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PRELIMINARY REVIEW AND EVALUATION OF THE MILLSTONE-3 PROBABILISTIC SAFETY STUDY

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ABSTRACT

A preliminary Level 1 review of the containment failure modes and consequence analysis in the Millstone-3 Probabilistic Safety Study (MPSS) is presented. The review identifies the major features of the plant as they relate to risk assessment, including comparisons to the Zion and Indian Point studies. Future plans and a list of preliminary questions is also included.

CONTENTS

CONTENTS (Cont.)

LIST OF FIGURES

LIST OF TABLES

1. INTRODUCTION

1.1 Background

After the accident at Three Mile Island, the Nuclear Regulatory Commission (NRC) recognized the need to reexamine the capabilities of nuclear power plants to accommodate the effects of hypothetical severe accidents beyond the design basis. This reexamination included consideration of potential design modifications to mitigate the consequences of these degraded and core melt accidents.

The Zion and Indian Point power plants were chosen to initiate this activity because of the large populations surrounding the two sites. The concern was that due to the proximity of these two sites to high population densities, they could comprise a disproportionately high component of the total societal risk from U.S. commercial nuclear power programs.

As part of this continuing effort, programs to evaluate the risk from plant sites situated near high population centers have been set in motion, in order to introduce design modifications and mitigation features, which can reduce the public risk.

Probabilistic Risk Assessment (PRA) studies have been undertaken by a number of utilities[1-3] and reviewed by Brookhaven National Laboratory (BNL) under contract to the NRC. BNL was also actively involved in preparation of a preliminary report[4] (NUREG-0850) which represented the staff's initial contribution to the understanding of severe accident progression and mitigation.

This report is a preliminary evaluation of the containment failure modes and consequence analysis of the Millstone Unit 3 Probabilistic Safety Study (MPSS) completed by Northeast Utilities in August 1983.[5]

1.2 Objectives

The objectives of this review report are to provide the NRC staff with a preliminary (Level 1) review of the MPSS as part of a broader objective involving an in-depth review and evaluation of the technical basis for the subject PRA, which will be performed in the coming year. In particular, core melt phenomenology, containment response, containment event trees, release categories, and site consequence models are to be examined.

This Level 1 review, which was performed over only four weeks duration, highlights important features of the plant design and the MPSS as compared to the PRAs of the Zion[1] and Indian Point[3] facilities. The report also provides an initial assessment of the PRA method, validity of major assumptions, and relevance and adequacy of conclusions.

Areas needing further verification and study are identified, and finally, questions for the applicant or Licensee pertaining to the Millstone Unit 3 are addressed.

1.3 Organization of the Report

A brief review of the Millstone-3 design and features is presented in Chapter 2 along with comparisons to Zion and Indian Point Plant designs.

Chapter 3 contains the preliminary assessment of the Millstone PRA. Specifically, analytical methods, containment event trees, accident phenomenology, containment matrix, uncertainty analysis, accident source terms, and off-site consequences are reviewed.

In Chapter 4 the results of this preliminary Level 1 review are summarized and areas needing further study are also highlighted along with need for additional information and questions to the applicant or the Licensee.

2. PLANT DESIGN AND FEATURES

In this section, those plant design features that may be important to an assessment of degraded core and containment analysis are reviewed. These important features are then compared with the Zion and Indian Point facilities in order to identify commonalities for benchmark comparisons.

2.1 Plant Design

Millstone-3 is a four-loop Pressurized Water Reactor (PWR). The core and reactor coolant systems are of the standard Westinghouse design, while the major balance of plant systems and the containment design are of Stone and Webster design.[5]

Major characteristics of the plant are a 3411 MWt (1150 MWe) core power reactor employing the Westinghouse 17 x 17 core design. The reactor coolant system is a four-loop configuration with U-tube recirculating steam generators. The emergency core cooling system consists of 4 accumulators containing 6358 gallons of water each, which are designed to discharge when the reactor coolant system pressure falls below 600 psia, a safety injection system which draws water from a 1.2 million gallon refueling storage tank and is delivered to the reactor coolant system via either the charging pumps, high head safety injection pumps or low head safety injection pumps. The long-term core cooling is handled via a completely independent recirculation cooling system (whose major components are shared with the recirculation spray system) which consists of four (4) pumps and four (4) heat exchangers which are cooled by the service water system.

The auxiliary feedwater system also provides a core cooling function by removing heat from the RCS after reactor shutdown via the steam generators. This system, which consists of two (2) 50 percent motor driven pumps and one

(1) 100 percent turbine driven pump takes suction from the condensate storage tank.

Containment cooling following an accident which initiates the ESF signal is accomplished via two completed independent spray systems. The quench spray system draws from the refueling water storage tank while the recirculation spray system draws from the containment sump (see Figure 2.1). Together, the systems are designed to reduce the pressure in the containment to a subatmospheric condition (normal operating state) within approximately one hour for design basis accident sequences.

The containment geometry design in the area underneath and around the reactor vessel precludes water from entering the reactor cavity area until a major portion of the Refueling Water Storage Tank (RWST) has been exhausted via the quench spray system (see Figure 2.2). This is referred to as a dry cavity configuration. The same geometry is expected to preclude the dispersion of core debris from the reactor cavity to the general containment area following postulated failure of the reactor vessel during core melt sequences. The cavity area geometry also would preclude the establishment of effective convective air currents between the cavity and general containment area for heat removal of core debris in the reactor cavity area. The containment design also includes a permanent seal ring between the reactor vessel flange and the biological shield walls, which would prevent introduction of water into the reactor cavity from either break flow or spray flow in the area of the reactor vessel or the refueling cavity. The containment building basemat and the internal concrete structures are composed of basaltic-based concrete.

2.2 Comparison to Zion and Indian Point Plant Designs

Table 2.1 sets forth the design characteristics of the Zion (Units 1 or 2) and the Indian Point (Unit 2) facilities as they compare to the Millstone Unit 3 plant.

It is seen that the three plants are quite similar in containment building and primary system design while they differ markedly in containment cooling mechanisms and lower reactor cavity configuration and chemical compositions of the concrete mix.

As concrete is heated, water vapor and other gases are released. The initial gas release consists largely of carbon dioxide, the quantity of which depends on the amount of calcium carbonate in the concrete mix. Limestone concrete can contain up to 80% calcium carbonate by weight, which could yield up to 53 lb of carbon dioxide per cubic foot of concrete. However, basalt-based concrete contains very little calcium carbonate and would not release a significant amount of carbon dioxide.[4]

These inherent design differences are expected to alter the course of the accident sequences; in particular, following failure of the reactor vessel, where the containment pressurization is significantly influenced by the debris bed coolability and water availability.

The absence of fan coolers in the Millstone plant can also effect the accident progression and containment pressurization effects.

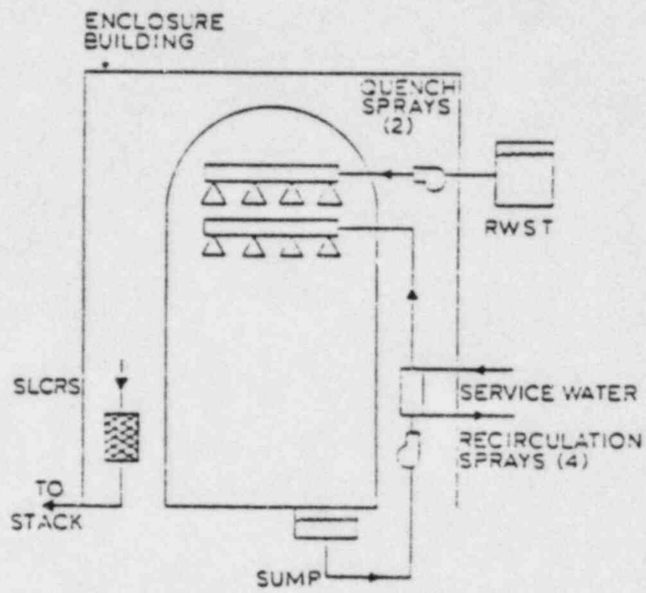

Figure 2.1 Schematics of the containment cooling systems in the Millstone-3

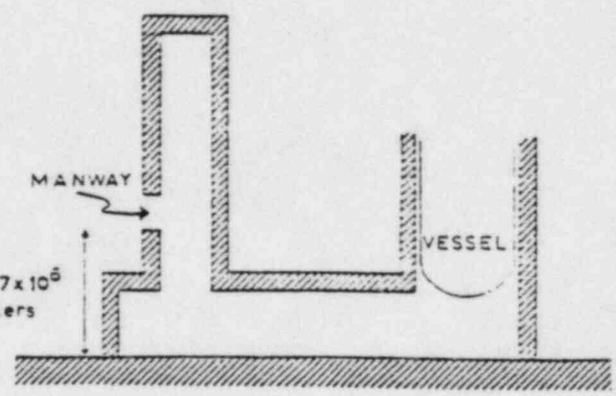

Figure 2.2 Schematics of the lower reactor cavity in the Millstone-3

Table 2.1 Comparison of design characteristics

Design Parameters		Zion Unit 1[4]	Indian Point Unit 2[3,4]	Millstone Unit 3[5]
Reactor Power	[MW(t)]	3250	3030	3411
CONTAINMENT BUILDING:				
Free Volume	(ft^3)	2.72×10^6	2.61×10^6	2.3×10^6
Design Pressure	(psia)	62	62	59.7
Initial Pressure	(psia)	15	14.7	12.7/9.1
Initial Temperature	(°F)	120	120	120/80
PRIMARY SYSTEM:				
Water Volume	(ft^3)	12,710	11,347	11,671
Steam Volume	(ft^3)	720	720	?
Mass of UO_2 in Core	(lb)	216600	216600	222739
Mass of Steel in Core	(lb)	21,000	20,407	?
Mass of Zr in Core	(lb)	44,500	44,600	45,296
Mass of Bottom Head	(lb)	87,000	78,130	87,000
Bottom Head Diameter	(ft)	14.4	14.7	14.4
Bottom Head Thickness	(ft)	0.45	0.44	0.45

Table 2.1 Comparison of design characteristics

(Continued)

Design Parameters		Zion Unit 1	Indian Point Unit 2	Millstone Unit 3
RESIDUAL HEAT REMOVAL EXCHANGERS (HX):				
Total Rated Capacity	(Btu/hr)	5.6×10^7 (2HX)	6.16×10^7 (2HX)	7.05×10^7 (2HX)
Total Primary Flow	(lb/hr)	3.9×10^6	2.88×10^6	1.98×10^6
Total Secondary Flow	(lb/hr)	4.96×10^6	4.92×10^6	3.3×10^6
Primay Inlet Temperature	(°F)	137.5	135	120
Secondary Inlet Temperature	(°F)	107.1	88.3	92.2
CONTAINMENT BUILDING COOLERS:				
System		Fans	Fans	Sprays
Number		5	5	(2 Quench 2 Recirculation)
ACCUMULATOR TANKS:				
Total Mass of Water	(lb)	200,000	173,000	348,000
Initial Pressure	(psia)	665	665	665
Temperature	(°F)	150	150	80

Table 2.1 Comparison of design characteristics

(Continued)

Design Parameters		Zion Unit 1	Indian Point Unit 2	Millstone Unit 3
REFUELING WATER STORAGE TANK:				
Total Mass of Water	(lb)	2.89×10^6	2.89×10^6	10^7
Initial Pressure	(psia)	14.7	14.7	12.7/9.1
Temperature	(°F)	100	120	50/40
REACTOR CAVITY:				
Design		Dry	Dry	Dry
Concrete Material		Limestone	Basaltic	Basaltic

3. PRA REVIEW

In this section a brief review of the Millstone Unit 3 Probabilistic Safety Study (MPSS) is presented. Specifically, the computer codes and calculational methods used to carry-out the degraded core and containment response analyses are identified. Where possible, parallels between this study and other existing PRA studies are set forth. Finally, the relevance and validity of the conclusions is addressed.

3.1 Analytical Methods

A brief description of the computer codes used to perform the transient degraded core and containment response analyses is provided in this section.

Table 3.1 summarizes the code package as applied to various phases of the accident. It is seen that the MARCH code is used to model the core and primary system behavior and to obtain the steam and water energy releases for (1) the entire transient in the case of non-dispersal accident events and (2) until the vessel failure for the dispersal scenarios. These mass and energy releases form the input for the other computer codes used to evaluate the containment response for the non-dispersal cases (see also Figure 3.1).

For sequence classes in which the reactor coolant system remains at an elevated pressure until the vessel failure (dispersal cases), the MODMESH code is used. This code calculates the steam and hydrogen blowdown from the reactor vessel using an isothermal ideal gas model. The water boil-off from the reactor cavity floor is modeled using a saturated critical heat flux correlation. Additionally, the accumulator discharge following depressurization caused by the vessel failure is also considered.

For the non-coolable debris bed and core-concrete interaction, the INTER subroutine of MARCH is replaced by the CORCON-MOD1 code modified by Westinghouse.

The output from MARCH or CORCON is used as input, after preprocessing by MODMESH, to the COCOCLASS9 code. The COCOCLASS9 code replaces the MACE subroutine of the MARCH code. In COCOCLASS9 code, the containment steam/water, noncondensibles, and the sump water are modeled by a single volume. The code also includes a structural heat transfer model, hydrogen combustion, and capability for containment heat removal through containment sprays and sump recirculation systems, as described in Section 4.3.2 and Appendix 4-E of the MPSS report.[5]

Fission product transport and consequence calculations are performed using CORRAL-II and CRAC-2 computer codes, respectively. (See Section 3.6 and 3.7 for more details.)

This preliminary review of the approach used in the MPSS for quantification of core and containment response is directed to a review of the consistency of the approach. However, detailed verification of the results obtained in the MPSS cannot be made at this stage, and is thus deferred to a later date.

3.2 Containment Event Tree and Accident Phenomenology

An important step towards the development of the containment matrix involves the quantification of branch point probabilities in the containment event trees. The probabilities depend heavily on the analyses of degraded core phenomenology and the containment building response described in Sections 4.2 through 4.7 of the MPSS.[5]

In the MPSS[5], the containment event tree is divided into six distinct time frames, which represent the time phases during an accident event in which potential containment failure is considered. Table 3.2 summarizes the six time frames along with the corresponding containment event tree nodal questions, as reproduced from the MPSS.[5]

Detailed assessment of the nodal questions and the assigned probabilities cannot be made at this time; however, a preliminary evaluation of the nodal questions as they relate to Zion (ZPSS[1]) and NUREG-0850[4] is made in Table 3.3.

A node by node comparison between the MPSS and ZPSS is not possible because of the differences in the plant designs and containment event tree structures. However, in arriving at nodal probabilities, significant credit has been taken for:

1. Core-wide incoherencies during meltdown progression as attributed to the recent TMI-2 heat-up calculations, also identified for ZPSS.[1]
2. Reduced energetics, as a result of in-vessel core debris-water interaction, leads to low probability events early in the accident.
3. Successful quenching of debris bed as a result of high pressure discharges following vessel failure.

Therefore, it is essential to review the event tree structure, and its associated nodal questions and quantifications as they affect the overall risk before a complete assessment could be made.

3.3 Containment Matrix (C-Matrix)

The sixteen nodes in the Millstone-3 containment event trees were outlined in the previous section. A negative response at any of seven nodes (CI1, CI2, CI3, CI4, CI5, CI6, and BM6) in the containment event trees result in failure of the containment building by a variety of failure modes. Each of these failure modes results in a particular radiological release category. For those paths that do not have a negative response at any of the seven nodes, the path will eventually result in <u>no failure</u> of the containment. The containment event trees, therefore, link damage states to a range of possible containment failure modes via the various paths through the tree. For a given tree, each path ends in a conditional probability (CP) of occurrence and these CPs should sum to unity. The quantification of an event tree is the process by which all the paths are combined to give the conditional probabilities of the various release categories. In the MPSS, thirteen release categories were used for the quantification process as summarized in Table 3.4. Note that one of these release categories (namely, M12) correspond to <u>no</u> containment failure. Fission product release for this category would, therefore, be via normal leakage paths in the containment building.

The quantification of the MPSS containment event trees was a significant task, and it was necessary to use a computer code, ARBRE, to group the various path probabilities into the thirteen release categories.[5] However, the containment matrix 'C' is a concise summary of the quantification process.

Table 3.5 is a reproduction of the 'C' matrix for the MPSS.[5] It lists the conditional probabilities of the release categories given the plant damage state; with the plant damage states defined in Table 3.6.

A simplification to the C-Matrix is obtained in Table 3.7 by disregarding all of the very low probability values. This simplification is not expected to influence the risk calculations as discussed in [6].

The comparable simplified C-Matrix for ZPSS is reproduced in Table 3.8 while Table 3.9 lists the ZPSS C-Matrix as determined by BNL calculations.[4]

Due to the distinct differences in plant designs and progression of the accidents, an exact correspondence in release categories of ZPSS and MPSS cannot be made. However, similarities in the release categories are identified in Tables 3.8 and 3.9.

3.3.1 Release Category M1A

The conditional probability and plant damage states are identical in the three C-matrices.

3.3.2 Release Category M1B

Unique to MPSS and not identified as a failure mode in ZPSS.

3.3.3 Release Category M2

This release category was identified in ZPSS, but neither the BNL study nor MPSS results seem to indicate M2 as a significant contributor.

3.3.4 Release Categories M3 and M4

These failure modes were found to be insignificant for both plants.

3.3.5 Release Category M5

Given the plant damage state SL, the probability of this release category is calculated to be about 0.01 in MPSS and it was found to be insignificant in the other studies.

3.3.6 Release Category M6

This category is found to be only significant for MPSS.

3.3.7 Release Category M7

This release category applies to plant damage states with insufficient or no containment heat removal systems operating. The relatively smaller probabilities calculated for plant damage states AE, AL, SE, and SL are associated with the difference in cavity concrete structure. In Zion, the limestone concrete with high calcium carbonate content causes high CO_2/CO releases, and thus higher containment pressures.

3.3.8 Release Category M8

This failure mode was found to be insignificant in ZPSS, MPSS, and the BNL study. However, in the BNL study for Zion,[6] this release category was found to influence the overall risk calculations, and thus it needs to be further assessed for MPSS.

3.3.9 Release Category M9

This failure mode applies to plant damage states AEC' and SEC' which are LOCA's with early core melt in the absence of recirculation spray system in MPSS; however, this was not found to be a significant contributor in the Zion plant primarily, due to smaller amounts of the water available leading to less steam generation and overpressurization.

3.3.10 Release Categories M10 and M11

This failure mode could potentially impact all damage states. For plant states without containment heat removal and early core melt (namely, loss-of-ECC injection), there would be limited water in the reactor cavity and thus a potential for basemat penetration. However, containment failure would occur prior to basemat penetration and thus a higher probability is associated for release category M7 for three damage states. All other damage categories will have significant quantities of water in the reactor cavity.

It must be noted that impact of basemat penetration on risk is believed to be negligible and thus this failure mode can be neglected.

In general, it is found that containment integrity in Millstone Unit 3 can be assured only if both the containment recirculation spray and quench spray systems are available. Of the two, however, the long-term heat removal capability of recirculation spray system is more important. In all instances, hydrogen generation by molten-core-concrete-interaction and likelihood of hydrogen burns were found to be high.[5]

3.4 External Events

In MPSS, containment response to accidents initiated by external events (fires and seismic events) are also considered.

The external containment event trees make use of the same event tree structure as is used for the internal initiating events. The impact of differences in event sequence course is accounted for in the assignment of the split fractions and uncertainty assignments.

Containment thermal response analyses were not performed for externally initiated events, but rather, engineering judgment has been used in the assignment of each accident sequence to a particular release category as described in Section 4.7.5 of MPSS.[5]

3.5 Uncertainty in the C-Matrix

The containment event tree quantification described in the earlier section was based on the assessment of point-estimate probabilities for the split fractions at various nodes of the containment event tree. In order to account for inherent uncertainties associated with phenomenological questions, the Discrete Probability Distribution (DPD) methodology was implemented in MPSS.[5]

The DPD is described in Section 4.7.4 and Appendix 4-N of the MPSS. The distributions are constructed based upon the following criterion:

1) Definition of a reasonable upper and lower range of the nodal probabilities which represent an upper and lower 95 percent one-sided confidence level.

2) A weighting factor for the point-estimate, upper and lower range values resulting in a three-interval DPD.

These DPD's are then propagated through the dominant paths of the containment event trees using DPD arithmetic.

The MPSS results for the containment uncertainty analysis show that the range of uncertainty is low for the more probable sequences (higher point-estimates) as compared to the less probable sequences.

3.6 Accident Source Terms

In this section the approach utilized to determine the fraction of fission products originally in the core and leaked to the outside environment will be outlined. The fission product source to the environment as calculated by this approach will be compared with those for similar plants. The calculations to be included in this comparison are those done for the Zion and Indian Point Probabilistic Risk Assessments, (ZPSS[1] and IPPSS[3], respectively), the Indian Point Study carried out for the NRC and presented as testimony at the Indian Point hearings (IPS)[7], and finally, releases determined for the Surry plant using the methods proposed by the Accident Source Term Program Office (ASTPO). The first three calculations are based on the methods used in the Reactor Safety Study (RSS) and published as WASH-1400;[8] the last calculation is based on more mechanistic methods which form the basis of the revised source term and is published as BMI-2104 Volume 1.[9]

As in the RSS, the CORRAL-II code is the most important tool for determining the fission product leakage to the environment. This code takes input from the thermal-hydraulic analysis carried out for the containment atmosphere. In addition, it needs the time dependent emission of fission products. The fission product release is divided up into the customarily used phases, i.e., Gap, Melt, and Vaporization releases. The time dependence of these phases is determined by the core heatup, primary system failure and core/concrete interaction times. In all, thirteen releases were determined ranging from the containment bypass sequence (V-sequence) to the no fail sequence. These sequences are summarized in Section 3.3 and the results are shown on Table 3-10.

Four of the thirteen releases outlined above are based on the RSS releases. M-1A and PWR-2, M-10 and PWR-6, and, M-11 and PWR-7 are all identical in both fractional release and timing. The release M-1B, which corresponds to a steam generator tube rupture, is determined by dividing PWR-2 or M-1A by ten. Noble gases and organic iodine are not subject to this reduction in release.

The energy of release for the overpressurization failures (M-2, M-3, M-4, M-5, and M-7) are high in comparison to the corresponding releases reported in the RSS. The effect of a high plume energy is that the plume is lifted higher into the atmosphere and in most cases the fission products are spread over a larger area. The net effect of this is that potentially more people will be exposed, however, the concentration of the dose will be lower. Thus, any consequence which is characterized by a dose threshold could be lowered by increasing the plume energy. This general trend could be affected very strongly by population distribution, the protective action taken, and weather.

Finally, a release path customarily included in Probabilistic Risk Assessment is missing from the list. This is a release characterized by a steam explosion. It is not clear how much it would contribute to risk. Steam explosions are very remote, and the early overpressurization failure release, M-2, has a similar release character as a steam explosion, except the Ruthenium (Ru) and Tellurium (Te) releases are low compared to the customarily assumed release fractions for a steam explosion.

In order to allow for uncertainties in the transport of fission products, a similar method using Discrete Probability Distributions (DPD's) described earlier in Section 3.5 was used. This method assumes that the RSS method of determining fission product release fractions, has a relatively large uncertainty associated with it, and thus at major steps in the sequence probability distributions are assigned to the phenomena taking place. These distributions are combined to give a probability distribution for each release category.

There are three major phenomena in the transport of fission products from the core to the environment where there is considerable uncertainty surrounding the attenuating mechanisms. These phenomena are:

a) Deposition and holdup of fission products in the primary system,

b) Agglomeration and settling of aerosols in the containment building, and

c) Attenuation of fission products as they pass out of the building into the environment.

Of the three phenomena mentioned above, only the first two are considered in the MPSS.[5]

In the MPSS, the major releases are broken down into their contributing plant damage states, and distributions are assigned to the above-mentioned phenomena as they occur in each damage state. These distributions are then combined to form a damage state distribution, which are further combined with other damage state distributions to result in a DPD appropriate for a fission product release. These distributions are a measure of the uncertainty of a particular release. They also give an indication of how different the RSS based release is from the best-estimate release.

Tables 3-11 through 3-15 show comparisons between selected releases computed for the MPSS, the ZPSS,[1] IPSS,[3] IPS,[7] and those determined for the ASTPO.[9] The first four analyses all used similar methods and were based on methods outlined in the RSS. The determinations for ASTPO were based on more mechanistic methods and used an improved and extended data base.

Table 3-11 shows the release fractions, timing, and uncertainty vector and associated probabilities for the interfacing LOCA (V-sequence). Comparing the first two columns (MPSS release and ASTPO release), it is seen that besides the noble gas release fraction, there is a substantial reduction in fission product release when using the ASTPO methods. This is particularly true for the Ruthenium (Ru) group which shows a reduction of approximately 25; the Tellurium (Te) and Barium (Ba) groups are reduced by 6. The uncertainty multiplier used in the MPSS peaks at a 50% reduction in the release fractions. Furthermore, it has a 17% weight for no reduction and 28% weight for a reduction by a factor of four. A comparison with the last column, which corresponds to the IPS[7] analysis shows that the fission product releases are comparable with the RSS determined release fractions. However, the energy of the release is much higher in the MPSS case, the ratio being 40.

Table 3-12 shows a comparison between the MPSS, ASTPO, and ZPSS for the early overpressurization release. It is seen that the M-2 release, which corresponds to the MPSS analysis, is equal to or lower than the ASTPO release fractions for the volatile fission product groups (Xe-Kr-Te-Sb). However, for the less volatile fission products (Ba-Sr-La) the MPSS releases are higher than the ASTPO releases. The uncertainty factor for this release is seen to imply a substantial reduction in fission product release fraction. In comparison to the ZPSS release fractions, very small differences are seen. Furthermore, the uncertainty factors for the ZPSS also imply a reduction in release, but to as large as for the MPSS. Finally, the energy of the release for the MPSS is comparable to the energy of release for the ZPSS.

Table 3-13 compares the releases for an isolation failure. M-4, the release determined for the MPSS, is compared to two releases determined for the IPS.[7] The release fractions determined for the IPS corresponded to leakage through openings 8" and 4" in diameter, while the M-4 release corresponds to leakage through a 6" diameter hole. It is seen that the M-4 release is comparable to that corresponding to the 8" diameter hole determined for the IPS. The uncertainty multiplier for M-4 is seen to peak at a 50% reduction in the release fractions.

Table 3-14 shows a comparison between MPSS releases M-5 and M-7 with comparable releases determined for the ZPSS,[1] IPPSS,[3] IPS,[7] and the ASTPO.[9] These releases all correspond to intermediate or late (8 hr-20 hr) overpressurization failures. It is seen that the fractional releases computed for M-5, M-7, ZPSS, IPSS, and IPS are all similar. The largest discrepancy occurring for iodine, which is much higher for the ZPSS than all the others. However, in comparing these releases with the ASTPO determined releases, it is seen that the ASTPO releases are substantially lower, except for noble gases which are comparable. This large reduction is attributable to the improved modeling of attenuating mechanisms in the ASTPO method. It is seen from the uncertainty vector that a large weight is placed on reducing the source terms, particularly for M-5 and M-7. There is also a large difference in the energy of the released plume for the MPSS results and the remaining calculations. The MPSS energy being four times higher than the IPS result and approximately twenty-five times higher than the ZPSS and IPPSS results.

Table 3-15 shows a comparison between the M-12 release from the MPSS, the no-fail release from the IPS[7] and the no fail or 8B release used in the ZPSS[1] and the IPPSS.[3] These sequences are all based on a leak rate of

1%/day. It is seen that the M-12 release fractions are comparable to the ZPSS and IPPSS release fractions. However, they are all substantially lower in comparison to the IPS release fractions. Since all the fission product release fractions are quite low for this sequence, the differences between them, measured in terms of consequences, will be small.

3.7 Off-Site Consequence Analysis

The off-site consequences due to airborne fission products were determined using the CRAC2 code. The output of these calculations is in the form of conditional cumulative probability distribution functions (CCDF's) which form the basis of the S-matrix. The S-matrix is used to determine the overall risk of the plant. The CRAC2 code requires input for site specific data (population distribution, economic parameters, topographical), health data (dose conversion factors for latent and early consequences), meteorological data (wind rose, wind speed, rain), plume characteristics (isotopic content, physical description), and population response (evacuation parameters). Of all the data which is outlined above, we will discuss only the plume characterization, as it affects the uncertainty of the analysis through the fission product release fraction multipliers discussed above, and the evacuation model input.

3.7.1 Evacuation Model

A summary of the evacuation schemes is shown on Table 3-16. The evacuation schemes used in this study are divided into two categories, i.e, general and seismic. The general scheme is represented by two schemes, depending on the weather. Normal weather is expected to occur 88% of the time and adverse weather occurs the remaining 12%. Thus, these two schemes are weighted in the ratio of .88: .12. The normal evacuation scheme allows for a speed of 10 mph and a delay time of .92 hr, while the one corresponding to adverse weather reduces the speed to 7.5 mph. In comparison, the parameters used in the IPS[7] for these parameters are 1.5 mph and 2 hr, respectively. This is a large difference, considering the sensitivity of early health effects to these two parameters.

In the case of the evacuation scheme corresponding to seismic conditions in the MPSS, the speed is reduced to 2 mph and the delay time increased to 3.38 hr. In comparison, in the IPS,[7] no evacuation was allowed following a seismic event. Furthermore, the plume shielding factors were set to unity (no shielding) since the population may be outside (destroyed dwellings).

The remaining evacuation schemes (S1-S6) are a series of evacuation schemes used for releases M-1A and M-4, since they were determined to be particularly sensitive to evacuation schemes. Evacuation speed and delay times were varied between 1.2 mph and 10 mph, and the delay time was varied between .92 hr and 2 hr. A probability associated with each of these schemes is shown in the bottom row. It is seen that the general scheme with normal weather has approximately a 40% weight. All these probabilities were determined by subjective judgment.

3.7.2 Plume Characteristics and Uncertainty

A rigorous treatment of uncertainty of this analysis is not practical. In the MPSS, it is pointed out that some of the data and/or models are either in state-of-the-art or because of their firmness, have relatively small bands of uncertainty. The overall uncertainty from the CRAC2 code will thus only be treated as an uncertainty in the dose delivered to an individual. This uncertainty will be measured by changing the isotopic content of the plume. In a manner similar to that used above to estimate uncertainty associated with primary system holdup and containment building attenuation, a DPD is defined which characterizes the uncertainty in dose delivered to an individual. Table 3-17 shows the variation in source multiplication and appropriate probability. It is seen that a small probability of doubling the source exists. However, the largest probability indicates that the source will either stay the same or will be halved (80%). A reduction by an order of magnitude is also given a small probability.

This DPD is then folded into the individual release DPD's outlined above and results in an overall DPD for each release. It is seen that the overall DPD will now have a finite probability of doubling the source strength in selected cases. Table 3.18 shows a list of the CRAC2 calculations which were carried out. It is interesting to note that the releases M-8 through M-12 were all carried out with a multiplication of unity, thus ignoring any uncertainty in the CRAC2 calculation. These are, however, low consequence sequences and ignoring the uncertainty will have small effect on the overall risk. The overall uncertainty determination is made by carrying out CRAC2 calculations for the various fission product sources appropriately modified by the multiplication factors and weighted by the corresponding probabilities.

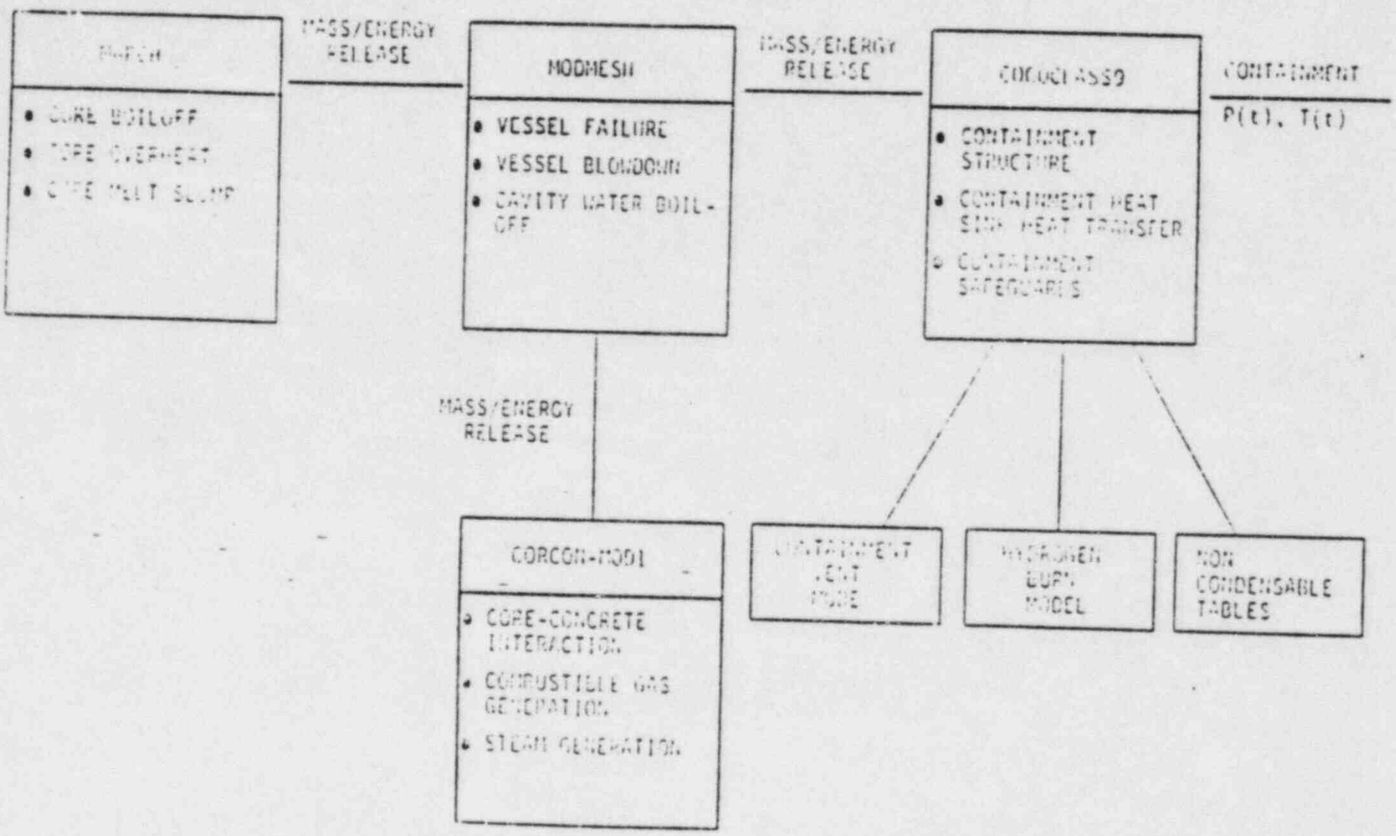

Figure 3.1 The MPSS computational approach.

Table 3.1 Summary of computational tools

Computer Code	Accident Phase
MARCH	1. Non-dispersal Events - Total transient 2. Dispersal Events - Initial blowdown, slump, and vessel failure
MODMESH	1. Non-dispersal Events - Interface to other codes 2. Dispersal Events - Discharge and scatter, cavity boil-off
CORCON-MOD1/W	Core-concrete interaction for dry cavity
COCOCLASS 9	Containment building pressurization and hydrogen combustion
CORRAL-II	Fission product transport
CRAC2	Consequences

Table 3.2 Summary of containment event tree time frames and nodal questions[5]

Time Frame I:	Accident Initiation $\leq t <$ Core Degradation		
	CI1	-	Is the containment intact?
Time Frame II:	Core Degradation $\leq t <$ Significant Debris Accumulation in Lower Plenum		
	NB2	-	Does the hydrogen not burn?
	CI2	-	Does the containment remain intact?
Time Frame III:	Significant Debris Accumulation $\leq t <$ Vessel Failure in Lower Plenum		
	CD	-	Is the core melt incoherent?
	NB3	-	Does the hydrogen not burn?
	CI3	-	Does the containment remain intact?
Time Frame IV:	Vessel Failure $\leq t \leq$ Complete Depressurization		
	QUE	-	Is the core debris quenched?
	NB4	-	Does the hydrogen not burn?
	CI4	-	Does the containment remain intact?
Time Frame V:	Complete Depressurization $< t \leq$ 4 Hr* After Vessel Failure		
	CD5	-	Is the debris coolable?
	NB5	-	Does the hydrogen not burn?
	CI5	-	Does the containment remain intact?
Time Frame VI:	4 Hr After Vessel Failure $< t \leq$ One Day		
	CD6	-	Is the debris coolable?
	NB6	-	Does the hydrogen not burn?
	CI6	-	Does the containment remain intact?
	BM6	-	Does the basemat remain intact?

*It should take 4 hr to boil-off the accumulator water from the lower reactor cavity.[5]

Table 3.3 Assessment of MPSS event tree nodal questions

Node	Evaluation
CI1	Identical to Node A of ZPSS and in agreement with NUREG-0850
NB2	Similar to Node B of ZPSS and differences with NUREG-0850 will not affect the probability value
CI2	Similar to Node C of ZPSS and the assessment is appropriate
CD	Identical to Node D of ZPSS; however, large incoherencies assumed as compared to NUREG-0850, and NUREG/CR-3300 comments apply
NB3	Identical to Node F of ZPSS. H_2 generation is equivalent to 20% Zr/H_2O reaction compared with 100% in NUREG-0850
CI3	Similar to Node E of ZPSS and the probabilities seem to be reasonable
QUE	The high probability assigned to the high pressure discharges (small breaks and transients) need to be assessed in light of recent experimental measurements
NB4	Adequate, except for the core and no quench where further assessment is needed
CI4	The arguments seem to be valid; however, further calculations are needed to ascertain the assigned probability with maximum uncertainties regarding H_2
CD5	The probabilities assigned to the nodes need to be verified in light of large phenomenological uncertainties on debris bed coolability
NB5	Similar to Node (O) of ZPSS, and the assessment is reasonable
CI5	Due to the strong dependence of the containment failure probability on pressure, an audit calculation is needed to confirm the pressure peaks
CD6	Same as CD5, where the success probability needs to be verified
NB6	Reasonable
CI6	Similar to Node (R) of ZPSS and in agreement with NUREG-0850
BM6	Adequate

Table 3.4 Notation and definitions for release categories

Release Category	Description
M1A	Containment Bypass, V-Sequence
M1B	Containment Bypass, SGTR
M2	Early Failure/Early Melt, No Sprays
M3	Early Failure/Late Melt, No Sprays
M4	Containment Isolation Failure
M5	Intermediate Failure/Late Melt, No Sprays
M6	Intermediate Failure/Early Melt, No Sprays
M7	Late Failure, No Sprays
M8	Intermediate Failure With Sprays
M9	Late Failure With Sprays
M10	Basemat Failure, No Sprays
M11	Basemat Failure With Sprays
M12	No Containment Failure

Table 3.5 C-matrix internal initiating events

Category Damage State	M1A	M1B	M2	M3	M4	M5	M6	M7	M8	M9	M10	M11	M12
AE	----	----	2.87(-4)	----	2.00(-4)	----	6.16(-1)	2.90(-1)	----	----	9.34(-2)	----	[illegible]
AEC	----	----	2.90(-4)	----	2.00(-4)	----	----	----	6.37(-4)	1.18(-4)	----	4.99(-2)	9.48(-1)
AEC'	----	----	2.40(-4)	----	2.00(-4)	----	----	----	6.77(-4)	9.98(-1)	----	9.77(-4)	2.16(-5)
AEC"	----	----	2.90(-4)	----	2.00(-4)	----	----	----	6.63(-4)	1.19(-4)	----	9.89(-1)	1.01(-2)
AL	----	----	3.07(-4)	----	2.00(-4)	----	5.38(-1)	3.49(-1)	----	----	1.13(-1)	----	1.59(-5)
ALC	----	----	2.90(-4)	----	2.00(-4)	----	----	----	6.37(-4)	1.18(-4)	----	4.99(-2)	9.48(-1)
ALC'	----	----	2.40(-4)	----	2.00(-4)	----	----	----	6.77(-4)	9.98(-1)	----	9.77(-4)	2.16(-5)
ALC"	----	----	2.90(-4)	----	2.00(-4)	----	----	----	6.63(-4)	1.19(-4)	----	9.89(-1)	1.01(-2)
SE	----	----	1.15(-3)	----	2.00(-4)	----	6.17(-2)	8.91(-1)	----	----	4.63(-1)	----	1.40(-5)
SEC	----	----	2.90(-4)	----	2.00(-4)	----	----	----	6.37(-4)	1.18(-4)	----	4.99(-2)	9.48(-1)
SEC'	----	----	2.40(-4)	----	2.00(-4)	----	----	----	6.77(-4)	9.98(-1)	----	9.77(-4)	2.16(-5)
SEC"	----	----	2.90(-4)	----	2.00(-4)	----	----	----	6.63(-4)	1.19(-4)	----	9.89(-1)	1.01(-2)
SL	----	----	----	2.29(-4)	2.00(-4)	9.48(-3)	----	7.95(-1)	----	----	1.96(-1)	----	2.94(-5)
SLC	----	----	----	2.90(-4)	2.00(-4)	----	----	----	6.37(-4)	1.18(-4)	----	4.99(-2)	9.48(-1)
SLC'	----	----	----	2.90(-4)	2.00(-4)	6.39(-1)	----	9.98(-1)	----	----	----	9.64(-4)	[illegible]
SLC"	----	----	----	2.90(-4)	2.00(-4)	----	----	----	6.63(-4)	1.91(-4)	----	9.89(-1)	[illegible]
S'E	----	----	2.14(-4)	----	2.00(-4)	----	1.10(-3)	9.89(-1)	----	----	9.88(-3)	----	1.10(-5)
S'L	----	----	----	2.14(-4)	2.00(-4)	1.10(-3)	----	9.89(-1)	----	----	9.88(-3)	----	1.10(-5)
TE	----	1.90(-4)	----	1.01(-4)	2.00(-4)	4.36(-3)	----	8.97(-1)	----	----	9.83(-2)	----	1.98(-5)
TEC	----	1.90(-4)	----	9.99(-5)	2.00(-4)	----	----	----	6.37(-4)	1.18(-4)	----	4.99(-2)	9.48(-1)
TEC'	----	1.90(-4)	----	9.99(-5)	2.00(-4)	6.38(-4)	----	9.98(-1)	----	----	----	9.64(-4)	3.51(-5)
TEC"	----	1.90(-4)	----	9.99(-5)	2.00(-4)	----	----	----	6.63(-4)	1.19(-4)	----	9.89(-1)	1.01(-2)
V	1.0	----	----	----	----	----	----	----	----	----	----	----	----
V2	----	1.0	----	----	----	----	----	----	----	----	----	----	----

Table 3.6 Notation and definitions for plant states (internal)

Symbol	Description
AEC	Large LOCA, Early Melt
AEC'	Large LOCA, Early Melt, Failure of Recirculation Spray
AE	Large LOCA, Early Melt, No Containment Cooling
ALC	Large LOCA, Late Melt
ALC'	Large LOCA, Late Melt, Failure of Recirculation Spray
ALC"	Large LOCA, Late Melt, Failure of Quench Spray
AL	Large LOCA, Late Melt, No Containment Cooling
SEC	Small LOCA, Early Melt
SEC'	Small LOCA, Early Melt, Failure of Recirculation Spray
SE	Small LOCA, Early Melt, No Containment Cooling
S'E	In-Core Instrument Tube LOCA, Early Melt, No Containment Cooling
SLC	Small LOCA, Late Melt
SLC'	Small LOCA, Late Melt, Failure of Recirculation Spray
SLC"	Small LOCA, Late Melt, Failure of Quench Spray
SL	Small LOCA, Late Melt, No Containment Cooling
S'L	In-Core Instrument Tube LOCA, Late Melt, No Containment Cooling
TEC	Transient, Early Melt
TEC'	Transient, Early Melt, Failure of Recirculation Spray
TE	Transient, Early Melt, No Containment Cooling
V2EC	Steam Generator Tube Rupture, Steam Leak, Early Melt
V2EC'	SGTR, Steam Leak, Early Melt, Failure of Recirculation Spray
V2E	SGTR, Steam Leak, Early Melt, No Containment Cooling
V2LC	SGTR, Steam Leak, Late Melt
V2LC'	SGTR, Steam Leak, Late Melt, Failure of Recirculation Spray
V2LC"	SGTR, Steam Leak, Late Melt, Failure of Quench Spray
V2L	SGTR, Steam Leak, Late Melt, No Containment Cooling
V	Interfacing Systems LOCA

Table 3.7 Simplified 'C' matrix for MPSS

Plant State	M1A	M1B	M5	M6	M7	M9	M10	M11	M12
AE				0.62	0.29		0.09		
AEC							0.05		0.95
AEC'						1.0			
AEC"								0.99	0.01
AL				0.54	0.35		0.11		
ALC								0.05	0.95
ALC'							1.0		
ALC"								0.99	0.01
SE				0.06	0.89		0.05		
SEC								0.05	0.95
SEC'						1.0			
SEC"								0.99	0.01
SL			0.01		0.79		0.20		
SLC								0.05	0.95
SLC'					1.0				
SLC"								0.99	0.01
S'E					0.99		0.01		
S'L					0.99		0.01		
TE					0.90		0.10		
TEC								0.05	0.9
TEC'					1.0				
TEC"								0.99	0.0
V	1.0								
V2		1.0							

Table 3.8 Simplified ZPSS containment matrix C

Plant State	Release Category				
	Z-1(M2)	2(M1A)	2R(M7)	8A(M12)	8B(M12)
SEFC					1.0
SEF					
SEC					1.0
SE			1.0		
SLFC					1.0
SLF				1.0	
SLC					1.0
SL			1.0		
TEFC					1.0
TEF				1.0	
TEC					1.0
TE			1.0		
AEFC					1.0
AEF				1.0	
AEC					1.0
AE	1.0				
ALFC					1.0
ALF				1.0	
ALC					1.0
AL	1.0				
VE		1.0			

Table 3.9 Simplified BNL containment matrix C for ZPSS

Plant State	Release Category						
	2(M1A)	2R(M7)	Z-5(M8)	6(M10)	7(M11)	8A(M12)	8B(M12)
SEFC			0.02		.1		0.88
SEF	0.4			.1		0.5	
SEC			0.02				0.88
SE		1.0					
SLFC			0.01		.1		0.89
SLF	0.01			.1		0.89	
SLC			0.01		.1		0.89
SL		1.0					
TEFC			0.02		.1		0.88
TEF	0.4			.1		0.5	
TEC			0.02		.1		0.88
TE		1.0					
AEFC			0.02		.1		0.88
AEF	0.4			.1		0.5	
AEC			0.02		.1		0.88
AE		1.0					
ALFC			0.01		.1		0.89
ALF	0.01			.1		0.89	
ALC			0.01		.1		0.89
AL		1.0					
VE	1.0						

Table 3.10 Release category summary

Category	Release Start Time (hrs)	Release Warning Time (hrs)	Release Duration (hrs)	Release Energy (Btu/hr)	Point Estimate Source Term Values* Xe-Kr	OI	I-Br	Cs-Rb	Te-Sb	Ba-Sr	Ru	La
M-1A	2.5	1.0	1.0	20 E6	0.9	7E-3	0.7	0.5	0.3	6E-2	2E-2	4E-3
M-1B	2.5	1.0	1.0	20 E6	0.9	7E-3	7E-2	5E-2	3E-2	6E-3	2E-3	4E-4
M-2	0.75	0.2	2.0	150 E6	0.7	5E-3	0.5	0.6	0.2	7E-2	2E-2	3E-3
M-3	6.0	0.5	2.0	190 E6	0.8	5E-3	0.5	0.6	0.2	8E-2	3E-2	3E-3
M-4	0.2	0.0	2.0	70 E6	0.9	6E-3	0.2	0.6	0.5	7E-2	5E-2	7E-3
M-5	8.3	4.1	0.5	450 E6	0.9	6E-3	1E-2	0.5	0.5	5E-2	4E-2	6E-3
M-6	4.3	4.1	0.5	440 E6	0.9	6E-3	1E-2	0.5	0.5	5E-2	4E-2	7E-3
M-7	20.1	16.0	0.5	540 E6	0.9	6E-3	9E-3	0.3	0.3	3E-2	2E-2	4E-3
M-8	4.5	4.0	0.5	22 E6	0.9	7E-3	8E-3	1E-5	1E-5	1E-6	1E-6	2E-7
M-9	21.0	20.0	0.5	22 E6	0.9	6E-3	2E-3	2E-6	1E-6	2E-7	9E-8	1E-8
M-10	95.0	80.0	10.0	NA	0.3	2E-3	8E-4	8E-4	1E-3	9E-5	7E-5	1E-5
M-11	95.0	80.0	10.0	NA	6E-3	2E-5	2E-5	1E-5	2E-5	1E-6	1E-6	2E-7
M-12	0.5	0.0	5.0	NA	1E-3	9E-6	6E-6	1E-6	9E-7	2E-7	8E-8	1E-8

Table 3.11 Interfacing LOCA - V sequence

	Sequence		
	M-1A	BMI-2104 V-Sequence Surry	IPS[7] V+α
Xe-Kr	.9	1.0	1.0
I+OI	.707	.2	.7
Cs-Rb	.5	.2	.5
Te-Sb	.3	5(-2)	.1
Ba-Sr	6(-2)	1(-2)	6(-2)
Ru	2(-2)	7(-4)	2(-2)
La	4(-3)	2(-3)	2(-3)
Release Time (hr)	2.5		2.0
Warning Time (hr)	1.0		1.0
Duration (hr)	1.0		1.0
Energy (Btu/hr)	20(6)		.5(6)

Probability

U	
1	.17
1/2	.55
1/4	.28
1/10	-
1/100	-

Table 3.12 Early overpressurization release

	Sequence		
	M-2	BMI-2104 TMLB'-δ_e Surry	ZPSS[1] 2B
Xe-Kr	.7	1.0	.9
I+OI	.505	.7	.707
Cs-Rb	.6	.6	.5
Te-Sb	.2	.5	.3
Ba-Sr	7(-2)	1(-2)	6(-2)
Ru	2(-2)	8(-4)	2(-2)
La	3(-3)	2(-3)	4(-3)
Release Time (hr)	.75		2.5
Warning Time (hr)	.2		1.0
Duration (hr)	2.0		.5
Energy (Btu/hr)	150(+6)		250(+6)
Probability			
U			
2	-		-
1	.25		.25
1/2			.60
1/4	.25		
1/10	.5		.15
1/100	-		

Table 3.13 Isolation failure

	Sequence		
	M-4 (6" dia.)	IPS[7] (8" dia.)	IPS[7] (4" dia.)
Xe-Kr	.9	.989	.7
I+OI	.206	.28	.26
Cs-Rb	.6	.48	.26
Te-Sb	.5	.36	.21
Ba-Sr	7(-2)	5.5(-2)	2.9(-2)
Ru	5(-2)	3.2(-2)	1.8(-2)
La	7(-3)	4.9(-3)	2.8(-3)
Release Time (hr)	.2		
Warning Time (hr)	-		
Duration (hr)	2.0		
Energy (Btu/hr)	70(+6)		

Probability

U	
1	.4
1/2	.6
1/4	-
1/10	-
1/100	-

Table 3.14 Intermediate and late overpressurization

(No Sprays)

	Sequence					
	MPSS M-5	MPSS M-7	BMI-2104 (Surry)	IPS[7] TMLB'-δ	ZPSS[1] (2R)	IPPSS[3] 2RW*
Xe-Kr	.9	.9	1.0	.96	.9	1.0
OI+I	.016	.015	1(-3)	1.05(-1)	.7	9.3(-2)
Cs-Rb	.5	.3	8(-4)	.34	.5	.26
Te-Sb	.5	.3	7.0(-4)	.38	.3	.44
Ba-Sr	5(-2)	3(-2)	3(-5)	3.7(-2)	6(-2)	2.5(-2)
Ru	4(-2)	2(-2)	1(-6)	2.9(-2)	2(-2)	2.9(-2)
La	6(-3)	4(-3)	9(-6)	4.9(-3)	4(-3)	1.0(-2)
Release Time (hr)	8.3	20.1		13.0	10.0	12.0
Warning Time (hr)	4.1	16.0		8.0	9.0	11.0
Duration (hr)	.5	.5		.5	3.0	2.0
Energy (Btu/hr)	450(+6)	540(+6)		98(+6)	20(+6)	19(+6)
			Probability			
U						
2					-	
1				1.0	.1	.3
.5					.2	.55
.25	5(-2)					
.1	.64	.11			.7	.15
.01	.31	.89				

*Sum of multi-phase releases.

Table 3.15 No fail

	Sequence			
	M-12	IPS[7]	ZPSS[7] 8B	IPPSS[3] 8B
Xe-Kr	1(-3)	5(-4)	2.7(-2)	2.7(-2)
OI+I	1.5(-5)	5(-6)	2.0(-4)	2.0(-4)
Cs-Rb	1(-6)	1(-5)	8(-7)	8.0(-7)
Te-Sb	9(-7)	1(-5)	1.5(-7)	1.5(-7)
Ba-Sr	2(-7)	1(-6)	1(-7)	1(-7)
Ru	8(-8)	1(-6)	3(-8)	3(-8)
La	1(-8)	2(-7)	3(-9)	3(-9)
Release Time (hr)	.5	2	2	2
Warning Time (hr)	-	1.0	1.0	1.0
Duration (hr)	5.0	8.0	10.0	10.0
Energy (Btu/hr)	-	-	-	-
	Probability			
U				
2			-	-
1	1.0	1.0	.5	.5
.5			.4	.4
.25				-
.1			.1	1
.01				-

Table 3.16 Summary of evacuation schemes and their probabilities

ANALYSIS CATEGORY	GENERAL		SEISMIC	SPECIAL TREATMENT FOR M1 AND M4					
Evacuation Scheme	1	2	3	S1	S2	S3	S4	S5	S6
Initiating Event	Non-Seismic	Non-Seismic	Seismic	Non-Seismic	Non-Seismic	Non-Seismic	Non-Seismic	Non-Seismic	Non-Seismic
Weather Condition	Normal	Adverse	Any	Any	Any	Any	Any	Any	Any
Radius of Evacuation Sector (Mi)	10	10	10	10	10	10	10	10	10
Radius of Evacuation Circle (Mi)	5	5	5	5	5	5	5	5	5
Distance traveled by evacuees (Mi)	15	15	15	15	15	15	15	15	15
Evacuation Speed (Mph)	10	7.5	2	1.2	3.0	10	1.2	3.0	10
Delay Time before evacuation (Hr)	0.92	0.92	3.38	0.92	0.92	0.92	2.0	2.0	2.0
Probability	0.88	0.12	1.0*	0.07	0.19	0.39	0.05	0.14	0.16

*Probability is 1.0 for Evacuation Scheme 3 if the release is from a seismic induced event. Otherwise it is zero. Also, the probability of Evacuation Schemes 1 and 2 and S1 through S6 will be zero for seismic initiated releases.

Table 3.17 Subjective discrete probability distribution for site consequence uncertainty evaluation

Release Fraction Adjustment Factor*	Discrete Probability
2	0.10
1	0.35
0.5	0.45
0.1	0.10

*Adjustment factor of 1 is always used for noble gas releases.

Table 3.18 List of DPD runs performed

Release Category / Source Term Multiplier*	2	1	1/2	1/4	1/10	1/30	1/100
M1A	X	X	X	X	X	X	X
M2		X	X	X			
M3		X		X			
M4	X	X	X	X			
M5		X		X			
M6	X	X	X	X	X		
M7	X	X	X	X	X	X	X
M8		X					
M9		X					
M10		X					
M11		X					
M12		X					

*Multiplier of noble gases remains 1.0 for all runs.

4. SUMMARY

4.1 Results of Level 1 Review

The preliminary evaluation of MPSS indicates that the study is of a high quality in both technical content and the material presented.

The major conclusions of the study concerning accident phenomenology, accident sequences and release categories, source term, and site consequences appear to be generally valid.

Comparison of Millstone-3, Zion, and Indian Piont plants shows significant similarities in plant and containment designs. However, variation in containment heat removal and cavity configuration and construction influence the accident progression. With the specific cavity design employed in the Millstone-3 reactor, it should provide for higher assurance of core debris retention during high pressure discharges, while in a design such as Zion, debris removal is nearly certain during high pressure discharges

The point-estimate release fractions used in the Millstone Probabilistic Safety Study are comparable in magnitude to those used in the RSS, IPPSS, and ZPSS. In those cases where comparisons can be made to the more mechanistic source term study being carried out by ASTPO,[9] it was found that the MPSS releases were either higher than or similar to the ASTPO release fractions. In the case of the early overpressurization release, the release fractions were found to be similar, while for the intermediate or late overpressurization failure, the ASTPO release is found to be substantially lower. In the case of the interfacing LOCA sequence, the ASTPO release is approximately half the MPSS release. It was found that the energy of release was substantially higher in the MPSS than in all the other studies.

Detailed evaluation of the accident quantification is not possible at this stage, and is thus planned for the future, when audit calculations will be performed in order to verify the plant response and accident progression paths, and therefore, the site consequence and risk evaluation.

4.2 Suggestions for Future Work

In order to check the methods used in the MPSS, it is proposed to analyze the late overpressurization failure sequence. In this sequence, the containment building is calculated to fail after approximately 8-15 hours. The MPSS uses release fractions based on the RSS methods and then modifies them by multiplication with uncertainty factors. Since, this modification implies a substantial reduction in this sequence, it would be a good candidate for an audit calculation. Furthermore, the effect of containment leakage rather than an abrupt failure could be determined using this sequence.

4.3 Questions and Additional Information Needs

4.3.1 Analytical Models and Phenomenology

1. Which version of the MARCH code was used for the analysis?
2. What are the implications of the single-volume containment model used in COCOCLASS9 code?
3. Is the steam velocity low enough to limit Zr oxidation by H_2 blanketing? How is the velocity estimated to be several cm/s?
4. What is the implication of including the mechanical erosion process during molten jet-concrete interaction?

4.3.2 Uncertainty Analysis

1. How important is the range of nodal probability?
2. Does a nodal probability having a range of 0.99 to 0.9999 have any meaning?

3. What is the sensitivity of the final outcome to the values of the weighting factors and the probability range?

4.3.3 Source Term and Site Model

1. What is the reason for the higher energy of release of the plume for the overpressurization failures in comparison to the other studies?
2. The evacuation model following a seismic event does not seem to account for the fact that there would be substantial damage to buildings, roads, etc.

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