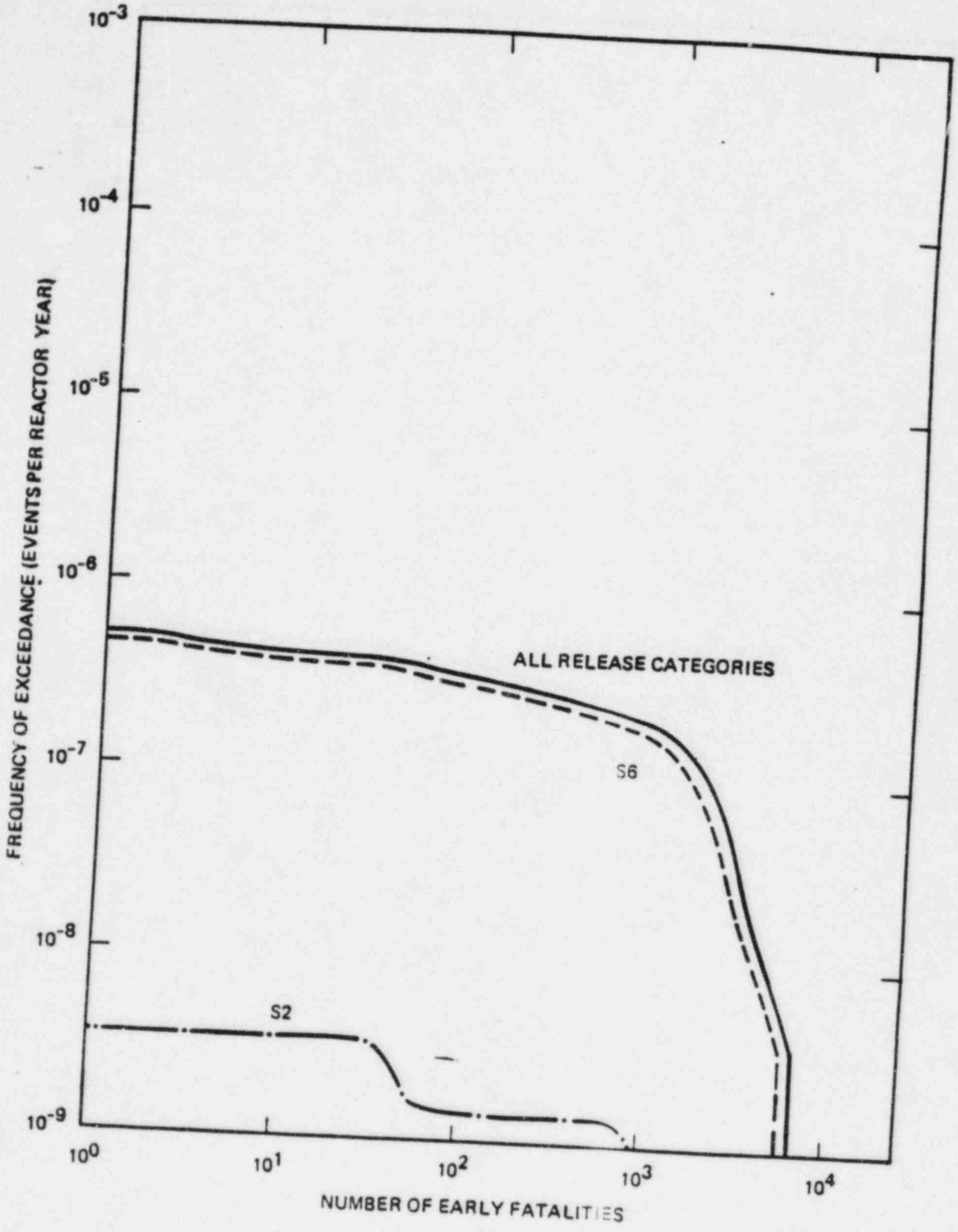


FIGURE 1. CONTRIBUTION OF RELEASE CATEGORIES TO RISK OF EARLY FATALITIES (MEAN VALUES) AS CALCULATED IN SSPSA (1983)

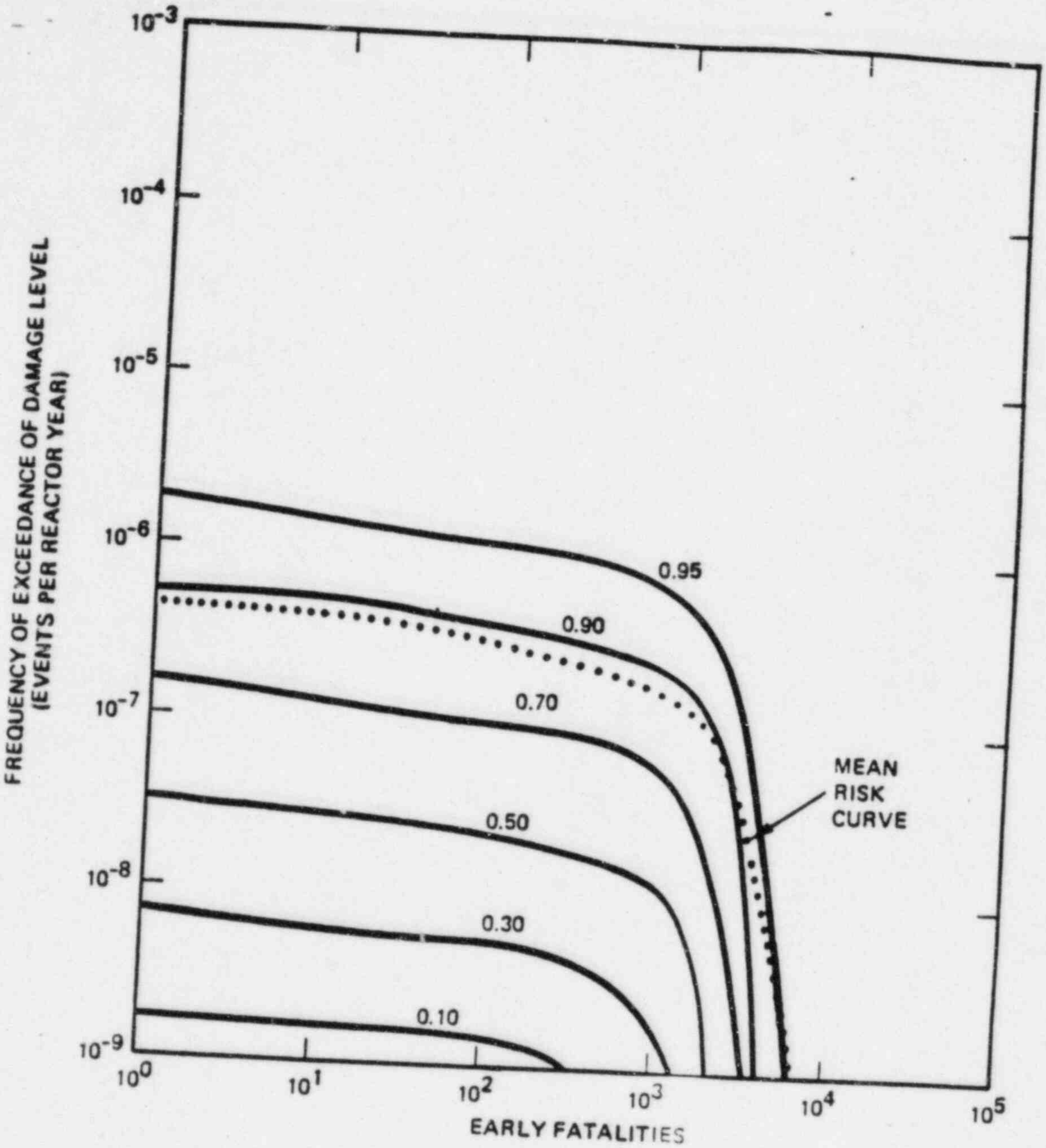


FIGURE 2. RISK OF EARLY FATALITIES WITH UNCERTAINTIES FROM SSPSA (PLG-0300, 1983)

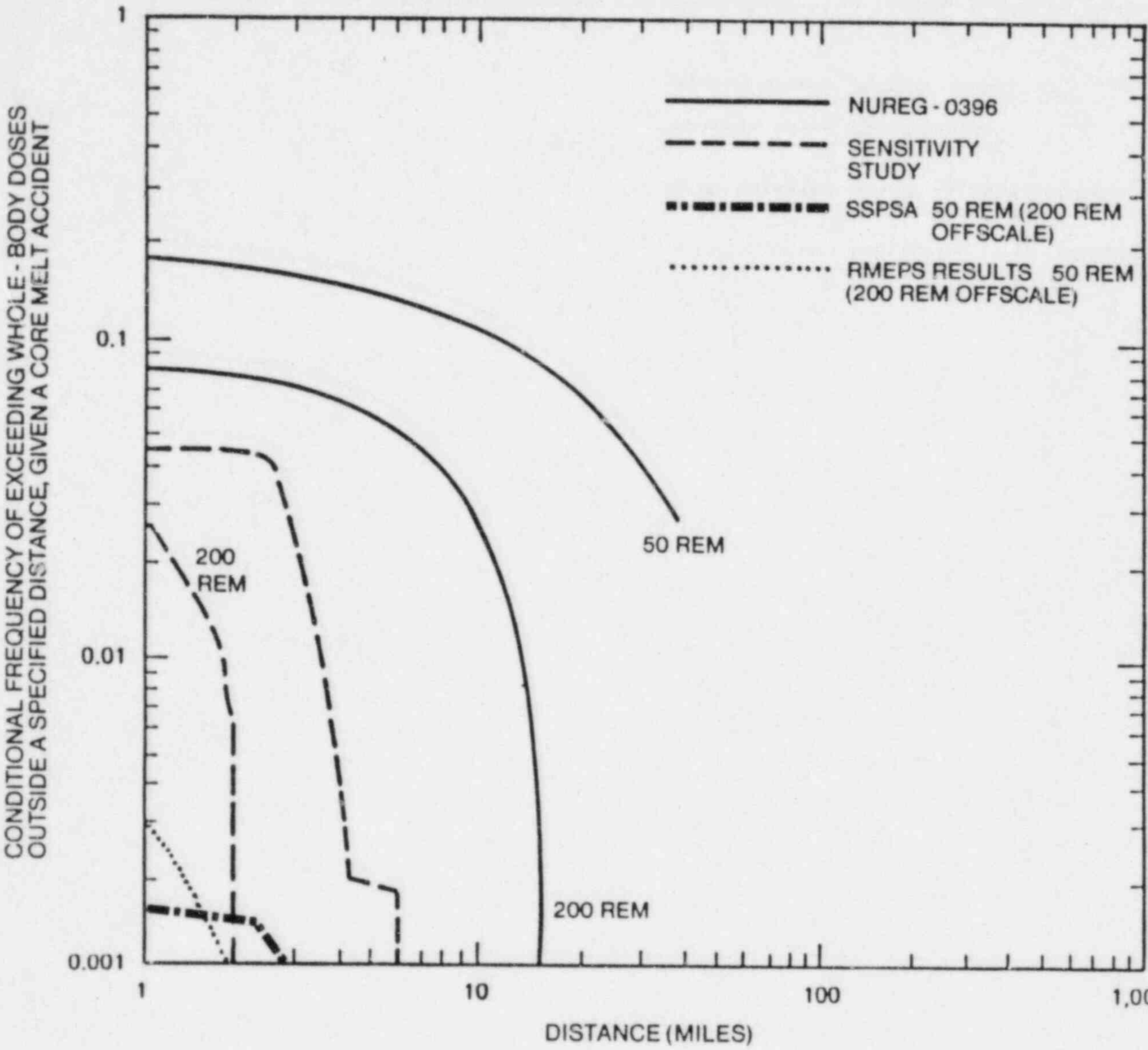


FIGURE 3 COMPARSION OF SSPSA, SENSITIVITY STUDY, AND RMEPS WITH NUREG-0396 — 200 REM AND 50-REM WHOLE BODY DOSE PLOTS FOR NO IMMEDIATE PROTECTIVE ACTIONS

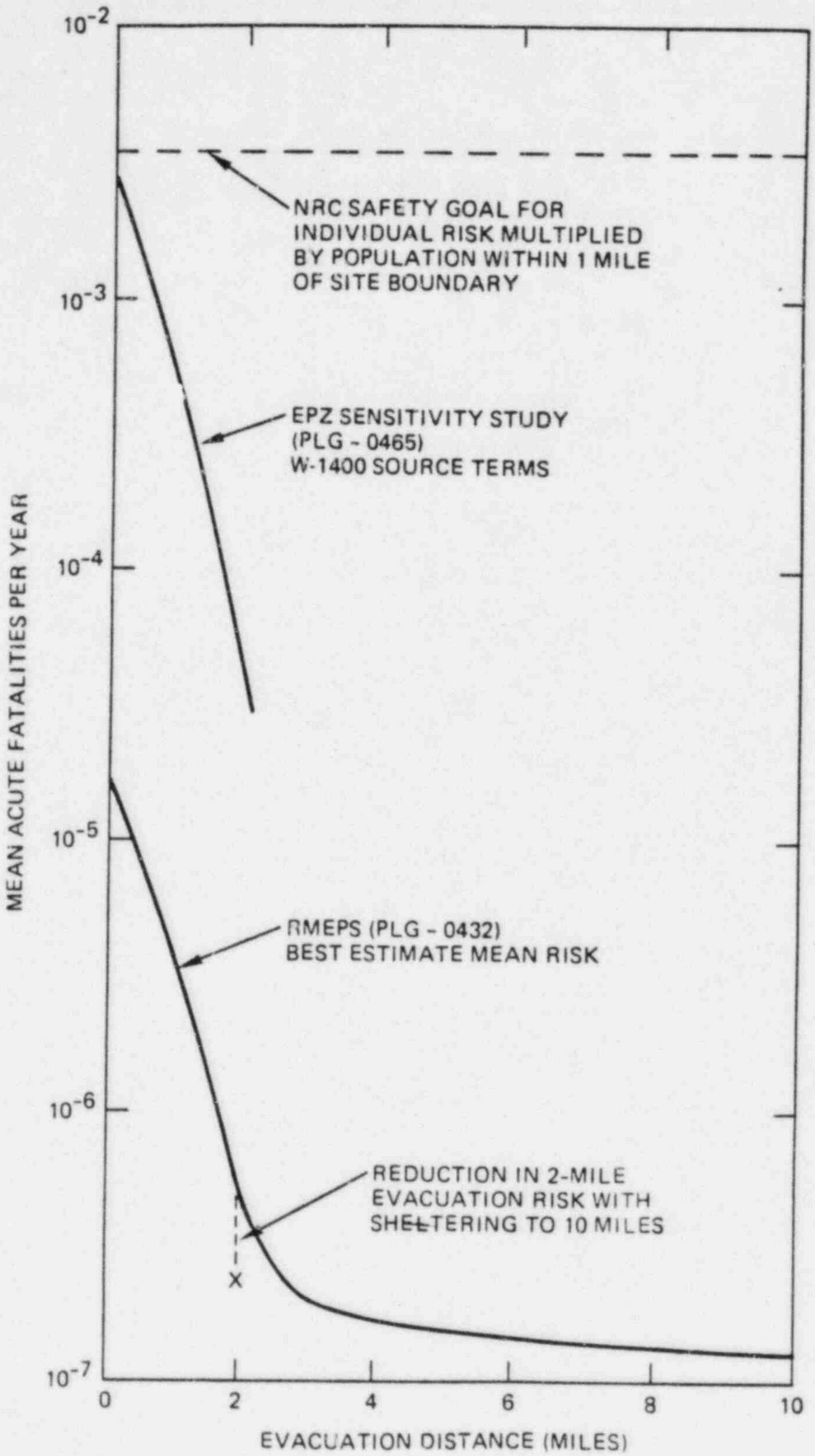


FIGURE 4 ACUTE FATALITY RISK AS A FUNCTION OF PROTECTIVE ACTION
(From RMEPS; PLG-0432)

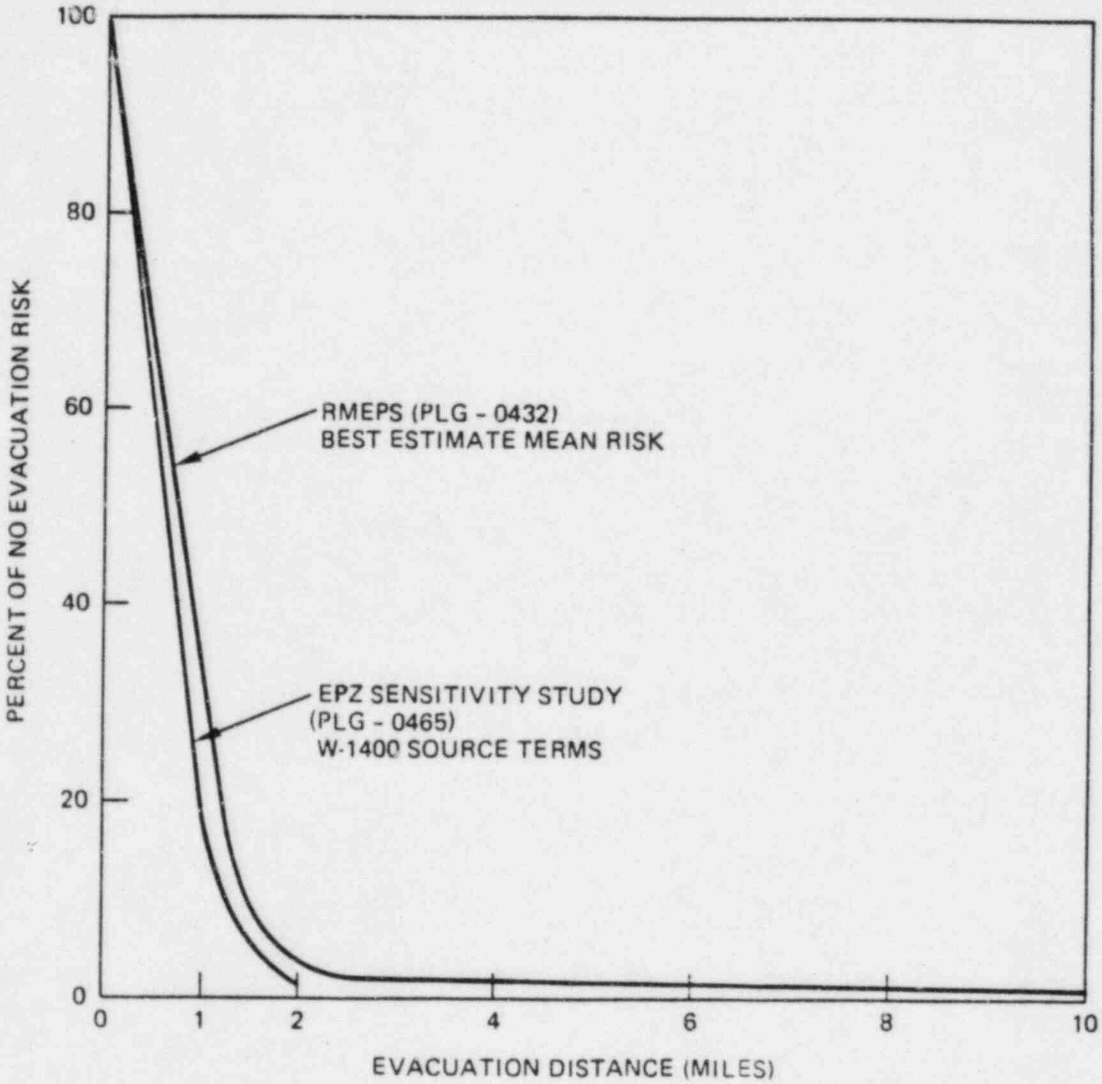


FIGURE 5 RISK REDUCTION AS A FUNCTION OF EVACUATION DISTANCE AT SEABROOK STATION

RAI 24

Provide a quantitative description of the effects of the following differences between the original PSA and the current study:

a. reduction in probability of core-melt V sequences

b. factor of 1000 scrubbing of releases through RHR seals

c. change of release category (S6 to S1) for unscrubbed event V sequences.

The effects should be described in terms of differences in risk curves for early fatalities and for 200 rem vs distance.

RESPONSE 24

Reference: Response 23 Re: 200 REM and early fatalities

In response to part a) of question 24, the following highlights the key factors that result in a major reduction in risk levels for the core melt V-sequence in the updated Seabrook probabilistic safety assessment (PSA) results [per the Risk Management and Emergency Planning Study (RMEPS), PLG-0432, 1985] in comparison with the Seabrook Station Probabilistic Safety Assessment (SSPSA) (PLG-0300, 1983). Qualitatively, the key differences fall into three main areas of the analysis: initiating event frequency. The response to 24 b. will be provided in the response to 30. plant response to various types of interfacing loss of coolant accident (LOCA) scenarios, and operator actions to prevent core melt and isolate the bypass. Description of each of these areas follows.

1. INITIATING EVENT FREQUENCY

The initiating event frequency model in the SSPSA considered four residual heat removal (RHR) cold leg injection paths, each having two series check valves, and two RHR hot leg suction paths, each having two series motor-operated valves (MOV). The check valve model considered successive, independent ruptures of, first, the inboard and, second, the outboard check valve. The second failure was assumed to occur at the same rate as the first at any random time between the first failure and the next test (refueling). The MOV model included a similar sequence of ruptures (inboard, then outboard), as well as the possibility that the outboard valve is already open when the inboard valve fails. The updated model included all the above failure modes plus several more. For the check valve failures, it was conservatively assumed that the outboard → inboard sequence could also occur and that it would occur at the same rate as the inboard → outboard sequence. In addition, the possibility of instantaneous failure of the second valve in a sequence at the time the first valve failed was also considered. Hence, the model used in the update is more complete. The net effect of those model enhancements is worth about a factor of 2 increase in the frequency of a given leak size.

A second difference between the two studies was the definition of the initiating event. In the SSPSA, the V-sequence initiator was defined as a major rupture leading to RHR overpressurization. In the RMEPS, any rupture with a leak flow exceeding 150 gpm (capacity of one charging pump) was considered an initiator. Such flows are not capable of overpressurizing the RHR system when the RHR relief valves operate properly.

A third difference in the initiating event frequency was in the treatment of check valve data. The SSPSA used check valve rupture data--actually zero failures in a large sample of component hours per population--taken from the Indian Point 2 and 3 PSAs. In RMEPS, a different approach was based on a frequency-magnitude correlation of nuclear grade RHR and reactor coolant system (RCS) check valve experience in U.S. pressurized water reactors (PWR). These data are documented in RMEPS and in a separate submittal.

To put the corresponding analyses on a common footing, the frequencies of RHR overpressurization events can be compared as follows:

SSPSA: 1.8×10^{-6}/reactor year.
RMEPS Update: 7.1×10^{-7}/reactor year (leak > 1,800 gpm).

Thus, the net effect of the model differences (which have an increasing effect in the update) and the data treatment (which has a decreasing effect in the update) is a reduction in the frequency of valve ruptures leading to RHR pressurization by a factor of 2 to 3. Hence, if no other changes would have been made to this analysis, the V-sequence risk contribution (and its early release frequency contribution) would have decreased by this same factor.

PLANT RESPONSE

The plant response to RHR interfacing valve ruptures in the SSPSA and in most previously published probabilistic risk assessments (PRA) on PWRs has been treated rather simply, according to the following assumptions, without consideration of their incremental probability.

- Valve ruptures produce a shock wave with peak dynamic pressures significantly greater than the RCS pressure that travels down the low pressure RHR piping.
- RHR piping ruptures outside the containment.
- RCS and refueling water storage tank (RWST) inventories leak outside the containment via a piping break.
- Core melt occurs with unsubmerged bypass.
- No credit is taken for any operator actions.

Therefore, the plant response to the V-sequence in the SSPSA was treated as a single sequence. It was assigned to plant damage state 1F, which, in turn, was assigned to release category S6. In the RMEPS update, a large number of alternative scenarios were identified to provide a more complete picture of plant response. The most important variables introduced in the update to consider alternative plant responses are the size of the valve ruptures that initiate the event, the response of the RHR relief valves inside the containment, the pressure capacity of RHR low pressure piping, the response of RHR pump seals to overpressure, and the configuration of the RHR pump vaults with regard to source term implications.

The major differences in plant response to RHR interfacing valve ruptures, as modeled in the SSPSA and the RMEPS update, are illustrated in Figure 1. This figure is a highly simplified version of the event sequence model that was developed and quantified in the RMEPS update. The chief differences in the update in this regard are a lower frequency of unsubmerged pipe rupture-type bypasses because of the high capacity of the RHR piping. A more likely outcome is a submerged bypass via the RHR pump seals.

OPERATOR RESPONSE

Because of a different treatment of hardware and plant response, the potential for operator actions to mitigate the effects of the interfacing valve ruptures was appropriately considered in the update. The two key actions, which are illustrated in Figure 1, are those to prevent melt and to isolate the bypass. If the RHR piping remains intact, there is a high chance, as assesed in RMEPS, that the operators would prevent core melt whether or not the bypass was isolated. The key is to diagnose the bypass at the RHR pump seal and to provide long-term makeup of coolant to the RWST. If this action is not successful, there is some chance that the operator can isolate the bypass, but only for the discharge check valve rupture case (VI).

The net effects of the major update factors are summarized in Table 1.

TABLE 1. IMPACT OF KEY FACTORS IN UPDATED V-SEQUENCE ANALYSIS

Factor	SSPSA	RMEPS
Frequency of RHR System Overpressurization	1.8×10^{-6} per Reactor Year	7.1×10^{-7} per Reactor Year
Percent of Overpressurization Events that End with:		
• No Core Melt	0	93
• Melt with No Bypass	0	~1
• Melt with RHR Seal Submerged Bypass	0	~5
• Melt with Unsubmerged RHR Pipe Rupture Bypass	100	~1

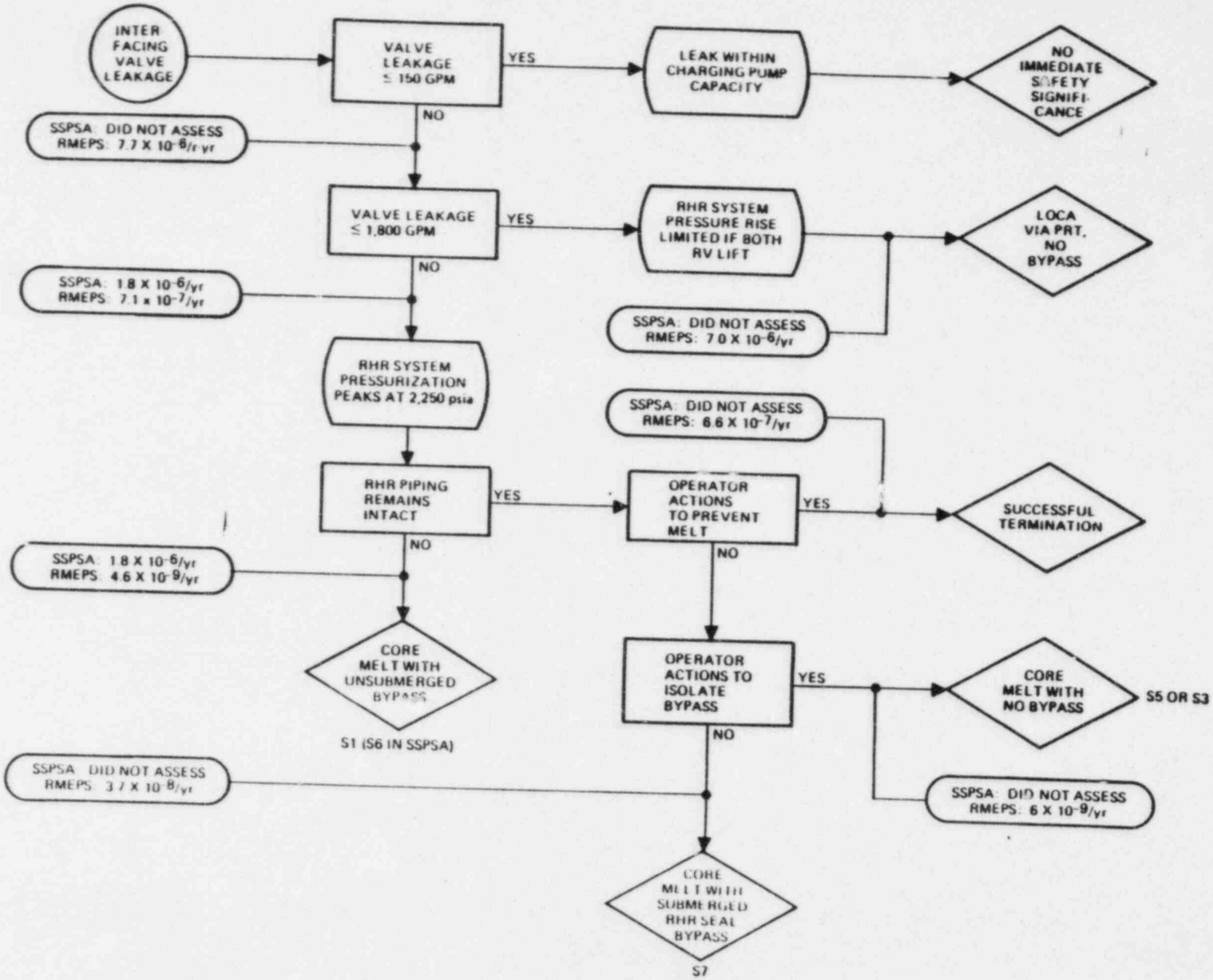


FIGURE 1. SIMPLIFIED EVENT SEQUENCE DIAGRAM FOR V-SEQUENCE

RAI 25

Provide a list of all paths for loss of RCS inventory outside containment. Show how these have been considered with respect to LOCA and with respect to containment bypass for radioactive materials following core damage.

RESPONSE 25

Loss of RCS inventory outside containment could occur via a flow path which directly links the RCS to systems outside containment or in conjunction with the secondary side of the plant assuming steam generator tube rupture has occurred. In either case, a specific containment penetration can be identified as being associated with that flow path.

The Containment Penetration Analysis (PLG-300, Table D.13-3) in the SSPSA accounts for all such penetrations and either provides or references a description of the equipment failure or maloperation necessary to bypass containment. Penetrations associated with flow paths which would be operating during an accident are also listed. Penetrations which are associated with flow paths that could line the RCS directly with systems outside containment are: X-9, 10, 11, 12, 13, 24 through 31, 33, 35, and 37.

Penetrations which are associated with flow paths that could link the RCS with systems outside containment assuming a steam generator tube rupture are: X-1 through 8, 63 through 66.

An evaluation of LOCA outside containment was performed for the SSPSA, examining each line which communicated with the RCS and penetrated the containment. Based on that evaluation, six lines were considered in detail - four RHR cold leg injection lines and two hot leg suction lines. (See SSPSA §6.6.3.2.1). These are the classic "V-sequence" lines first discussed in WASH-1400.

This evaluation of LOCA outside containment was not documented explicitly in the SSPSA but can be reconstructed from the SSPSA §D.13. Table D.13-3 contains a list of all the containment penetrations with the related isolation valves and affected system. In order to be of interest in the evaluation of LOCA outside containment, the line must not only penetrate the containment to the atmosphere but also communicate with the RCS. Each penetration is discussed below. Reference is made to Table D.13-3 in SSPSA, PLG-300.

LOCA OUTSIDE CONTAINMENT

Containment Penetration	System	Penetrations Quantified in SSPSA	Comments
X-1, X-2, X-3, X-4	Main Steam	Yes	Quantified in §5.3.11, Steam Generator Tube Rupture with secondary side leak to atmosphere.
X-5, X-6 X-7, X-8	Main Feedwater	No	See Table D.13-3 "Comments"
X-9, X-10	RCS/RHR	Yes	Quantified in §6.6.3.2.1, RHR hot leg suction.
X-11, X-12	RHR	Yes	Quantified in §6.6.3.2.1, RHR cold leg injection.
X-13	RHR	No	Discussed in §6.6.3.2.1, RHR hot leg injection.
X-14, X-15	Containment Building Spray	No	No direct communication with the RCS.
X-16, X-18	Containment Online Purge	No	No direct communication with the RCS.
X-17	Equipment Vent System	No	No direct communication with the RCS.
X-20, X-21, X-22, X-23	PCCW	No	No direct communication with the RCS.
X-24, X-25, X-26, X-27	Safety Injection	No	Each line has at least two check valves and one normally closed MOV in series.
X-28, X-29, X-30, X-31, X-33	CVCS	No	See Table D.13-3 "Comments.
X-32, X-34	Floor and Equipment Drain	No	No direct communication with the RCS.
X-35	SI	No	3/4" test line, has two normally closed AOVs in series.
	RCS	No	1/2" sampling lines
X-36	Deminieralized Water	No	No direct communication with the RCS.

Containment Penetration	System	Penetrations Quantified in SSPSA	Comments
	Nitrogen Gas	No	One check valve and four normally closed valves.
	Reactor Makeup Water	No	Two check valves and two normally closed valves.
X-37	CVCS	No	See Table D.13-3 "Comments".
X-38	Combustible Gas Control Fire Protection	No	No direct communication with the RCS.
X-39	Spent Fuel Pool Cooling and Cleanup	No	See Table D.13-3 "Comments".
X-40	Nitrogen Gas	No	No direct communication with the RCS.
	RCS Sampling	No	No direct communication with the RCS.
X-43, X-47 X-50	RCS, SI	No	See Table D.13-3 "Comments".
X-52	Post Accident Monitoring	No	See Table D.13-3 "Comments".
X-57	SS, SI	No	See Table D.13-3 "Comments".
X-60	CBS	No	See Table D.13-3 "Comments".
X-63, X-64, X-65, X-66	S/G Blowdown	Yes	Quantified in §5.3.11.4 Steam Generator Tube Rupture with secondary side leak. See top event IV (p.5.3-95).
X-67	Service Air	No	No direct communication with the RCS.
X-71, X-72	Combustible Gas Control	No	No direct communication with the RCS.
HVAC-1, HVAC-2	Containment Air Purge	No	No direct communication with the RCS.

Penetration #	Isolation Valve	Nominal Diameter Inches	Comments
X-72	CGC-V-10	1"	Manual locked closed valve that isolated Train A H_2 analyzer input line.
X-72	CGC-V-3	1"	Manual locked closed valve that isolated Train A H_2 analyzer return line.
X-39	SF-V-86, 87	2"	Manual locked closed valves. Would only be utilized in conjunction with the refueling canal skimmer pump.
X-67	SA-V-229, 1042	2"	Manual locked closed valve outside containment, normally closed valve inside containment.

Note: Test connections were not considered in this analysis. Test connections are 3/4" with normally closed manual valves and a pipe cap. Per the Seabrook Station Technical Specifications there are verified closed every 31 days.

RAI 26

Indicate the extent to which the effect of local deflagration/detonation of hydrogen gas concentration is localized areas both inside and outside the containment has been considered in the assessment of risk. Include a discussion of how weak areas of containment have been considered in your assessment, for example, the containment is considerably weaker in its resistance to pressure loading from outside the containment.

RESPONSE 26

A separate probabilistic analysis of the effects of hydrogen combustion has been performed for each plant damage state. The analysis accounted for uncertainties in the hydrogen generation, release from the primary system, ignition, and containment atmospheric conditions. Vessel breach discharge burns and global burns at different times in the accident progression were treated separtely. All hydrogen burns were treated as instantaneous adiabatic combustion events. The analysis is documented in Section 11.5.2 and in Tables 1.7-1 and 11.7-7 of the SSPSA (PLG-0300). Local hydrogen deflagrations or detonations were considered and dismissed as requiring conditions of nearly stagnant or quiescent atmospheres which are not considered credible under accident conditions. In a large dry containment thermal and mass transfer induced mixing of the containment atmosphere under accident conditions is considered assured particularly on those accident phases where rapid releases of hydrogen into the containment are possible such as at vessel breach.

Weak areas in the containment with respect to the capability to contain hydrogen burns could not be identified. A maximum adiabatic post burn presure of 128 psia was determined for a limiting vessel breach discharge burn. The lowest containment failure pressure was identified at 181 psia for the type A failure (0.5 square inch leak area).

Hydrogen burns outside the containment in the annulus between the primary and secondary containment could be postulated. Such hydrogen burns would have no impact on the calculated risk or conclusions with respect to emergency planning requirements for two reasons:

1. In order for hydrogen to accumulate to a flammable mixture in the annulus region, the hydrogen must be released from the containment and this requires that the containment is already failed. Furthermore, the concentration in the annulus would be lower than in the containment due to the additional mixing with the annulus air.

2. A hydrogen burn in the annulus would impose an equal load on the enclosure building and on the containment building. Even though the containment may be weaker for external loads than for internal loads, the external load capacity is definitely much greater than the internal load capacity of the containment enclosure building which was evaluated in the SSPSA Appendix H.1, Section 6. The weak elements in the enclosure pressure boundary are identified as the HEPA filters and the sheet metal duct work, each of which is expected to fail at a pressure between 1 and 2 psid.

RAI 28

Identify any penetrations connected directly into the containment atmospheric which rely on any remote manual or manual valves for isolation.

Response 28

Penetration #	Isolation Valve	Nominal Diameter Inches	Comments
X-14 X-15	CBS-V-11 CBS-V-17	8" 8"	To CBS spray rings. Normal closed motor operated valves open on Hi-3 containment pressure or manual. Valve closure is manual.
X-60 X-61	CBS-V-14 CBS-V-8	16" 16"	Containment sump return line MOV. Normally closed. Valves open automatically on ECCS/CBS recirculation signal indicated by 2/4 Lo-Lo level in the RWST. Valve closure in manual.
X-38	CGC-V-43 CGC-V-44 CGC-V-45	2" 2" 10"	Compressed air line to containment for pressure testing. Manual locked closed valves. Line is used only for containment pressure testing.
X-38	FP-V-592	4"	Manual locked closed valve outside containment, normally closed valve inside containment.
X-71	CGC-V-36	2"	Manual locked closed valve. This valve is in series with CGC-V-28 located inside containment which will isolate on a containment isolation signal.
X-71	CGC-V-32	1"	Manual locked closed valve that isolates the Train B H_2 analyzer input line.
X-71	CGC-V-24	1"	Manual locked closed valve that isolates the Train B H_2 analyzer return line.
X-72	CGC-V-15	2"	Manual locked closed valve. This valve is in series with CGC-V-14 located inside containment which will isolate on a containment isolation signal.

Penetration #	Isolation Valve	Nominal Diameter Inches	Comments
X-72	CGC-V-10	1"	Manual locked closed valve that isolated Train A H_2 analyzer input line.
X-72	CGC-V-3	1"	Manual locked closed valve that isolated Train A H_2 analyzer return line.
X-39	SF-V-86, 87	2"	Manual locked closed valves. Would only be utilized in conjunction with the refueling canal skimmer pump.
X-67	SA-V-229, 1042	2"	Manual locked closed valve outside containment, normally closed valve inside containment.

Note: Test connections were not considered in this analysis. Test connections are 3/4" with normally closed manual valves and a pipe cap. Per the Seabrook Station Technical Specifications there are verified closed every 31 days.

RAI 33

Confirm that a complete and independent check will be performed for the containment strength calculations that served as the basis for the EPZ sensitivity study.

RESPONSE 33

A complete and independent check will be performed for the containment strength calculation. This effort will be completed on approximately November 25, 1986.

RAI 34

Fully address the effect of uncertainty in the ultimate strength of Cadweld splices on the pressure capacity of the containment. As discussed in the meeting your response should address potential, non-ductile failure of the Cadweld splices.

RESPONSE 34

See response to RAI 8.

RAI 35

Assess the response of the containment sump encapsulation vessel on the containment integrity.

RESPONSE 35

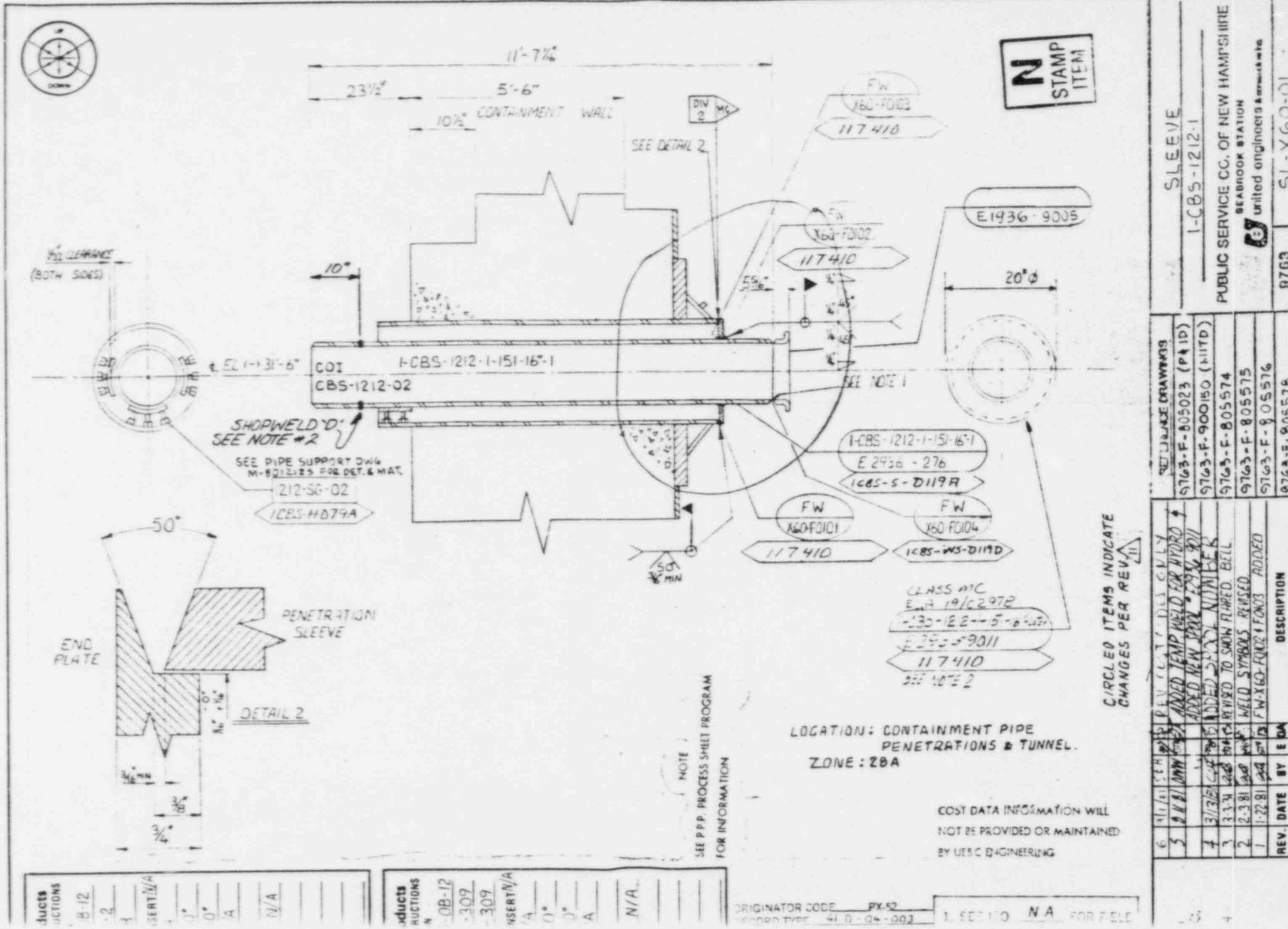
The sump encapsulation vessel is not a part of the primary containment pressure boundary. See attached drawings for penetration X-60 (typical of X-61 also)

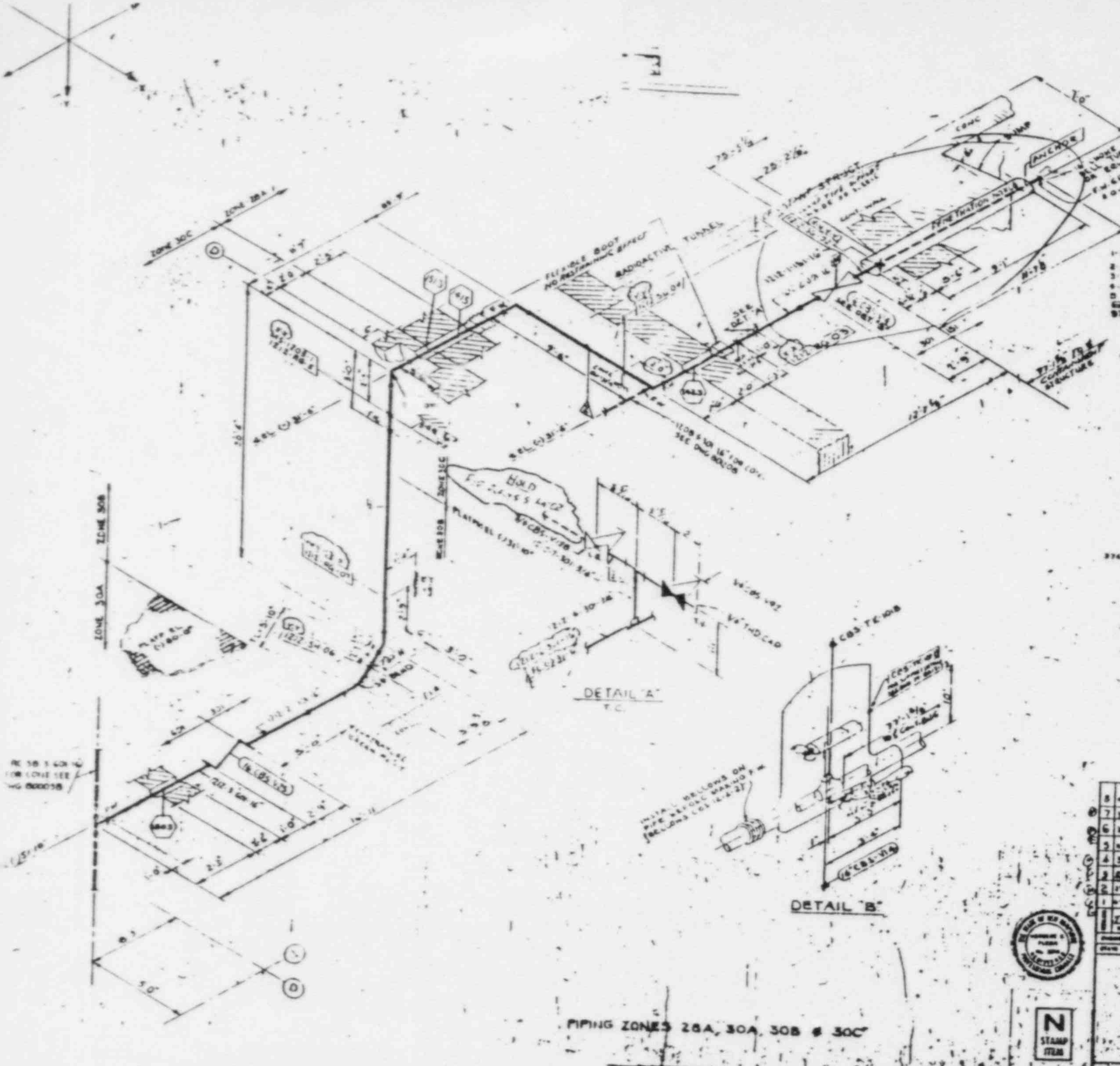
9763-SL-X60-01
9763-D-801212

Inside containment, CBS line 1212-1-151 is welded by partial penetration welds to a flat plate penetration closure. In turn, the flat plate is welded to the penetration sleeve. Containment pressure is present in the piping up to Motor Operated Valve CBS V-14. The encapsulation vessel surrounds this valve and is present only as a secondary boundary to contain valve stem leakage, if any.

This penetration is also not subject to the large shell deflections estimated in the SMA ultimate capacity analysis since the line penetrates containment below the basemat elevation, (-31'-6").

INFORMATION ONLY





NOTES:

1. ⬡ INDICATES PIPE SLEEVE NO.
2. E.P. INDICATES EMBEDMENT PLATE.
3. ALL ELBOWS TO BE LONG RADIUS, UNLESS OTHERWISE NOTED.
4. F.W. DENOTES REQUIRED FIELDWELD LOCATIONS.
5. ALL BENDS ARE 5D UNLESS OTHERWISE NOTED.

REFERENCES:

9763-F-808060 RHR & CNTMNT SPRAY EQUIP. VAULT GENERAL
805180 RADIOACTIVE TUNNEL COMPOSITE PIPING ZONE
805200 RAD. VAULTS PIPING ZONE 30A PLAN
805201 RAD VAULTS PIPING ZONE 30B PLAN
805202 RAD VAULTS PIPING ZONE 30C PLAN
805023 CNTMNT SPRAY SYSTEM P&I DIAGRAM
805008 RESIDUAL HEAT REMOVAL SYSTEM P & I
101625 MECHANICAL PENETRATION AREA, CONCRETE
101547 RHR & CNTMNT SPRAY EQUIP. VAULT - STEEL
101510
101658 RHR REMOVAL & CONT. SPRAY EQUIP VAULT STEEL
808540 CNTMNT SPRAY
808868 RHR & CNTMNT SPRAY
808866 RADIOACTIVE TUNNEL
808876 CNTMNT STRUCT
808047 COMPOSITE PIPING
218-1 PIPE FABRICATION
248-8 PIPE SUPPORT SPEC.
PIPE INSULATION SPEC.
VALVE LIST
PIPE SUPPORT & RESTRAINT
CONCENTRATED LOAD DATA SHTS.
LINE LIST
9763-D-808000 WELD END PREP. DETAILS
WELD END WELD
248-43 PIPING DESIGN SPEC.

RECEIVED
U.E.&C.
DEC 7 1981
SEABROOK STATION

PIPING ZONES 28A, 30A, 30B & 30C

N STAMP ITEM

ISOMETRIC PIPING
CBS SYSTEM
LINE NO. 1212
PUBLIC SERVICE CO. OF NEW HAMPSHIRE
SEABROOK STATION
united engineers

9763-D-801212

RAI 40

What indication are available if RHR is lost during shutdown (e.g. spurious closure of suction valve)?

RESPONSE 40

This is dependent primarily on the type of failure that causes the loss of the Residual Heat Removal System. The list below indicates some various alarms and indications for two scenarios;

1 RHR Pump trip on overcurrent (faulty overcurrent relay)
- RHR PUMP TRIP BREAKER LOCKOUT alarm on the Video alarm system (VAS)

2 Spurious failure CLOSED of an RHR suction valve.
- RHR LOW-LOW FLOW alarm on the VAS (335 GPM)
- RHR LOW FLOW on overhead alarm annunciator (555 GPM)
- RHR HEAT EXCHANGER LOW DISCHARGE FLOW alarm on the VAS (1500 GPM)
- Abnormal current indication on the Main Control Board
- Valve Position on the Main Control Board

Other indications and alarms of a malfunction in the RHR available to the operator at the Main Control Board are;

INDICATIONS (for both 'A' and 'B' train)

- RHR pump amperage (MCB and MPCS)
- RHR pump discharge flow (MCB and MPCS)
- RHR pump discharge pressure (MCB and MPCS)
- RHR heat exchanger inlet temperature (MPCS point)
- RHR outlet exchanger outlet temperature (MPCS point)
- RHR pump motor winding temperature (MPCS point)
- RHR pump radial bearing temperature (MPCS point)
- RHR pump thrust bearing temperature (MPCS point)
- RHR pump PRIMARY COMPONENT COOLING WATER flow (MPCS point)
- RHR heat exchanger PRIMARY COMPONENT COOLING WATER flow (MPCS point)

ALARMS (for both 'A' and 'B' train)

- RHR pump motor winding high temperature (VAS)
- RHR pump radial bearing high temperature (VAS)
- RHR pump thrust bearing high temperature (VAS)
- RHR pump discharge pressure high (VAS)
- RHR pump PRIMARY COMPONENT COOLING WATER low flow alarm (VAS)
- RHR heat exchanger PRIMARY COMPONENT WATER low flow alarm (VAS)

RAI 41

What indication is available for vessel level during shutdown and refueling modes?

RESPONSE 41

As the plant is brought out of the standby condition and into hot shutdown and cold shutdown, the Reactor Vessel Level Instrumentation System (RVLIS) indicates at the main control board the level in the vessel. This system is available both with the RC pumps operating (dynamic-range) and without the RC pumps running (full-range). The RVLIS has two electrically-independent trains of instrumentation and indication. The RVLIS is not available when the head is off the vessel. During shutdown when the vessel head is off the reactor vessel, or the RCS is otherwise open to the atmosphere, two methods of vessel level determination are available. First, a clear plastic tube is connected to one of the loop drain lines and routed vertically to an elevation above the reactor vessel head to serve as a level guage. The RCS is drained either to just below the vessel flange (refueling) or to the midpoint of the loop piping (steam generator maintenance). The loop drain line connection is made at the bottom point of the RC piping. Therefore, this method of temporary level measurement is available throughout the range of draining required for all maintenance which may take place while fuel is in the vessel. The second method of vessel level determination is by level tranmitter RC-LT-9405 which indicates on the MCB (BF) by RC-LI-9405. This method is also available only when the RCS is open to atmospheric pressure, and is valved-out during normal operation.

A final metod of vessel level determination is by the Refueling Cavity Level Transmitter SF-LT-2629. SF-LI-2629 indicates on the MCB (BF). This method is available and applicable only when the refueling cavity is in communication with the RCS; that is, when the cavity level is above the reactor vessel flange with the head off and the fuel transfer tube gate open.

RAI 42

Does loss of power to the pressure transmitter that provides input to the autoclosure interlock for RHR suction valve cuase the valves to close?

RESPONSE 42

Loss of power to the pressure transmitter that provides input to the autoclosure interlock for RHR suction valve will not cause the valve to close.

RAI 43

To what level(s) is the RCS drained for maintence activities while shutdown with fuel in the vessel? What level is necessary to maintain connection with the ultimate heat sink?

RESPONSE 43

For maintenance and refueling the RCS may be drained to the level required for the activity. The two typical stopping points are 6-inches below the reactor vessel head and to the midplane of the reactor coolant loop piping for steam generator, reactor coolant pump and valve maintenance. The midplane of the reactor coolant loop piping is the minimum level to ensure proper operation of the Residual Heat Removal System. Estimates of the time that might be expected at the different water levels based on Zion data is provided in response to RAI 21.

RAI 44

Describe the availability of the SI pumps while shut down. How difficult would it be to restore the SI function to respond to transients during shutdown and refueling conditions? Consider maintenance of the SI system in your response.

RESPONSE 44

The Safety Injection (SI) pumps (intermediate head safety injection) are to be SAFETY TAGGED (with a CAUTION TAG) with the motor circuit breakers open and rack-out within four hours after entering MODE 4 (average RCS temperature <325 degrees F but >200 degrees F) OR any one of four RCS cold legs temperatures <325 degrees F, whichever comes first.

To restore the SI function of these pumps, the SAFETY TAGS must be removed from the breaker(s) and the breaker(s) racked into the OPERATE position. They may then be started manually or automatically by a Safety Injection Actuation Signal. A more complete discussion of this subject is in response RAI 21.

RAI 45

Provide the procedure for establishing cold overpressure protection when shutting down.

RESPONSE 45

Cold Overpressure Protection at Seabrook does not require manually arming. It is armed automatically on Reactor Coolant System temperature decreasing to less than 342 degree F. The attached illustration below shows the logic employed in arming and operating the LTOP system (Low Temperature Overpressure Protection) during an RCS cooldown and depressurization for Protection Train 'A', Train 'B' logic is similar.

PORV (Power Operated Relief Valve) Block Valve auto-open (NOTE: this is a NORMALLY OPEN valve.);

- Power available to valve AND
- valve control in remote AND
- valve control in automatic AND
- LTOP train 'B' armed (auctioneered low wide range RCS cold leg temperature is less than 342 degrees F.)

PORV Train 'A' will auto open if;

- Valve in automatic AND
- Wide Range Reactor Coolant pressure channel (PT-403) is greater than the LTOP setpoint (see attached graph, Protection Train 'A' LTOP setpoint generated by auctioneered low RCS hot leg temperature.)

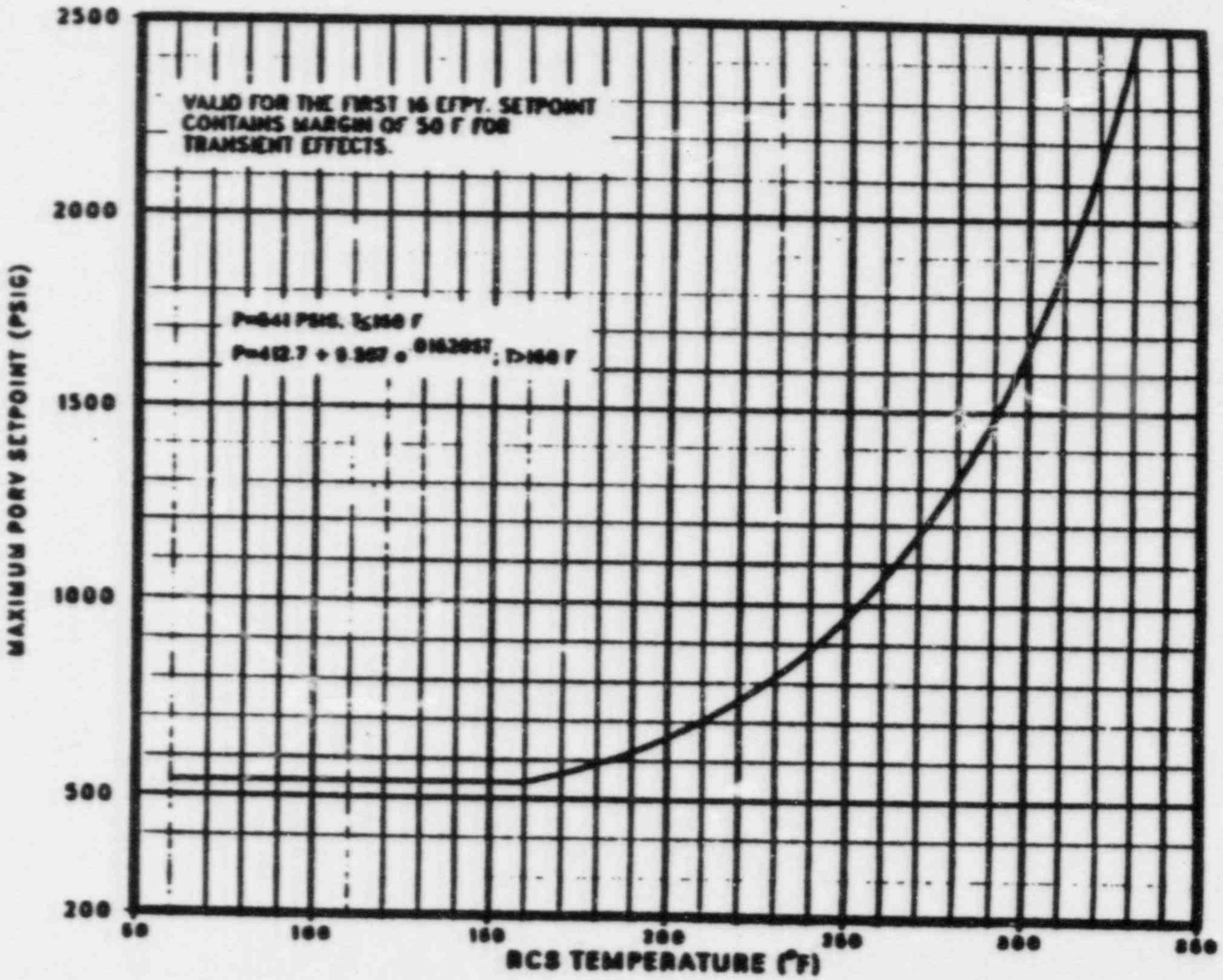


TABLE 1. RCS COLD OVERPRESSURE PROTECTION SETPOINTS

QUESTION 46

Is the primary system made water-solid during shutdown?

RESPONSE 46

There is at present no plans to operate the RCS for long periods of time in the water solid condition. If the plant is shutdown to refuel, or for other maintenance, the pressur'zer will be taken solid per operating procedure to collapse the steam bubble, then drained to the required level. Another condition when the RCS is water solid would be when drawing a pressurizer steam bubble following filling and venting of the RCS. This would only be a transitory condition. Another possible mode of operation is to have the RCS filled and vented to the PRT.

RAI 49

The FSAR gives RHR relief valve flow rate as 900 gpm with a set pressure of 450 psi. The flow rate does not agree with the value used in Reference 1, section 3, page 6. Please explain.

RESPONSE 49

The correct rated flow of each RHR relief valve is 900 GPM at 450 PSI. Although the value of 990 GPM stated is incorrect, the V-sequence probability modeling used a total of 1800 GPM for both relief valves. It was assumed that a valve rupture flow of greater than 1800 GPM would result in RHR pressurization to 2250 psia. For the flow calculations in MAAP a pressure dependent flow capacity was used as described in RMEPS. This is somewhat conservative since the initial driving pressure is over 2000 PSI.

RAI 50

Please describe the mechanism for assuring that plant changes and new knowledge are promptly factored into the technical considerations which form a part of the foundation for staff consideration of a reduced emergency planning zone radius.

RESPONSE 50

The PSA will be kept up-to-date with regard to plant changes and new knowledge. Wherever major plant modifications are performed which significantly affect the PSA, the report will be updated accordingly. To ensure prompt and effective input to plant change decisions, the PSA will be considered in the design change review process. PSA considerations will include the impact on public health risk.

The RMEPS results already reflect some of the changes to plant Technical Specifications that have been made since the SSPSA was completed in 1983. Recently, a review has been completed by the PSA team of changes made to the plant since 1983. No changes were identified that would impact the current PSA results in any significant way.

RAI 51

Reference 1, page 3-7, paragraph 5) references both high and low level sump alarms. What is s sump low level alarm?

RESPONSE 51

The CBS pump cubicle sump pump is tripped on low sump level and an alarm is annunciated at the waste management system control panel (CP-38A) and remotely in the main control room via the VAS. The pump trip and low level alarm are set to protect the pump from cavitation.

RAI 53

Reference is made on page 3-7 to the RHR system crosstie line and RHR system response due to flow in this line as well as in the miniflow bypass lines. The conclusion is drawn that the RHR system pressure will tend to be uniform as a result. Are flow conditions such that this is realistic? What is the impact of this assumpation on conclusion pertinant to the discussion.

Response 53

Page 3-7 states that "...the entire RHR system will tend to pressurize uniformly after valve failure (neglecting the time it takes for pressure valves to traverse the system)". The figure included with this response provides a diagram of the RHR system. The configuration of the Residual Head Removal System when it is aligned as part of the emergency core cooling system is for both RHR pumps to be aligned through separate suction paths from the refueling water storage tank. The discharge side of the pumps are connected downstream of their respective heat exchangers by an 8-inch crosstie line containing two open motor operated valves. Each pump has a minimum flow line that connects the outlet of the RHR heat exchanger to the pump suction piping. The minmum flow line is a 3-inch pipe containing a flow restricting orifice (750 GPM at 185 PSI) and a normally open motor operated valve. In the RHR pump suction piping, branches from the refueling water storage tank, the containment sump and the reactor coolant loop join the RHR pump suction pipe. It is the pump suction pipe from the reactor coolant loop that contains the 3x4 inch relief valve set at 450 psig with a rated flow of 900 gpm. Table 4-6 and Figures 3-1 and 3-2 of PLG-0432 provides additional details of RHR flow paths.

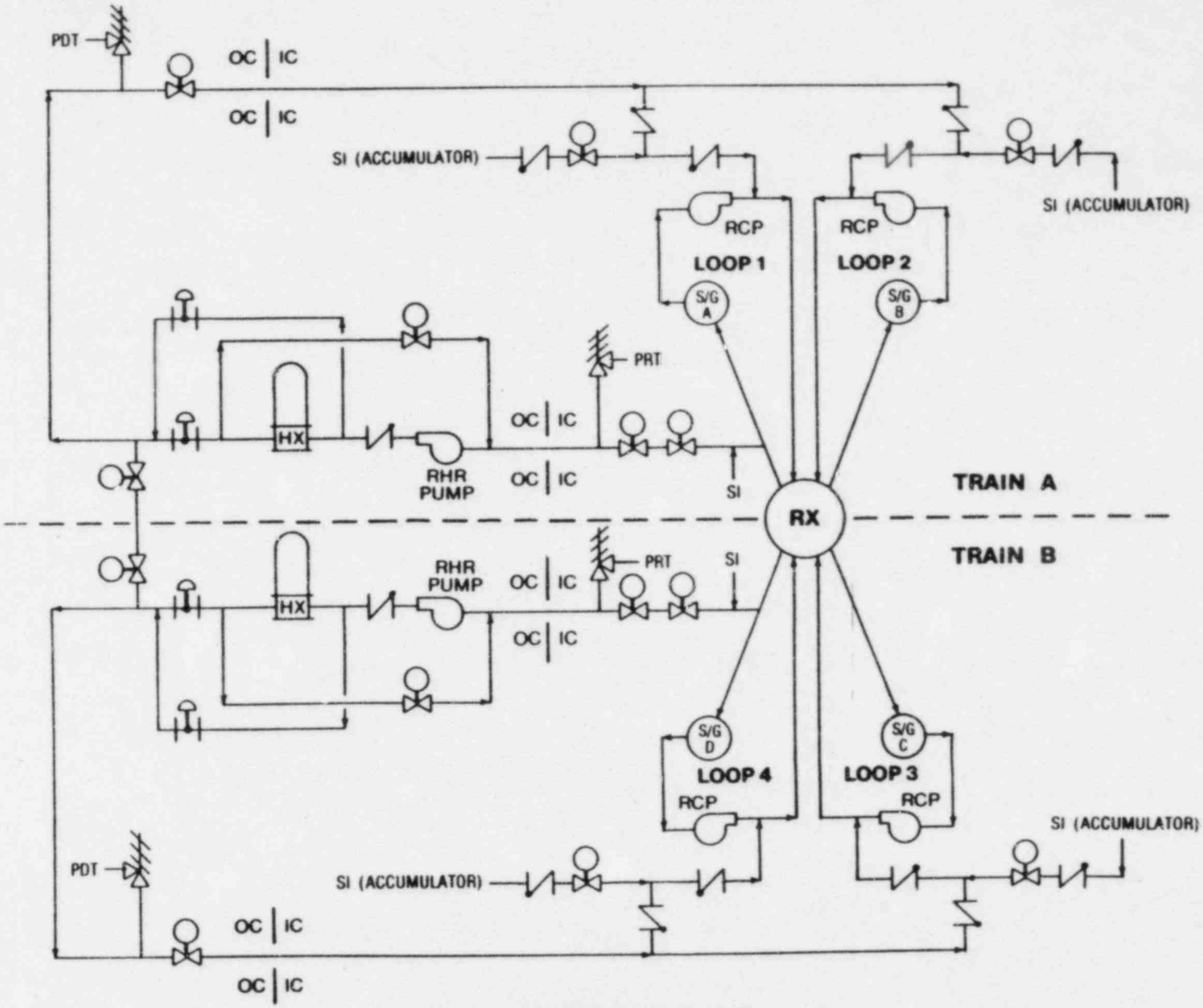
Except for the RHR pipe failure case (which maps to the most severe release category, S1), the interfacing LOCA flow rates through to RHR system are expected to be moderate (a few thousand GPM) relative to the RHR piping equipment size (see, for example section 3.1.4.2 and Figure 3-7 of PLG-0432). The pressure losses through the RHR system will not be large such that the pressure should be fairly uniform.

We see no significant "impact of this assumption on conclusions pertinent to the discussion" for the following reasons:

1) As discussed throughout PLG-0432, the RHR system overpressure failures modeled conservatively envelop the spectrum of possible failure modes.

2) The RHR pressure boundary failure(s), if any, are expected to be caused by the initial pressure wave or pulse, not from flow induced pressure drops.

3) The precise pressure distribution through the RHR is of little importance since the risk results are not extremely sensitive to the exact RCS blowdown/RHR pressurization rate.



RHR SYSTEM OVERVIEW

RAI 55

What is the justification for the statement on page 3-10 that the first sign of trouble wil' be pressurizer low level or low pressure alarms? We suspect a number of other indicators may be first, such as abnormal indication from the PRT or even a smoke alarm.

RESPONSE 55

The indications of an RHR Interface LOCA would primarily depend on the magnitude of the inleakage from the RCS as was stated on page 3-10. If the RCS to RHR leak flow rate is less than the capacity of the RHR suction relief valves, the first indications the operator may receive are;

- PRT (pressurizer relief tank) high pressure
- RHR discharge pressure meters readings abnormally high, 450 PSIG (0-700 PSIG scale)
- RHR heat exchanger outlet temperature recorders pegged high (0-500 Degrees F)
- Pressurizer Level low (-5% deviation from programed level)
- Pressurizer pressure low (2210 PSIG)

For in-leakage less than 1800 GPM, the RHR system would pressurize to approximately 450 PSIG (the setpoint of the suction relief valve) and stay at this point. The RHR suction relief valve(s) would cycle to maintain the RHR system at 450 PSIG. It is important to note that this would entail no threat to the RHR system integrity, the mechanical seal package would remain INTACT and no leakage would be seen outside containment.

If the RCS to RHR in-leakage was greater than 1800 GPM, the indications the operator would receive would be somewhat different. Based on the worst case RHR Interface LOCA sequence run on the MAAP code by Westinghouse, the following alarms/indications would be received by the operator. (please refer to plot of RCS Pressure vs Time for basis of alarm sequence) These alarms/indications are listed in CHRONOLOGICAL order and DO NOT take credit for any radiation monitoring systems alarms, although there are both area and ventilation process monitors that monitor the RHR equipment vault area;

- RHR DISCHARGE PRESSURE HIGH alarm train 'A' ('B') (560 PSIG)
- RHR discharge pressure meters pegged high (0-700 PSIG)
- RHR heat exchange outlet temperature recorders pegged high (0-500 Degrees F)
- PRT Pressure High alarm (4 PSIG)
- PRT Pressure and level increasing with no indication that the pressurizer PORV(s) or Safety Valves are discharging. Pressurizer PORV(s) and Safeties have the following devices to monitor their position and process flow thru them;
    - Valve position for PORV(s)
    - Tailpipe temperature for PORV(s) and Safety Valves
    - Acoustic flow monitoring for PORV(s) and Safeties
- PRESSURIZER PRESSURE LOW alarm (2210 PSIG)
- PRESSURIZER LEVEL LOW alarm (-5 deviation from program)
- RHR EQUIPMENT VAULT HIGH TEMPERATURE alarm Train 'A' ('B')
- RHR EQUIPMENT VAULT HIGH SUMP LEVEL alarm Train 'A' ('B')

RAI 57

The statement on page 3 - 11 that "As soon as the pumps begin to produce flow to the RCS, valves in the miniflow lines close and all RHR pump flow is injected into the reactor vessel via the RHR cold leg injection lines" is not correct. The sensors are not located at the RCS to detect flow at that location. Further, one is postulating a break in the RHR system, and a significant portion of the pump flow may never reach the RCS (as it stated in a later paragraph).

RESPONSE 57

The referenced paragraph states that "As soon as the RHR pumps begin to produce flow to the RCS, valves in the miniflow lines close and all RHR flow is injected into the vessel via the RHR cold leg injection lines". This paragraph describes the response of the Emergency Core Cooling Systems to a standard LOCA, and is correct. The RHR flow transmitters are not located on the RCS proper, but are indicative of RHR injection flow via the cold leg injection lines.

RAI 59

An item under consideration for advanced nuclear power plants is the ability to monitor pressure on the low pressure side of check valves. This could provide early warning of check valve leaks and would provide monitoring capability to help assure check valves were operating properly. The same monitoring capability with respect to RHR suction line valves could identify if individual valves were mispositioned or malfunctioning. Would such a system for Seabrook be of significant benefit in reducing risk in a reduced size emergency planning zone?

RESPONSE 59

The failure model and quantification for leaks greater than 150 gpm is summarized below for the four cold leg injection paths (two check valves in series) and the two hot leg suction paths (two MOVs in series).

4 COLD LEG INJECTION PATHS

NO. OF PATHS	FAILURE MODEL	QUANTIFICATION (ANNUAL TESTING)			CHECK VALVE CONTINUOUS MONITORING
4	2 (T/2)	$4[(4X10^{-4})^2+Variance]$	=	$3.5 X 10^{-6}$	0
4	2 d	$8X(4X10^{-4})X(2.7X10^{-4})$	=	$0.9 X 10^{-6}$	$0.9 X 10^{-6}$

2 HOT LEG SUCTION PATHS

NO. OF PATHS	FAILURE MODEL	QUANTIFICATION (18 MONTH TESTING)			
2	2 (T/2)	$2[(4X10^{-4})^2+Variance]X1.5$	=	$2.7 X 10^{-6}$	$2.7 X 10^{-6}$
2	2 d	$4X(4X10^{-4})X(2.7X10^{-4})$	=	$0.4 X 10^{-6}$	$0.4 X 10^{-6}$
2	g	$2X(4X10^{-4})X(1.1X10^{-4})$	=	$0.1 X 10^{-6}$	$0.1 X 10^{-6}$
				$7.6 X 10^{-6}$	$4.1 X 10^{-6}$

The last column above assumes perfect continuous testing (monitoring) of the two series check valves. As shown the total frequency changes very little (less than factor of 2). However, quantification of the other failure modes and treatment of MOV discs as check valve discs are believed to be conservative.

"Perfect continuous" leak testing of the RCS series check valves would require significant modification and is not practical. Frequent, periodic testing can be performed to verify that excessive leakage is not occurring through the inboard RCS isolation valves. Excessive leakage through the outboard check will possibly be detected by increased pressure in the RHR system and reduced accumulator level depending on the magnitude of the leakage.

Currently the normal RHR suction motor-operated valves cannot be tested as described above because permanent test lines are not connected to the process lines which the valves isolate.

In addition, as discussed in the response to RAI 48b, the interfacing LOCA event contributes approximately 12% to the total early release frequency. Therefore, if this was reduced to zero it would be a minor reduction to total release frequency. There is no significant benefit to reducing risk.

RAI 60

Please elaborate on the page 3-23 list of actions an operator can take to mitigate the accident. This list appears to be short. Include identification of what has been incorporated into operator training and procedures at Seabrook.

RESPONSE 60

At present, the following isolation sequence is being incorporated into training:

1. Check proper valve alignment

   RC-V22 RHR pump "A" Th suction CLOSED
   RC-V23 RHR pump "A" Th suction CLOSED
   RC-V87 RHR pump "B" Th suction CLOSED
   RC-V88 RHR pump "B" Th suction CLOSED

2. Identify and isolate leak

   CLOSE THE FOLLOWING VALVES

   - RH-V21
   - RH-V22

   OBSERVE THE RESPONSE OF THE RHR SYSTEM

   - Faulted train follows RCS pressure/temperature
   - Non-faulted train depressurizes

   CLOSE FAULTED TRAINS TO DISCHARGE VALVE

   - Energize MCC 522/622
   - Close RH-V14/RH-V26

   STOP FAULTED TRAIN RHR AND CBS PUMPS

   CLOSE FAULTED TRAINS RWST SUCTION VALVE

   - CBS-V2
   - CBS-V5

3. Check if break is isolated:

   RCS Pressure - increasing
   or
   Reactor Vessel level - increasing
   and
   Faulted RHR loop pressure - decreasing

4. If break is not isolated:

   - Energize MCC 522 and 622
   - Close RH-V14 and RH-V26

   RETURN TO STEP 3

This strategy allows:

   a. a quick reduction in leakage flow by isolating the faulted train from the non-faulted train

   b. prevents cycling the RH injection valves if not necessary

   c. allows better diagnosis by separation of trains.

RAI 61

What is the frequency of failures in the pipe tunnel that is mentioned on page 3-23, and which led the authors to conclude they are very low?

RESPONSE 61

As noted on page 3-24, top event PI in the VI and VS event trees. represents any failure of piping or the heat exchanger due to the RHR system high pressure challenge. Any piping failures are assigned to plant damage state 1 FV. As Table 4-17 shows, damage state 1FV always maps to release category S1- the most severe release. In other words, any RHR pipe failure is assumed to result in the most severe release, regardless of failure location; therefore results are insensitive to the pipe failure location.

RAI 62

Page 3-27 references situations where the combined sump pump capacity is sufficient to remove leaks and keep the vaults from flooding. In these cases, the RHR, SI, and CS pumps are assumed not to be impacted by flooding. What consideration was given to failure of one (or both) sump pumps?

RESPONSE

As shown in Figure 3-4 of PLG-0432, the sequences referred to in this question have frequencies on the order of 1E-8 or less; these frequencies, when multiplied by the chance of sump pump failure clearly make such sequences unimportant contributors.

RAI 63

What is the maximum flow rate that can be injected into the RCP pump seals? (Of potential interest since it may be an alternate path for injection into the RCS.)

RESPONSE 63

Flow to the reactor coolant pumps seals is significantly less than that required for core cooling.

RAI 63

What is the maximum flow rate that can be injected into the RCP pump seals? (Of potential interest since it may be an alternate path for injection into the RCS.)

RESPONSE 63

Normal seal injection flow into the reactor coolant system through the reactor coolant pump seal injection flow path is 5 GPM per reactor coolant pump or 20 GPM total. This flow path is judged insufficient by itself to satisfy core cooling requirements. Other flow paths such as the normal charging line, the charging/safety injection flow path to the cold legs or the safety injection pumps to the cold legs (at reduced RCS pressures) will provide greater flow capability. It should be noted that there are no risk significant core damage sequences in the PSA results in which the charging pumps are available.

RAI 64

Shutting an RHR system crosstie valve is identified on page 3-35 as an action to help isolate a LOCA outside containment involving the RHR/SI systems. Has a careful evaluation of these systems been performed to assess isolation strategy? If so, are procedures in place at Seabrook Station which reflect the work?

RESPONSE 64

See Question 60.

RAI 67

Page 3-37 contains the wording "End state DLOC contains sequences in which the interfacing LOCA has been terminated, and the ECCS has been degraded (D) (RHR or SI pumps have failed)...The point estimate frequency of DLOC is 4.0×10^{-7} per year. The additional failures required to achieve core melt would lower this frequency by at least one order of magnitude." What is the justification for this conclusion? (We have already lost a portion or all of the ability to inject water into the RCS via the usual paths).

RESPONSE 67

End state DLOC represents sequences in which the interfacing LOCA has been terminated (isolated) and ECCS has been degraded but some core make up capability remains available. (see top of page 3-27). "The additional failures required to achieve core melt would lower this frequency by at least one order of magnitude" means that, given the degraded ECCS state, the unavailability of remaining mitigation equipment is better (lower) than 0.1.

RAI 68

The bottom of page 3-37 contains a statement to the effect that failure of one charging pump will lead to core melt. Why is this the case? Our perception is that sufficient flow might be provided by alternate means to keep the core covered, such as use of the remaining two charging pumps, and perhaps the ractor makeup water pumps.

RESPONSE 68

For the DILOC accident sequences discussed at the bottom of page 3-37, the failure of one charging pump would not be expected to lead to core melt. This is a conservative assumption to estimate the frequency of failures with the rest of the plant that would be needed to produce a core melt.

RAI 69, 70, 71, 72, 73

RAI 69

What is to be the status of the "temporary" 34.5 kV power lines which are identified on page 3-45?

RAI 70

What is to be the status of the mobile power supplies which are identified on page 3-46?

RAI 71

What capability has been provided to connect external pumps as identified in the second and third paragraphs of page 3-46? (This was briefly mentionsed on page 3-48). Use of a pump to simply inject water into containment via the sprays on a short term basis (no recirculation) does not appear to be identified. Has this been considered?

RAI 72

Page 3-46 identifies a number of possibilities for recovery of various safety functions. Are there specific plans? If so, please provide them.

RAI 73

There have been several references to purchase of a mobile electric generator by pooled resources on the pages prior to page 3-49. What is the likelihood that such a generator would be needed by several plants at the same time, and hence might not be available to Seabrook Station when needed? Similarly, where is the generator to be stored, and how is it to be transported to Seabrook? Include consideration of post seismic and post severe storm in the response.

RESPONSE 69, 70, 71, 72, 73

These question refer to section 3.2 which discusses Containment Recovery following an extended loss of all AC power. The various potential recovery measures discussed are related to recovery which could prevent late containment failures predicted to occur one to several days after the initiating event. These late containment failures, and therefore, the recovery measures discussed have no effect on early health risk. These recovery measures are not related to emergency planning decisions. Reducing the chance of late containment failure could impact latent health effects which are not sensitive to evacuation assumptions. Alternate ways of recovering late containment failures are still being evaluated.