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On-Site Radiation Exposure in Severe Reactor Accidents: Scoping Study

Prepared by
Stone & Webster Engineering Corporation
Boston, Massachusetts

REPORT SUMMARY

SUBJECTS Light water reactor safety / Occupational radiation control / Radiation source term

TOPICS Reactor safety / Severe LWR accidents / Severe accident management

AUDIENCE Utility licensing engineers / Nuclear regulatory staff

On-Site Radiation Exposure in Severe Reactor Accidents: Scoping Study

This scoping study uses a number of simplifying assumptions to estimate the radiation dose rates and doses that could be received by plant personnel at various in-plant locations during a postulated severe reactor accident. The results should be helpful to individuals responsible for accident management procedures and training.

BACKGROUND An EPRI member utility, interested in whether extensive special facilities might be needed to treat plant personnel who could be exposed to radiation during a postulated severe reactor accident, requested this study. Key inputs to this problem are the dose rate in selected locations and the times available for evacuation. The report provides rough estimates of such dose rates and their time periods.

OBJECTIVE To provide a first approximation of the possible radiation dose rates or doses in key areas of a nuclear power plant in a postulated severe reactor accident.

APPROACH Because different types of accidents would be expected to have different results, this study investigated three postulated accidents:

- Dropping an unshielded fuel bundle, with consequent bundle damage and release of fission products
- An interfacing systems loss-of-coolant accident (the "V" sequence) at a PWR
- A failure-to-scram accident at a Mark I BWR

Both the internal (inhalation) and external radiation doses to personnel at selected locations for a stated time period were estimated. As a base case, it was assumed that workers would use mandated full-face respiratory protection equipment, but the dose received without it was also estimated.

RESULTS Although prohibitively high radiation levels could develop in the plant from the postulated accidents, they would develop slowly enough to allow plant personnel to exit the affected area or move to a shielded location. In accidents commencing with a large steam release, such as the "V" sequence, the high temperature resulting from the steam, not the radiation, is the controlling hazard.

The control room, protected by its shielding and its special ventilation system, would remain habitable throughout the accidents. However, use of the mandated self-contained breathing apparatus would be advisable in some cases.

EPRI PERSPECTIVE

The results of this study are senstive to the assumptions used but indicate that with proper training and the use of appropriate types of respiratory equipment, severe accidents in power reactors of the types existing in the U.S. need not produce severe radiation casualties. Additional studies are needed to provide practical tools for determining, in real time and from plant observations, the amount of time available before in-plant radiation levels would reach prohibitive values.

Because of the variation among plants, and in particular the differences in control-room ventilation systems, these results are regarded as only qualitatively applicable to plants other than the ones chosen as examples in this scoping study. The methodology of this scoping study is, however, suitable for application to more-refined and extensive studies of this type.

PROJECT

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NOTICE

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ABSTRACT

The results of a scoping study of onsite radiation exposures which could take place in each of three types of postulated reactor accidents are presented. The accident types are (1) a fuel handling accident at a Mark III BWR, (2) an interfacing system LOCA or V sequence at a PWR and (3) an Anticipated Transient Without Scram (ATWS) at a Mark I BWR. Both external and internal dose pathways are considered. The results of the study indicate that prohibitively high radiation doses could be received in some plant areas if personnel were to remain there. However, times of the order of a few minutes to a few hours, depending on the type of accident, would be available before life-threatening doses would be accumulated assuming that the provided full face respiratory protection equipment were used promptly.

In the case of the V sequence accident the escaping steam would produce intolerably high temperatures in the auxiliary building before an excessive radiation dose could be received. Thus, in that accident, it is steam and not radiation that is the controlling hazard.

Special attention was given radiation doses possibly received by control room personnel for several control room air in-leakage assumptions. For occupancy during severe accidents it would be advisable for control room personnel to use self-contained breathing apparatus (SCBA) to limit exposure via inhalation.

Due to the sensitivity of the results to the assumptions necessarily made, and due also to the variation in plant designs, it would be desirable to do a similar, more refined analysis for each plant of interest.

The results of this study will be useful to individuals responsible for accident management procedures. It is indicated that it will be important for each plant to develop estimates of the time of onset of prohibitively high radiation levels in various important plant areas. It is concluded that respiratory protection is a major factor owing to the large inhalation doses which might otherwise be encountered.

CONTENTS

Section 1
INTRODUCTION

This report documents a brief study of radiation exposures which might be received by onsite workers during several postulated accidents, including scenarios involving severe core damage. This is a scoping study and is not intended as a detailed analysis of onsite exposures during accident conditions. Rather, it is intended to be indicative of the order of magnitude of the various radiation exposures to which on-site workers may be exposed for several selected accident scenarios. The onsite workers could be exposed to the accident environment if they were a part of emergency response teams. In this case the workers would be required to be fitted with the appropriate protective clothing and respirators. It is also possible that operator or maintenance personnel could be present in the area of an accident. These personnel would already be equipped with protective clothing and respirators of the type required for the area and work before the accident. If the area or work did not contain or involve potential airborne contamination prior to the accident, the personnel would be without respirators. They would egress rapidly to beyond the accident confines in the order of minutes. The doses are calculated both without and with respiratory protection in order to simulate doses to individuals who are escaping from an accident condition as well as entering an area in an emergency and also for comparison purposes to demonstrate the need for respiratory protection. While doses are reported for various times during an accident scenario for cases with and without respirators, one should bear in mind that the large doses resulting from severe core damage conditions occur hours after the start of the accident. Unprotected personnel would have evacuated or been rescued prior to the occurrence of the large doses resulting from the degraded core.

The study uses whole body dose commitments incurred per unit time in the respective radiation environments because the times elapsed to don respiratory protection and for egress from an area in which a release may have occurred are not known. Internal exposure resulting from inhalation of airborne radioactivity and its retention is a component of the radiation exposures reported herein. The lifetime whole body dose commitment from inhaled radioactivity per unit interval of breathing contaminated air, both with and without respiratory protection, is shown with the whole body dose received from immersion in contaminated air and from deposition of radioactivity on surfaces, for each of the accident sequences considered. The inhalation doses reported are the lifetime dose commitments per unit exposure interval.

The internal doses for both air-purifying full face masks and self-contained breathing apparatus (SCBAs) are provided. The protection factor (PF) used for the full air-purifying face mask is 50, and for the SCBA is 10,000 in accordance with Reference (20). The PF for the full air-purifying face masks of 50 is applicable to the particulate form of airborne activity. The predominant contributors to internal dose are of this type. If gaseous iodine were present it could be removed by a charcoal filter cartridge.

The studies utilized simplifying methods concerning the fission product groups, their chemical and physical forms and their distribution in the plant. No cases considered dose contributions from contained sources such as pipes, equipment, sumps, etc.

The following accident scenarios were included in the scoping study:

- A fuel handling accident at a Mark III BWR

  Radiation exposures in the containment were calculated first with a suspended, unshielded, fuel assembly and second with the postulated release of the gap activity resulting from dropping of the assembly with no water scrubbing of the released fission products.

- An Interfacing Systems LOCA at a PWR

  Radiation exposures in the auxiliary building were calculated first based on the release of the fission product inventory of the reactor coolant system (RCS) and second based on the release of fission products following severe core damage. The Interfacing Systems Loss of Coolant Accident (LOCA) (3), also referred to as a V-sequence, was based on valve failure and a Residual Heat Removal (RHR) system pump seal failure which allowed a pathway directly from the core to the RHR compartments in the auxiliary building, thus bypassing the containment.

- An Anticipated Transient Without Scram (ATWS) at a Mark I BWR
  Radiation exposures in the reactor building and control room were calculated based on the progression of an Anticipated Transient Without Scram (ATWS) sequence to severe fuel damage.

For the last two accidents, the radiological analyses relied on input from previous studies by the Industry Degraded Core Rulemaking (IDCOR) Program (1). The IDCOR analyses establish the timing, thermal-hydraulic conditions, and rate of release of fission products into various buildings of the plants (2),(3),(4). Simplifying assumptions were made relative to the deposition of the radioactive material in the buildings in order to estimate the dose rates which might occur under the postulated conditions.

The fuel handling accident analysis for a Mark III BWR utilized data for the Grand Gulf plant. The V-sequence analysis for a PWR used data from the previous IDCOR analysis of the Zion plant as input. The previous IDCOR analysis of the Peach Bottom Plant served as input for the ATWS sequence for a Mark I BWR.

Section 2 summarizes the results of the scoping study and lists areas of investigation for more detailed study as a result of knowledge gained in this scoping study. Section 3 discusses the analyses and calculated dose rates.

Section 2

SUMMARY

The radiation exposure estimates reported herein are intended to indicate order of magnitude dose rates and doses which might occur under severe accident conditions. More rigorous analysis is required in order to quantify the exposures for a range of potential conditions. Nevertheless, the present study provides some insight as to the magnitude of potential exposures and highlights the contribution of internal exposure due to inhalation and the importance of respiratory protection.

From the results it can be seen that the use of SCBA's reduces the internal dose to negligible levels in comparison to the external doses. It is also seen that for most situations including degraded core situations, the air-purifying full face mask form of respiratory protection appears to be adequate to reduce internal exposure to the same level or less than the external dose. The deciding factors then concern the thyroid dose and the ALARA considerations for the work activity. These decisions would also require a more detailed sample analysis of the type of fission products and their chemical and physical forms.

The egress from an accident could occur without respiratory protection. The doses however would be primarily from reactor coolant activity or from fuel element gap activity. While in these situations the internal dose would be the predominant dose, it is of a much lower level than the dose rate later from severe core damage.

The results of the scoping analyses are briefly summarized in the following subsections:

2.1 Fuel Handling Accidents at a Mark III BWR

2.2 Interfacing Systems LOCA at a PWR

2.3 ATWS Sequence at a Mark I BWR

In the case of an exposed fuel handling accident, exposure to scattered radiation would result in a dose rate of 0.4 rem/min; however, if the operators were exposed to the line-of-sight with the fuel assembly, a dose rate of roughly 50 rem/min may result. If the fuel assembly were dropped resulting in release of the radioactivity in the gap between the fuel and the cladding an external dose rate of 0.6 rem/min would result. The internal exposure would result in a lifetime whole body dose commitment of up to 0.4 rem/min of inhalation with respiratory protection and 15 rem/min of inhalation without respiratory protection.

The analysis of the Interfacing Systems LOCA indicates that radiation exposures in the normally unmanned auxiliary building compartments resulting from release of the radioactive inventory of the RCS via a 0.1 ft^2 or a 0.2 ft^2 break would result in external whole body dose rates of 0.2 and 0.5 rem/min, respectively, at 1 min after the break and 0.9 and 2.3 rem/min, respectively, at 5 min. By comparison, the lifetime dose commitment per minute of inhalation during emergency egress without respirators would be more than an order of magnitude higher than the external exposures, but still would not be life-threatening provided egress from the affected area could be carried out within a few minutes. Temperatures approaching 212°F would be a much more serious immediate personnel safety problem as the RCS coolant flashes to steam. With respirators the inhalation dose is of less concern than the external dose.

Radiation levels in the auxiliary building for the V-sequence at the PWR studied, and in the reactor building for the ATWS sequence studied would prohibit access when core damage resulted in substantial release of fission products. Dose rates as high as one million rem/hr could be encountered in those buildings under those circumstances.

A range of dose rates and doses was calculated in the control room during the ATWS sequence. The principal factors affecting those analyses are:

- The rate of in-leakage to the control room (10 cfm and 100 cfm were considered here)
- The timing of containment venting given that in-leakage occurs
- The limitations of the instantaneous or "puff" release assumption, in the present case, whereas a much more gradual release would be expected.

As with the other portions of the scoping study, inhalation was a major concern in the control room dose analysis for the case without respirators. However, excellent respiratory protective devices and filter systems are available in control rooms and the operators are well trained in their use. The cases with respirators quantitatively illustrate the importance of protecting the inhalation pathway.

The results of each of the accident analyses included in the study are briefly summarized below.

2.1 FUEL HANDLING ACCIDENT AT A MARK III BWR

Dose rates in containment were calculated in two parts for a fuel handling accident at the Grand Gulf plant. The first involved a fuel assembly which becomes exposed while suspended, and the second assumed the fuel assembly is then dropped and damaged. The radioactivity inventory in the gap between the fuel and the cladding is assumed to be released without any scrubbing of fission products by fuel pool water. The source term used is based on equilibrium core activities (3-year burn-up) and a 24-hour delay before fuel handling based on table 12.1.3 of the GESSAR report (5). The radiation exposure calculations are based on the Grand Gulf containment geometry (6). For simiplicity only iodine and noble gases are considered to be released from the gap.

In the case of the exposed fuel assembly, direct line-of-sight exposure at 30 ft would result in a whole body dose rate of roughly 50 rem/min. If the operators are not exposed to the line-of-sight, the whole body dose from scattering off the containment walls, etc., would be roughly 0.4 rem/min. If egress could be carried out within 1 min without any direct line-of-sight exposure a dose of 0.4 rem would be received.

If the fuel assembly were dropped and the gap activity were released without any scrubbing through fuel pool water, the external whole body dose rate would be approximately 37 rem/hr. Thus, if egress could be carried out within 1 min under these conditions, an external dose of 0.67 rem would be received.

The lifetime dose commitment per minute for inhalation of unfiltered radioactive material, under these circumstances, would exceed the external exposures substantially. The use of air-purifying respirators reduces the internal dose significantly below the external dose and the use of SCBA results in accumulating a negligible internal dose.

	Lifetime Dose Commitment Per Minute of Exposure (rem/min)*
External Exposure from Immersion	0.01
External Exposure from Deposition	0.67
Internal Exposure** from Inhalation w/o Respiratory Protection	15.
Internal Exposure** from Inhalation Using Air-Purifying Full Face Masks*** (PF = 50)	0.30***

*Assuming 50% airborne and 50% plated out.
**Lifetime dose commitment per minute of inhalation
***Dose is negligible using SCBA with PF = 10,000

2.2 INTERFACING SYSTEMS LOCA AT A PWR

Radiation exposures in the auxiliary building of the Zion nuclear power plant were estimated based on a postulated loss of coolant accident (LOCA) in the residual heat removal system (RHR) which interfaces with the reactor coolant system (RCS). The containment is bypassed and radioactivity is released directly into the RHR compartments of the auxiliary building. Resulting radiation exposures were calculated first as a result of release of the radioactivity in the RCS coolant and second as a result of subsequent severe fuel damage.

The interfacing system LOCA, or V-sequence, as analyzed in the IDCOR analysis of Zion involves a failure of valves between the RCS and RHR pumps and the failure of the pump seals (2),(3). (Note: Although the RCS activity is released starting at the initiation of the accident, severe fuel damage resulting in the large release of fission products from the core occurs after 21 hr in the referenced analysis.)

The release of RCS activity was evaluated with two postulated release path sizes 0.1 ft^2 and 0.2 ft^2. The analysis considered the major isotopic groups in the analysis. The resulting radiation exposure in the lower level of the auxiliary building, where the postulated break occurs, is summarized below:

Committed Whole Body Dose Per Minute of Exposure (rem/min)

0.1 ft^2 Break Area

Time (min)	External Exposure	Internal Exposure from Inhalation* w/o Respiratory Protection	Internal Exposure from Inhalation* w/Air-Purifying Full Face Masks** (PF=50)
0.2	0.04	0.6	0.01
1	0.2	2.9	0.06
5	0.9	11	0.2
10	1.5	19	0.4

0.2 ft^2 Break Area

Time (min)	External Exposure	Internal Exposure from Inhalation* w/o Respiratory Protection	Internal Exposure from Inhalation* w/Air-Purifying Full Face Masks** (PF=50)
0.2	0.1	1.5	0.03
1	0.5	6.45	0.13
5	2.4	30	0.6
10	3.3	42	0.8

*Lifetime dose commitment per minute of inhalation.
**Dose is negligible with use of SCBA.

It is important to note that the ambient temperature in the auxiliary building increases from roughly 90°F to near 212°F within about two minutes of the release of the RCS coolant. Thus, exposure to radiation would be combined with exposure to hot steam conditions, and the steam would make the building uninhabitable before radiation did so.

If the V-sequence were to progress to include core uncovery and severe fuel damage, release of fission products from the core would be carried via hot gases and released at the break location. In the sequence analyzed here, release of fission products to the auxiliary building result in exposures ranging from 10^4 to 10^6 rem/hr, starting at about 21.4 hr. Occupancy under such conditions clearly would be prohibited. Local area emergency evacuation would have been completed before the severe core damage dose rates occur.

2.3 ATWS SEQUENCE AT A MARK I BWR

Radiation exposure rates in the reactor building and control room were calculated for an Anticipated Transient Without Scram (ATWS) sequence at the Peach Bottom BWR Mark I plant. The analyses used input from the IDCOR analysis of the TC-2, wetwell vented ATWS sequence (4). Radiation exposures in the reactor building were based on the amount of CsI in the reactor building as a function of time which was arrived at by subtracting the MAAP calculated releases to the environment from the releases from the containment. However, the protracted release was not modeled in the simplified analysis of radiation levels in the control room which was based on a puff release of 100 percent of the noble gases and 3 percent of the halogens in the core inventory with an atmospheric dispersion parameter χ/Q of 1×10^{-3}sec/m^3. An assumed unfiltered in-leakage of 100 cfm prior to control room purging was included in that portion of the analysis.

In this sequence the operator was assumed to vent the wetwell airspace when a containment pressure of 115 psia was reached. Subsequent severe fuel damage and release to the reactor building via the suppression pool results in external whole body dose rates in the reactor building of roughly 4,000 rem/hr at 15 hr, and about double that at 60 hr when revaporization of CsI from the RCS is complete.

Assuming that 50 percent of the CsI is airborne and 50 percent is deposited on surfaces, the whole body dose commitment per hour of inhalation without respiratory protection in the reactor building would be roughly 750,000 rem at 15 hr and double that at 60 hr. With SCBA respiratory protection it would still be at 75 rem and 150 rem, respectively.

These radiation levels clearly would preclude occupancy or reentry in the reactor building under the conditions analyzed.

The analysis of radiation exposure in the control room based on an assumed puff release of radioactivity included three cases, one with no purge of the control room air, a second with a 10,000 cfm purge starting at 6 min and extending for

30 min, and a third with a 10,000 cfm purge starting at 30 min and extending for 90 min. Doses are calculated for 10 days following the accident.

The resultant control room doses are summarized below for two unfiltered in-leakage rates, 100 cfm and 10 cfm. The latter rate is suggested in the NRC's Standard Review Plan Section 6.4 (7).

Whole Body Dose Commitment (rem)

	External Exposure	Internal Exposure* Without Respirator	With Air-Purifying Full Face Mask (PF = 50)	With SCBA (PF = 10,000)
No Control Room Purge				
100 cfm in-leakage	270	2,700	55	0.27
10 cfm in-leakage	27	270	5.5	0.03
10,000 cfm Purge at 6 min for 30 min				
100 cfm in-leakage	11	61	1.2	neg.
10 cfm in-leakage	1.1	6.1	neg.	neg.
10,000 cfm Purge at 30 min for 90 min				
100 cfm in-leakage	24	134	2.7	neg.
10 cfm in-leakage	2.4	13.4	0.3	neg.

*Lifetime dose commitment due to inhalation during the duration of the accident.

2.4 CONCLUSIONS

It is concluded that a severe reactor accident need not necessarily result in severe radiation casualties, but would require respiratory protection and, in some cases, prompt evacuation of work areas. The control room would be habitable throughout the accident, assuming the use of full face respiratory protection devices and, in some cases, self-contained breathing apparatus. Accidents commencing with the release of reactor coolant or steam to plant working areas will become unhabitable due to the high temperature before radiation levels become prohibitively high.

2.5 AREAS OF INVESTIGATION FOR MORE DETAILED STUDY

Although the present scoping study is not an exhaustive treatment of the subject of onsite radiation exposure during severe accidents, it has helped to identify areas requiring further investigation.

A more detailed study of onsite exposures during severe accidents should include, but not be limited to, the following areas of investigation:

- More realistic analyses of airborne concentrations as a function of time rather than the 50 percent airborne-50 percent deposited simplifying assumption used in this scoping study.

- Equipped with more realistic airborne concentration data, more detailed analyses of internal exposures via the inhalation pathway could be carried out. Emphasis should be placed on refinement of exposures with realistic estimates of reductions due to respiratory protection devices.

- More extensive analyses of the initial few minutes for a wide spectrum of accidents involving release of the radioactivity in the reactor coolant system (RCS). The present scoping study illustrates the order of magnitude of exposure associated with RCS release; however, only very limited cases were considered and several simplifying assumptions were made. The present study results indicate that exposures under a variety of conditions of release of RCS activity, ranging from small instrument line breaks to large pipe breaks, would be a very fruitful area of investigation of practical significance.

- The present analyses indicate that very high radiation levels would be expected under severe core damage conditions. Although occupancy would appear to be precluded in the affected buildings, more definitive studies might indicate that other features, such as HVAC operation, may greatly reduce radiation levels in some areas where access may be desirable under very controlled conditions.

- Time and motion type analyses coupled with realistic estimates of airborne vs. deposited fractions, and a larger number of radionuclides, could be used to quantify doses during the following periods:

  -- Initial discharge of radioactivity

  -- During egress from affected areas

  -- During potential life-saving entry to affected areas

  -- During re-entry in the long-term to secure manually operated valves, etc.

- Dose rates from contained sources, e.g., piping containing radioactive water, and sumps, etc. These were important in the TMI accident and any detailed analysis should reference the TMI information.

- Some estimate of the contamination of clothing and exposed body surfaces should be made using the detailed estimates of airborne concentrations, mentioned above.

- The ambient thermal conditions, e.g., the ambient air temperature, should be investigated with the objective of attempting to identify potential areas where thermal injury conditions would coexist with potentially large doses and/or skin surface concentrations in combination with burns.
- Existing control room design features which may mitigate exposures during severe accident scenarios should be investigated.
- Availability of respiratory protection devices in areas other than the control room and procedures by which personnel would be instructed to use them, and the time required to put them on.
- Existing and other appropriate procedures aimed at in-plant personnel protection in the event of various types of severe accidents.

Areas such as the above were not addressed in this scoping study. However, the need to develop more detailed estimates in these areas is more apparent as a result of the study.

Section 3

DESCRIPTION OF ANALYSIS

The analyses discussed below are based on data from three typical plants. However, they are only generally representative of those plants or others of their type, since individual plants of the same type may differ significantly. The analyses are intended to show illustrative examples of radiation exposures for selected accidents rather than provide detailed analytical results.

3.1 FUEL HANDLING ACCIDENT AT A MARK III BWR

3.1.1 Description of Accident

A fuel handling accident is postulated to occur at the Grand Gulf Station, a BWR Mark III nuclear power plant. It is assumed that a fuel assembly is being transferred when the fuel pool seal fails, resulting in loss of the water shielding the assembly. It is further assumed that the scenario progresses to include a fuel drop accident resulting in loss of assembly integrity and release of radioactivity from the gap between the fuel and the cladding without benefit of any water scrubbing of the released fission products.

3.1.2 Analysis Input and Assumptions

The following input and assumptions (1 and 6) were employed (see figure 3-1 for a schematic representation of the plant arrangement):

1. Equilibrium core inventory (3 year)
2. 24 hour delay before fuel handling
3. Location and geometry information:

    a. Raised assembly is 2½ ft above stored fuel
    b. Active fuel region is between El. 185'-8" and 198'
    c. Top of reactor cavity is at El. 208'-10"
    d. Top of dome is at El. 306'-6"
    e. Containment radius = 62'-6"
    f. Bend line is at El. 244'
    g. Reactor cavity dimensions: 37' x 37'

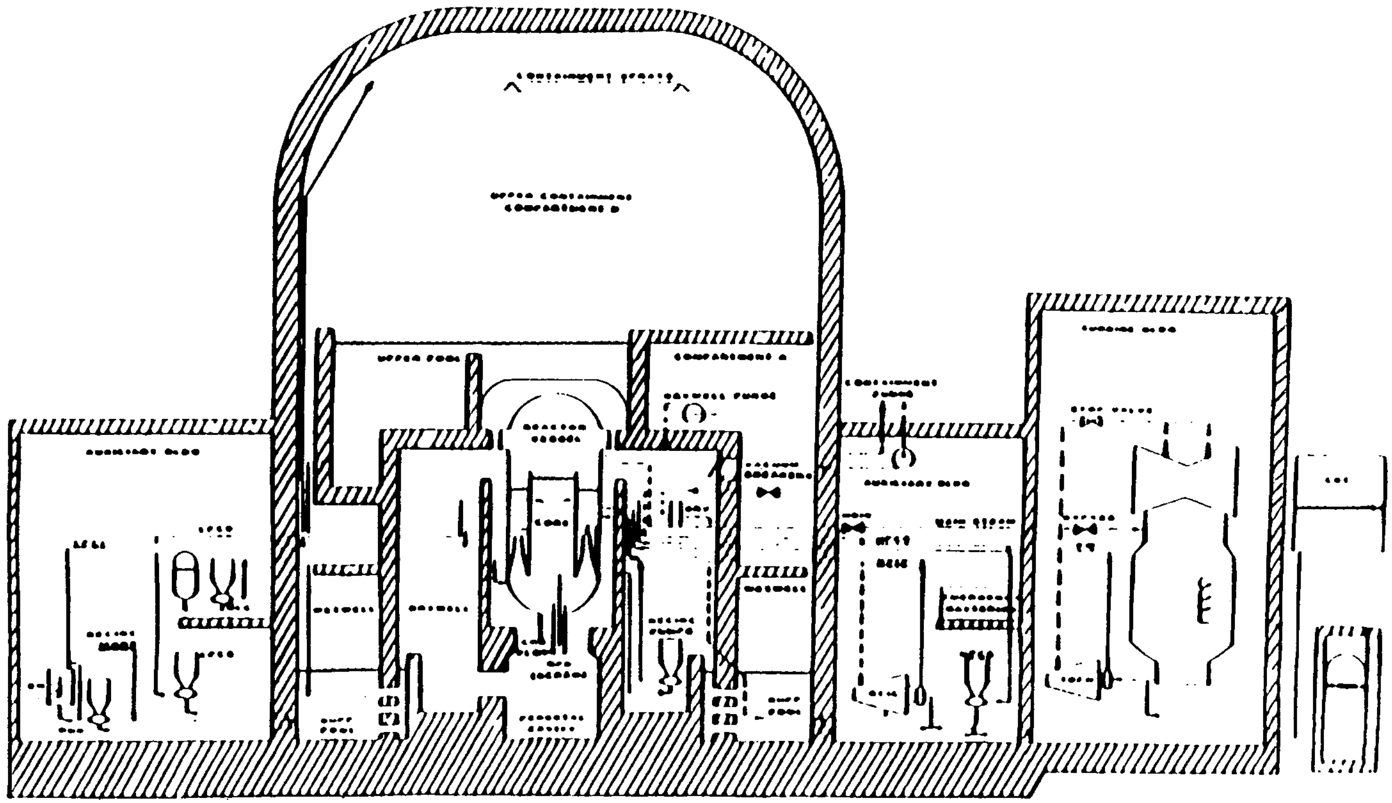

Figure 3-1. Schematic Representation of Grand Gulf Safety and Other Systems

h. Source height = 148 in.

i. Effective source area = 0.25 ft^2 = 36 $in.^2$

j. Effective radius = $(36/\pi)^{1/2}$ = 3.39 in.

k. Buildup factor = lead

4. Source Term for Analysis

The gamma energy spectrum (24 hr decay) is taken from GESSAR 251 (5, table 12.1.3) and normalized to Grand Gulf (6).

From (5), table 12.1.1:

Core equivalent radius = 98.58 in.
Top of active fuel = 364.31 in.
Bottom of active fuel = 216.31 in.

$$\text{GESSAR core volume} = \frac{(98.58\text{ in.})^2(364.31 - 216.31)\text{ in.}}{(12\text{ in./ft})^3}$$

$$= 2615\ ft^3$$

Total number of GESSAR assemblies = 848

$$\text{Volume, per assembly} = \frac{2615}{848} = 3.08\ ft^3$$

Active height = 148 in. = 12.33 ft

$$\text{Assembly area} = \frac{\text{volume}}{\text{height}} = 0.25\ ft^2 = 36\ in.^2$$

For a cylindrical assembly:

$$\text{Effective radius} = R=(36/\pi)^{1/2}=3.39\text{ in.}$$

Total number of assemblies: GESSAR = 848
Grand Gulf = 784

The source term, Sv, is then calculated as follows:

Sv(MeV/cc-sec) = (GE** source (Mev/sec-watt) x GG*
power (watts)/GG assemblies)/
(GE vol/GE assemblies)

*GG designates Grand Gulf
**GE designates GESSAR

$$Sv(MeV/cc\text{-}sec) = (GE(MeV/sec\text{-}watt) \times 3.833 \times 10^{9} \text{ (watts)}/784 \text{ assemblies})/(2.615 \times 10^{3}(ft^{3}) \times 2.83 \times 10^{4} \text{ } (cc/ft^{3})/848 \text{ assemblies})$$

$$GG \ Sv(MeV/cc\text{-}sec) = GE(MeV/sec\text{-}watt) \times 56$$

This results in the source term tabulated in table 3-1.

Table 3-1

SOURCE TERM FOR ANALYSIS OF FUEL HANDLING ACCIDENT

Energy Group (MeV)	GE(MeV/sec-watt)	GG Sv(MeVcc-sec)
0.1 - 0.4	1.8 + 9*	1.01 + 11
0.4 - 0.9	7.5 + 9	4.20 + 11
0.9 - 1.35	2.3 + 9	1.29 + 11
1.35 - 1.8	3.1 + 9	1.74 + 11
1.8 - 2.2	4.5 + 8	2.52 + 10
2.2 - 2.6	2.9 + 8	1.62 + 10
2.6 - 3.0	5.7 + 6	3.19 + 08
3 - 4	7 + 6	3.92 + 08
4 - 6	<1 + 6	5.60 + 07

3.1.3 Committed Dose Rates With Exposed Fuel Assembly

External Exposure from Exposed Assembly

It is assumed that the operators are not on a "line of sight" directly between them and the fuel assembly. Thus, the major dose rate is due to scattering off the containment walls and dome.

To calculate the scatter dose rate on the operating floor, El. 208'10", we calculate the rate at the containment axis thus allowing use of complete symmetry (in two dimensions, θ and ϕ) from any scatter surface. Due to the relatively large source-to-scatter surface distance, we may assume that the incident dose rate on the containment wall will not vary significantly (i.e., by more than a factor of about 3). If we take the "average" incident dose rate on the containment wall and assume complete symmetry, then simply multiplying by an "average total" albedo yields the approximate scatter dose rate.

The dose rates incident on the containment wall are as follows (see figure 3-2.):

Detector	Elevation (ft)	Dose Rate (rem/hr)
B	244	286
C	258	208
D	265	305
E	275	272
F	306	206

A somewhat conservative "average" incident dose rate of ~ 250 rem/hr, combined with an average total albedo of 10 percent (8) results in a scatter dose rate at the operating floor of approximately 25 rem/hr. If the operators were briefly exposed to the line-of-sight radiation from the exposed fuel assembly at a distance of approximately 30 ft, the dose rate would be roughly 3,000 rem/hr under those conditions.

3.1.4 Committed Dose Rates With Damaged Fuel Assembly

It was further assumed that the exposed fuel assembly is dropped following loss of that portion of the fuel pool water which would otherwise act to scrub released fission products. The resulting release of fission products in the gap between the fuel and the cladding would result in internal exposure from breathing contaminated air and external exposure from airborne and deposited material. The committed dose rates from these exposure pathways are discussed below.

Internal Exposure from Damaged Assembly

The internal exposure dose commitment per hour of inhalation of radioactive material resulting from the damage to the assembly was calculated as shown below.

The isotopic fuel inventory (Curies/MWt) for a 3-year burn-up followed by 1.0 day shutdown is as follows (9):

DETECTOR	DOSE RATE (REM/HR)	SOURCE CONFIGURATION SIZE
A	25*	(scattering only)
B	286	cylindrical r = 3.39", H = 148"
C	208	cylindrical r = 3.39", H = 148"
D	305	line H = 148"
E	272	line H = 148"
F	206	line H = 148"

* Assuming no line of sight contribution.
(i.e. scattering only)

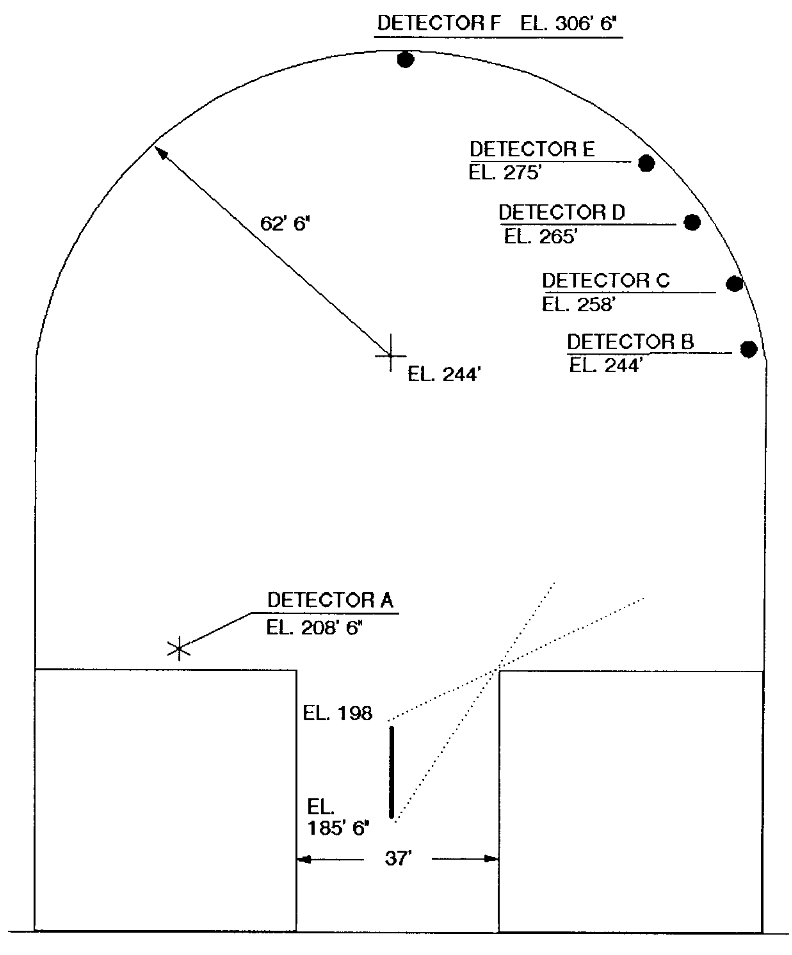

Figure 3-2. Detector Locations for Fuel Handling Accident

Isotope	Activity (Ci/MWt)	Activity (Ci/assembly)*
Kr - 85	3.02 + 2	1.48 + 3
Kr - 85M	1.66 + 2	8.12 + 2
Xe - 131m	1.58 + 2	7.73 + 2
Xe - 133	5.33 + 4	2.61 + 5
Xe - 133m	2.09 + 3	1.02 + 4
Xe - 135	1.22 + 4	5.96 + 4
Xe - 135m	6.36 + 2	3.11 + 3
I - 130	2.75 + 2	1.34 + 3
I - 131	2.45 + 4	1.20 + 5
I - 132	3.16 + 4	1.54 + 5
I - 133	2.53 + 4	1.24 + 5

*Grand Gulf has 784 assemblies for 3833 MWt, or 4.89 MWt/assembly. Other nuclides have been neglected for purposes of this scoping study.

Total upper containment volume, $V = 2.67 \times 10^{10}$ cc

According to the NRC regulatory position (10), 10% of the inventory escapes and of that one-half is plated out. Then the I-131 airborne concentration immediately after the drop is 2.25×10^5 pCi/cc.

With an average breathing rate of 1×10^7 cc of air per 8 hr working day (12),

$BR = 1.25 \times 10^6$ cc/hr

The dose rate conversion factors for I-131 (adult) are (13):

k (thyroid) = 1.49×10^{-3} mrem per pCi inhaled and

k (whole body) = 2.56×10^{-6} mrem per pCi inhaled

For I-133 (adult):

k (whole body) = 5.65×10^{-7} mrem per pCi inhaled

The committed dose per unit exposure interval from inhalation of the i-th isotope is given by:

$$DR_i = A_i \times BR \times k/V$$

where $A_i = 0.1 \times 0.5 \times$ Activity per Assembly for Isotope i and BR, k, and V are as defined above.

The effect of respiratory protection is calculated by dividing the calculated dose by the protection factor of 50 for a air-purifying full face mask or by a factor of 10,000 for SCBA.

The most significant contributors to the committed inhalation dose per unit exposure interval yield the following results in rem/hr:

	w/o Respiratory Protection	With Air-Purifying Full Face Masks (PF=50)	With SCBA (PF=10,000)
I-131 thyroid	4.2×10^5	8.4×10^3	42
I-131 whole body	7.20×10^2	14.4	7.2×10^{-2}
I-133 whole body	1.64×10^2	3.28	1.64×10^{-2}

External Exposure From Damaged Fuel Assembly

(a) The external whole body committed dose rates from deposited material from the damaged assembly were calculated as follows:

The total upper containment (walls and dome) surface area is:

$S = 3.56 \times 10^7 \text{ cm}^2$

The external dose factor standing on contaminated ground for I-132 (k) is 1.70×10^{-8} mrem/hr per pCi/m^2 (13).

The dose rate from deposition of the most significant contributor (I-132) is given by:

$$DR_i = A_i \times k/S$$

where $A_i = 0.1 \times 0.5 \times 1.54 \times 10^5$

resulting in:

I-132 dose rate = 37 rem/hr, whole body

(b) The external whole body dose rates from immersion in contaminated air from the damaged assembly were calculated as follows:

The dose rate due to immersion in a semi-infinite radionuclide plume is as follows (10):

$$DR\gamma = 0.25\ E\gamma\ X \text{ rem/hr}$$

where $E\gamma$ = average gamma energy = 1.0 Mev/disintegration (by assumption)

X = Concentration of nuclides of interest, Ci/m^3

The total iodine and noble gas concentration in the upper containment is:
$X = 2.0\ Ci/m^3$

Thus, the immersion dose rate is $DR\gamma = 0.5$ rem/hr and is negligible compared to the previous dose contributors.

Table 3-2 summarizes the committed dose per minute of exposure based on release of either 1 percent or 10 percent of the fission product inventory in a fuel assembly, with no scrubbing by water which may be present over the damaged fuel assembly. The whole body dose rates, although high, are such that rapid egress from the area would avoid receipt of a large dose. The thyroid dose rates are sufficiently high that respiratory protection would be very important. The effectiveness of respiratory protection included in this study is shown to be very effective, which can be expected due to the aerosol nature of the radioactive material.

3.2 INTERFACING SYSTEMS LOCA AT A PWR

This section presents a description of the accident analyzed, the input data, assumptions, and the methodology that was used to perform the analysis of the radiation exposures in the auxiliary building of a PWR with an interfacing systems LOCA. Resulting dose rates are included for the following two conditions: (1) the release of the radioactive inventory of the reactor coolant system cooling water and (2) the subsequent release of substantial quantities of fission products due to severe core damage.

Table 3-2

DOSE RATES IN BWR-MK III CONTAINMENT
WITH ONE SEVERELY DAMAGED FUEL ASSEMBLY

	Dose Commitment Per Unit Exposure Interval (rem/min)	
	1% Release	10% Release
Whole Body		
Internal *Exposure from Inhalation using Air-Purifying Full Face Masks (PF=50)	0.04	0.4
Internal* Exposure from Inhalation using SCBA (PF=10,000)	0.00015	0.0015
Internal* Exposure from Inhalation w/o Respiratory Protection	1.5	15
External Exposure from Deposition	0.067	0.67
External Exposure from Immersion	0.001	0.01
Thyroid Exposure from Inhalation* using Air-Purifying Full Face Mask (PF=50)	16.6	166
Thyroid Exposure from Inhalation* using SCBA (PF=10,000)	0.083	0.83
Thyroid Exposure Inhalation* w/o Respiratory Protection	830	8,300

*Lifetime dose commitment per minute of inhalation.

Assumptions:

(1) One of 784 fuel assemblies in 3,833 MWT core is damaged in air.
(2) One half of released iodine is uniformly mixed in air.
(3) One half of released iodine is deposited on surfaces.
(4) Breathing rate is 1.25×10^6 cc/hr (unfiltered).

3.2.1 Description of Accident

The interfacing systems LOCA, or V-sequence, as analyzed in the IDCOR analyses for Zion(2) involves a postulated dual failure of the motor operated double disk gate valves between the Reactor Coolant System (RCS) and the suction of the Residual Heat Removal (RHR) pumps. This, in turn, results in the release of high pressure fluid into the low pressure piping. This would likely result in a failure of the pump seals. A seal failure would have a comparatively small break area. The IDCOR analysis assumed a 0.1 ft^2 (93 cm^2) break area. This break location would vent the primary system coolant into the RHR pump rooms of the affected unit. Although the radioactivity in the RCS would be discharged within a few minutes, the accident sequences as analyzed by IDCOR did not result in release of fission products from severe core damage until more than 21 hours in this sequence.

3.2.2 Analysis Input and Assumptions

The inputs and assumptions that were employed in the analysis of the V-sequence are as follows:

1. Power level = 3250 MWt (14), p. 1.0-1)
2. Breathing rate = 1.25×10^6 cc/hr (12)
3. Volumes and surface areas of the three auxiliary building levels analyzed are as follows (2, table C-1):

Level	Volume (m^3)	Surface Area (m^2)*
1	8000	1015
2	8000	1313
3	4000	886

*Only outside walls are considered.

The following assumptions were used in the early RCS inventory release calculations (before fuel damage):

4. Mass flow rates of primary coolant through a 0.1 ft^2 break and into the lower level of the auxiliary building were obtained assuming 0.6 Moody flow (15) with saturated liquid and fL/D = 2 to approximate the homogeneous equilibrium model (HEM). The input pressure transient was taken from figure 4-6. of (2).
5. It was assumed that iodine and cesium are the major contributors to the dose.

6. Iodine and cesium concentrations in the primary coolant were taken from (14, table 14A.4-2). A simplified reactor coolant system radionuclide inventory is listed below.

Nuclide	Concentration (pCi/g)
I-131	2.24 + 6
I-132	8.28 + 5
I-133	3.59 + 6
I-134	4.75 + 5
I-135	1.84 + 6
Cs-134	2.21 + 5
Cs-136	1.22 + 5
Cs-137	1.10 + 6

7. Reactor coolant exits the break undiluted by ECCS flow.

8. It was assumed that the coolant mass is uniformly distributed very quickly over the entire lower level of the auxiliary building. Even though this mixing time would be short, it was assumed that it would be sufficient for the decay of N-16, so that N-16 dose rates, and doses, would be minimal compared to those due to iodine and cesium.

The following assumptions were used in the late (after fuel damage) release calculations:

9. Radioactive core inventory used to calculate the dose rates following core damage was the same as used in (16, table II).

10. The amount of fission products present in the auxiliary building, as a function of time during the postulated V-sequence, were taken from MAAP computer runs. Fission products in these runs are collected into groups as identified below:

Group No.	Species
1	Noble gases
2	CsI
3	Te
4	Ba, Sr
5	Ru, Rh, Mo, Tc
6	CsOH

11. The release groups and the mass of the core inventory for each group, from (17), table 6.6, are presented below.

Release Group	Initial Total Mass In Core (kg)	Fraction of Mass By Species
1	336.5	Kr (0.049), Xe (0.951)
2	29.8*	Cs (0.503), I (0.497)
3	30.8	Te (1.0)
4	134.5	Ba (0.561), Sr (0.439)
5	390.6	Ru (0.328), Rh (0.066) Mo (0.489), Tc (0.117)
6	146.*	Cs (1.0)

*(17) provides the masses of I = 14.8 Kg and Cs = 161.0 Kg

Assuming that all the iodine is in CsI form we have:

From (11), atomic weight of I = 126.9 g
atomic weight of Cs = 132.9 g

Then, $\frac{14.8 \times 10^3 g}{126.9 \text{ g/g-mole}}$ = 116.6 g-moles of iodine

combined with 116.6 g-moles of Cs, which in turn gives,

(116.6 g-mole of Cs) x (132.9 g/g-mole of Cs) = 15. kg of Cs

Since 15 kg of Cs is in CsI form, then (161-15) kg = 146 kg of Cs is in CsOH form (it is assumed that all Cs is in either CsI or CsOH form).

12. The MAAP results (3) include the mass of each radionuclide group in vapor, aerosol, and deposited form.

The curies of each radionuclide in each of the above forms is then calculated using:

$$A_i(t) = M(t) \times f_i \times Q_i \times \exp(-\lambda_i t)/m^s_i$$

where: $A_i(t)$ = Curies of radionuclide i in vapor, aerosol or deposited form

$M(t)$ = Amount of vapor, aerosol or deposited material of the group to which radionuclide i belongs (Kg)

f_i = Fraction of group mass for the species to which radionuclide i belongs

m^s_i = Mass of species to which radionuclide i belongs (Kg)

Q_i = Initial activity of radionuclide i (curies)

λ_i = Decay constant for radionuclide i

t = Time during the accident (hr)

The dose rates in each of the three levels of the auxiliary building were then calculated using the following equations:

(a) Immersion Dose Rates

Whole body dose rates resulting from immersion in contaminated air were calculated as follows:

$$DR_i^c\,(t) = (A_i^a(t) + A_i^v(t)) \times CF_\gamma{}^i/V$$

where:

$Dr^c{}_i(t)$ = Dose rate (rem/hr) due to radionuclide i at time t during the accident

$A^v{}_i(t)$ = Curies of isotope i present in vapor form at time t

$A^a{}_i(t)$ = Curies of isotope i present in aerosol form at time t

V = Volume of the auxiliary building level being examined

$CF_\gamma{}^i$ = Gamma dose rate conversion factor for radionuclide i

(b) Dose Rates Due to Deposited Aerosols

Whole body dose rates resulting from deposited radioactive material were calculated as follows:

$$DR^D{}_i(t) = 2 \times A^D{}_i(t) \times CF^D{}_i/S$$

where:

$DR^D{}_i(t)$ = Dose rate (rem/hr) due to aerosols deposited in the auxiliary building

$A^D{}_i(t)$ = Curies of isotope i, deposited in the auxiliary building at time t

$CF_i{}^D$ = Deposited activity dose conversion factor for isotope i

S = Surface of the auxiliary building portion being considered.

Note: A multiplier of 2.0 is applied to account for the fact that the individual is exposed to more than one contaminated surface, and the dose conversion factors are for a plane surface contaminated on one side only.

(c) Inhalation

The lifetime dose commitments per unit time breathing contaminated air were calculated as follows:

$$DR^{I}_{i}(t) = (A^{a}_{i}(t) + A^{v}_{i}(t)) \times BR \times CF^{I}_{i}/V$$

where: $DR^{I}_{i}(t)$ = Committed dose per unit exposure interval (rem/hr) due to inhalation of airborne radionuclides

BR = Normal breathing rate (1.25×10^{6} cc/hr)

CF^{I}_{i} = Inhalation dose conversion factor radionuclide i.

The above equation is divided by the protection factor (PF) for the appropriate respiratory protection (PF = 50 for air-purifying full face masks and PF = 10,000 for SCBA (20)).

3.2.3 Committed Dose Rates with Release of RCS Activity

Given the above input data and assumptions, the dose rates in the lower level of the auxiliary building at selected times during the accident, but prior to any core damage, were calculated and are presented in table 3-3 and figure 3-3.

For comparison, a 0.2 ft^2 RCS break in the lower level of the Zion auxiliary building is also shown. Flow rates through the 0.2 ft^2 break were obtained from

a V-sequence analysis performed on Surry and presented in the report of the ANSI Special Committee on Source Terms (18).

The ambient air temperatures in the three elevations of the auxiliary building are indicated in figure 3-4. for the 0.1 ft^2 break area case. The pump seal LOCA is assumed to occur in the lower level (Node 1); however, the temperatures in all three nodes are observed to be very similar owing to the rapid dispersal of the

Table 3-3

COMMITTED WHOLE BODY DOSE RATES IN THE LOWER LEVEL OF THE AUXILIARY BUILDING DUE TO RCS DISCHARGE

0.1 ft^2 Break Area

Time (min)	Inhalation w/o Respiratory Protection	Inhalation* Using Air Purifying Full Face Mask Protection (PF = 50)	Inhalation Using SCBA (PF = 10,000)	Immersion	Deposition
0.167	0.55	Neg.	Neg.	0.040	0.003
0.833	--	--	--	--	--
1.0	2.9	0.06	Neg.	0.21	0.020
5.0	11.2	0.22	Neg.	0.83	0.060
10.0	18.8	0.38	0.002	1.39	0.10
20.0	30.0	0.60	0.003	2.18	0.15
22.0	31.2	0.62	0.003	2.30	0.17

0.2 ft^2 Break Area

Time (min)	Inhalation w/o Respiratory Protection	Inhalation* Using Air Purifying Full Face Mask Protection (PF = 50)	Inhalation Using SCBA (PF = 10,000)	Immersion	Deposition
0.167	1.49	0.03	Neg.	0.092	0.028
0.833	6.45	0.13	Neg.	0.48	0.030
1.0	--	--	--	--	--
5.0	30.0	0.60	0.003	2.22	0.15
10.0	42.2	0.84	0.004	3.12	0.22
20.0	--	--	--	--	--
22.0	48.3	0.97	0.005	3.58	0.25

*Lifetime dose committment per minute of inhalation.

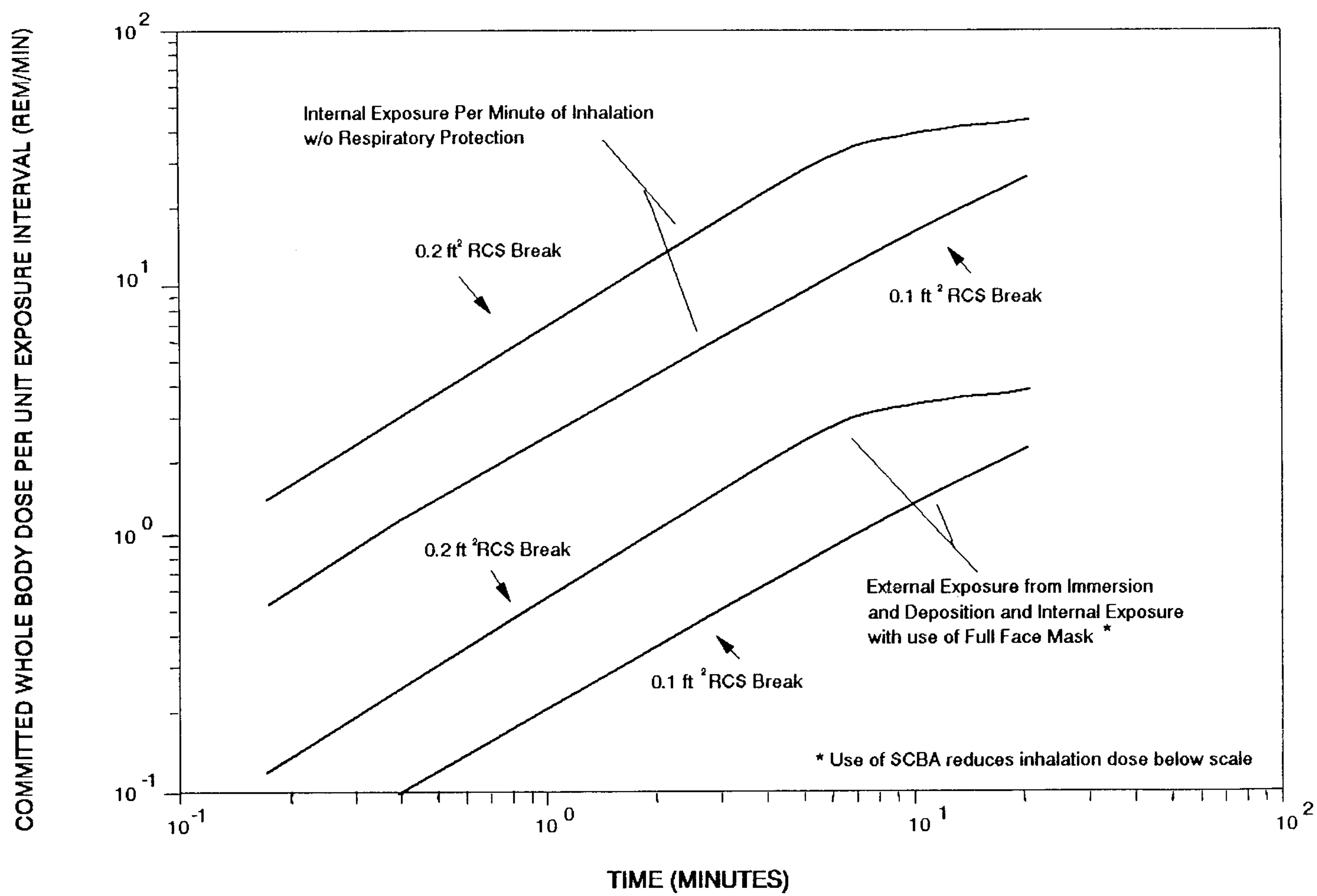

Figure 3-3. Committed Whole Body Dose Per Unit Exposure Interval Near LOCA in Auxiliary Building Due to RCS Leakage

steam released. Within approximately one minute the temperature increased from an assumed starting temperature of 90°F to slightly under 212°F and remained at that level for an extended period. This is a simplified analysis, and it is expected that a more rigorous analysis would indicate the temperature would drop below this level over the time period included in this figure. Nevertheless, these data are instructive relative to understanding the combined thermal and radiological conditions during such a postulated accident.

3.2.4 Committed Dose Rates With Severe Fuel Damage

The dose rates in the lower, first and second level of the auxiliary building during the severe fuel damage portion of the postulated V-sequence are summarized in table 3-4 and are presented graphically in figure 3-5. The internal exposure resulting from inhalation without respiratory protection is provided for comparison only. Emergency evacuation of these areas would have been completed long before severe core damage occurred so all reentry access would be under conditions dictated by station HP's using the appropriate respiratory protection.

Maximum whole body dose rates in the auxiliary building occur at 21.91 hr, as follows:

Whole Body Dose Commitment
Per Unit Exposure Interval (rem/hr)

	External	Internal from Inhalation*	Total
Lower Level	9.2×10^5	1.0×10^3	9.2×10^5
First Level	2.9×10^5	275	2.9×10^5
Second Level	1.6×10^5	137	1.6×10^5

*Lifetime dose commitment from internal exposure per hour of inhalation using respiratory protection with a PF = 10^4

Based on the dose rates calculated to occur earlier in the accident during the initial RCS inventory release, it is not expected that persons would be in the auxiliary building when the dose rates resulting from severe fuel damage, discussed in this section, would exist. Access to the auxiliary building under those conditions even with the use of SCBA respiratory protection would be precluded by the extremely high radiation levels resulting from the direct release of fission products from the severely damaged core.

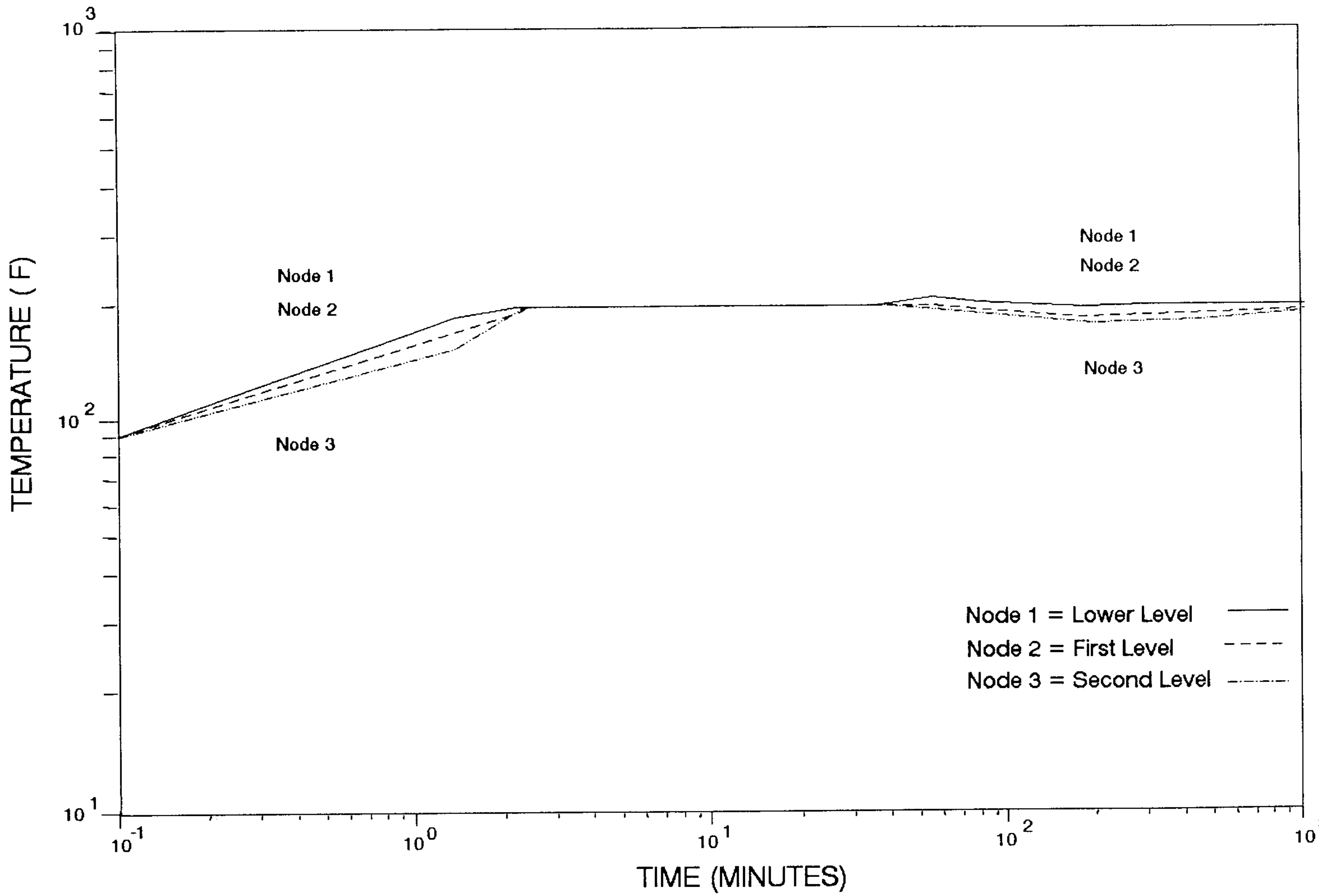

Figure 3-4. Air Temperature in Auxiliary Building Interfacing Systems LOCA

Table 3-4

WHOLE BODY DOSE COMMITMENT PER HOUR EXPOSURE IN AUXILIARY BUILDING WITH SEVERE FUEL DAMAGE V-SEQUENCE

		External Exposure (rem/hr)				Internal Exposure*	
Location	Time	Immersion	Deposition	Total	Inhalation (rem/hr)* w/o Respiratory Protection**	Inhalation Using Air Purifying Full Face Masks (PF = 50)	Inhalation Using SCBA (PF = 10,000)
Lower Level	21.16	5.56×10^{2}	2.08×10^{2}	7.64×10^{2}	1.31×10^{4}	2.62×10^{2}	1.31
	21.41	5.66×10^{5}	3.92×10^{4}	6.05×10^{5}	9.76×10^{6}	1.95×10^{5}	976
	21.91	7.09×10^{5}	2.11×10^{5}	9.20×10^{5}	1.00×10^{7}	2.00×10^{5}	1.00 x 3
First Level	21.16	3.91×10^{2}	1.63×10^{1}	1.33×10^{2}	4.60×10^{3}	92.0	0.46
	21.41	8.58×10^{4}	9.54×10^{2}	8.68×10^{4}	1.48×10^{6}	2.96×10^{4}	148
	21.91	2.20×10^{5}	1.73×10^{4}	2.87×10^{5}	2.75×10^{6}	5.50×10^{4}	275
Second Level	21.16	4.36×10^{2}	5.45×10^{0}	4.41×10^{2}	2.61×10^{3}	52.2	0.261
	21.41	6.81×10^{4}	2.38×10^{2}	6.83×10^{4}	8.83×10^{5}	7.66×10^{3}	88.3
	21.91	1.59×10^{5}	4.86×10^{3}	1.64×10^{5}	1.37×10^{6}	2.74×10^{4}	137

*Lifetime dose commitment per hr of inhalation

**The inhalation exposure without respiratory protection is not realistic for this case because severe core damage occurs 21 hours after the accident.

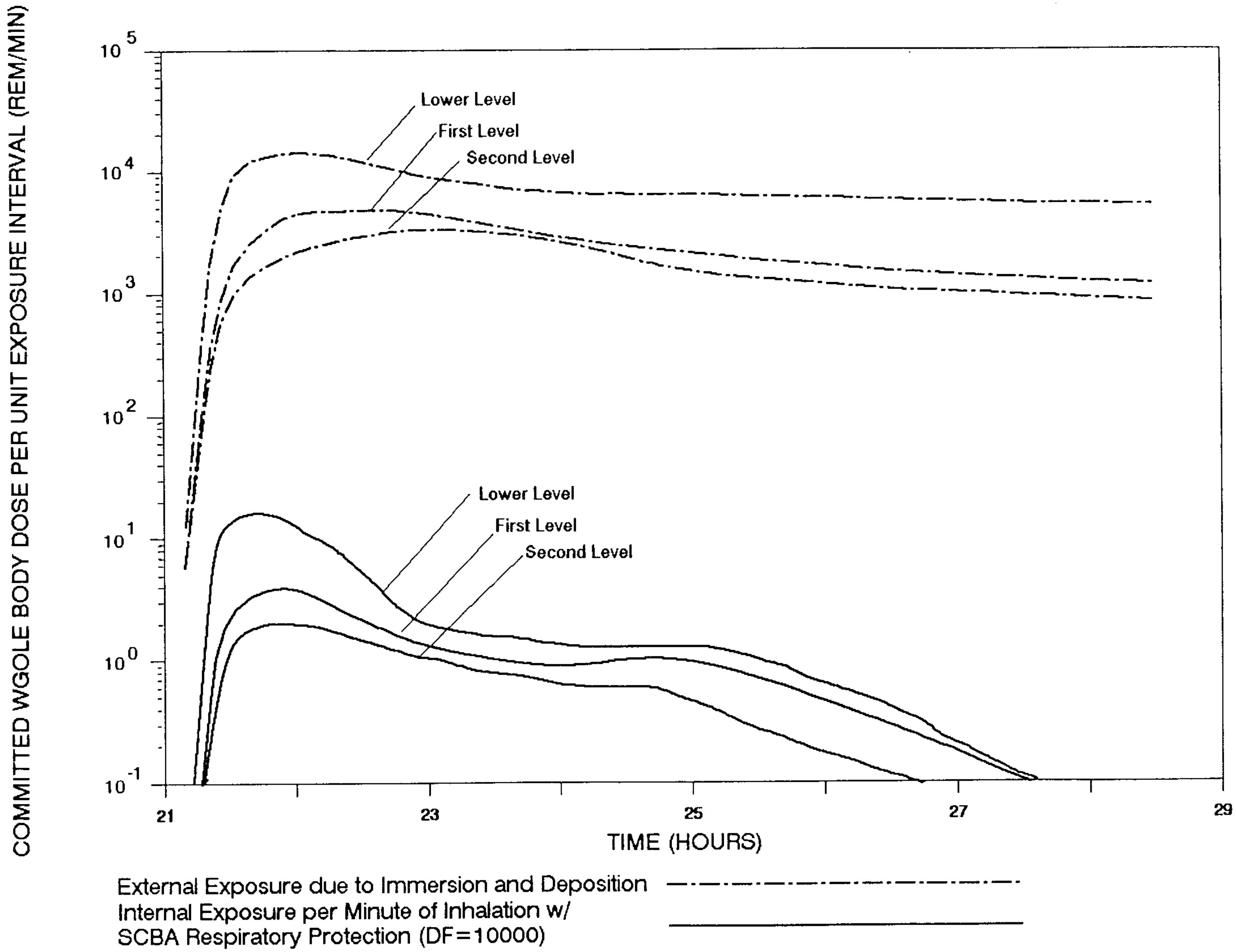

Figure 3-5. Committed Whole Body Dose Per Unit Exposure Interval in the Auxiliary Building During a V-Sequence

3.3 ATWS SEQUENCE AT A MARK I BWR

This section presents a description of the accident analyzed, the input data, assumptions, and the methodology that was used to perform the analysis of the radiation exposures associated with an Anticipated Transient Without Scram (ATWS) scenario for a Mark I BWR. Resulting dose rates are included for two locations, (1) the reactor building following venting via the suppression pool, and (2) the control room assuming a puff release with unfiltered air in-leakage.

3.3.1 Description of Accident

The ATWS scenario analyzed here is the TC-2 case reported in the IDCOR Technical Summary Report[1]. The details for the analysis, such as event timing and rate of release of fission products to the reactor building, are taken from IDCOR Technical Report 23.1PB, an analysis of the Peach Bottom Mark I BWR (4). (See figure 3-6 for a depiction of the Mark I containment and the flow paths in the MAAP computer program model.)

This sequence is initiated by Main Steam Isolation Valve (MSIV) closure followed by failure of the reactor protection system to initiate reactor scram, and by failure to initiate standby liquid control. Successful recirculation pump trip occurs, and the injection systems remain available until high suppression pool temperatures cause the loss of those systems. The control rod drive flow is assumed to continue until the condensate storage tank is depleted. The condensate pump is assumed to be unavailable. The operator is assumed to vent the wetwell airspace when the drywell pressure reaches 115 psia. The resulting release of fission products to the reactor building via the suppression pool forms the basis for the present analysis of reactor building radiation levels.

3.3.2 Analysis Input and Assumptions

The following input and assumptions were used in the analysis:

1. Timing of Events (4):

    a. Wetwell vented at 1.3 hr
    b. Core melt starts at 3 hr

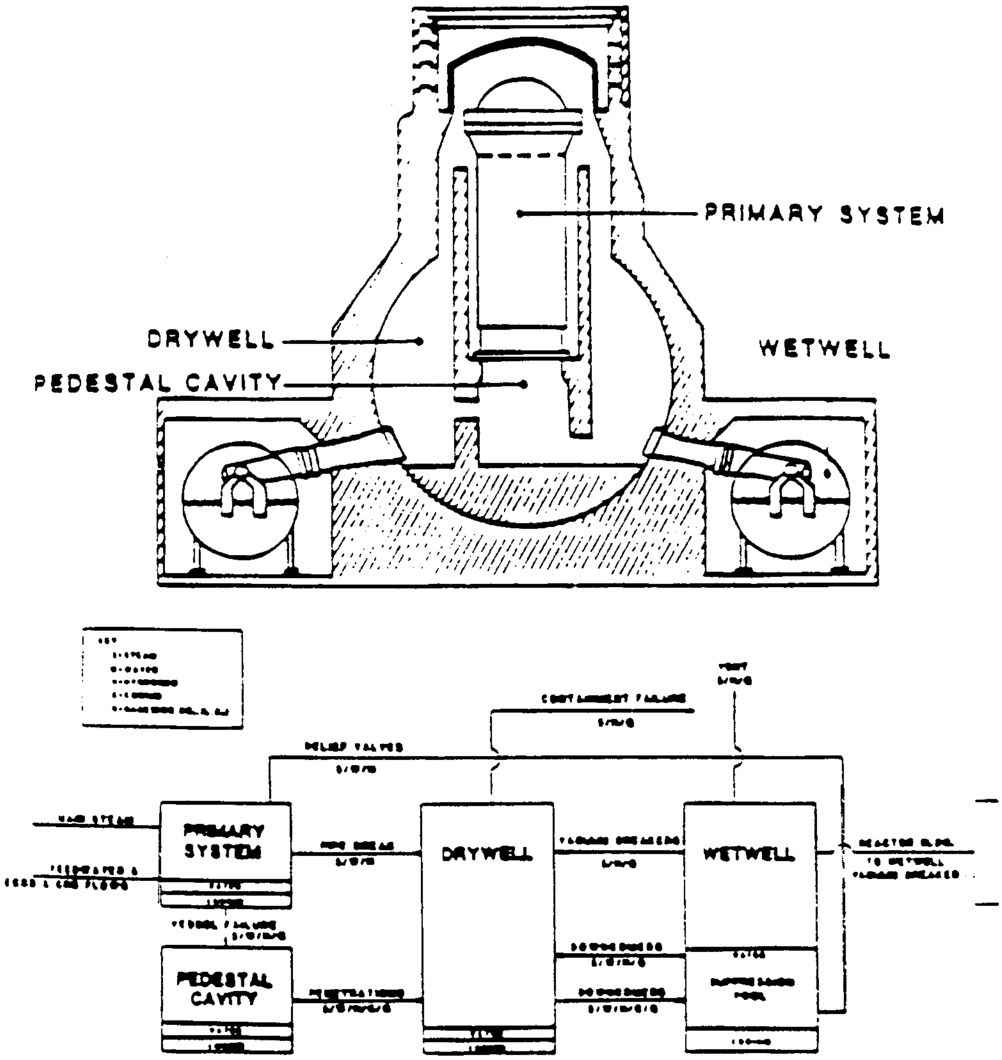

Figure 3-6. Mark I Nodalization

c. Vessel fails at 3.8 hr

d. Containment (drywell) fails at 12.8 hr

e. Doses calculated due to release following containment failure.

2. Initial Inventory: (4)

a.

Nuclide	Mass (Kg)
Kr	25.7
Xe	387.
Cs	207.
I	16.6
Te	34.9
Sr	62.7
Ru	172.
La	98.3

b.

Nuclide	Activity (Ci)
I-131	9.24 + 7
I-132	1.36 + 8
I-133	1.94 + 8
I-134	2.12 + 8
I-135	1.83 + 8

3. Release rates - see figures 3-7 and 3-8 (4).

4. For simplification, the ratio of Cs to I in CsI to be 1.0.

5. The source is uniformly distributed throughout the reactor building.

6. 50% of the iodine released to the reactor building is assumed to be airborne.

7. The reactor building volume = 50,000m^3 (4).

8. The reactor building inner surface area = 17,000m^2 (4).

9. Breathing rate = 1.25 x 10^{6}cc/hr (12).

10. Whole body dose conversion factors are as follows (13):

Nuclide	mrem per pCi inhaled
I-131	2.56 - 6
I-132	1.45 - 7
I-133	5.65 - 7
I-134	7.69 - 8
I-135	3.21 - 7

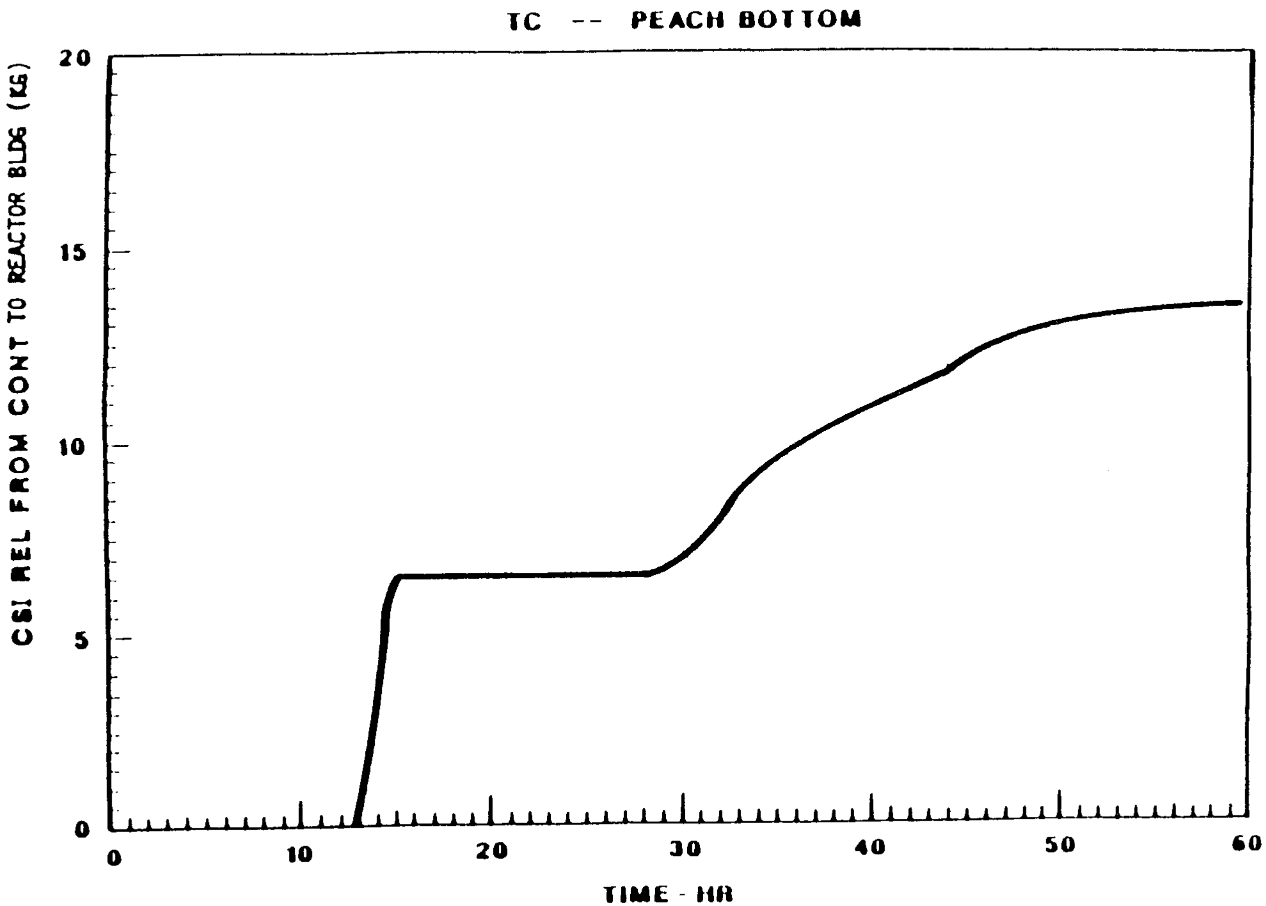

Figure 3-7. Cesium Iodide Released From Containment

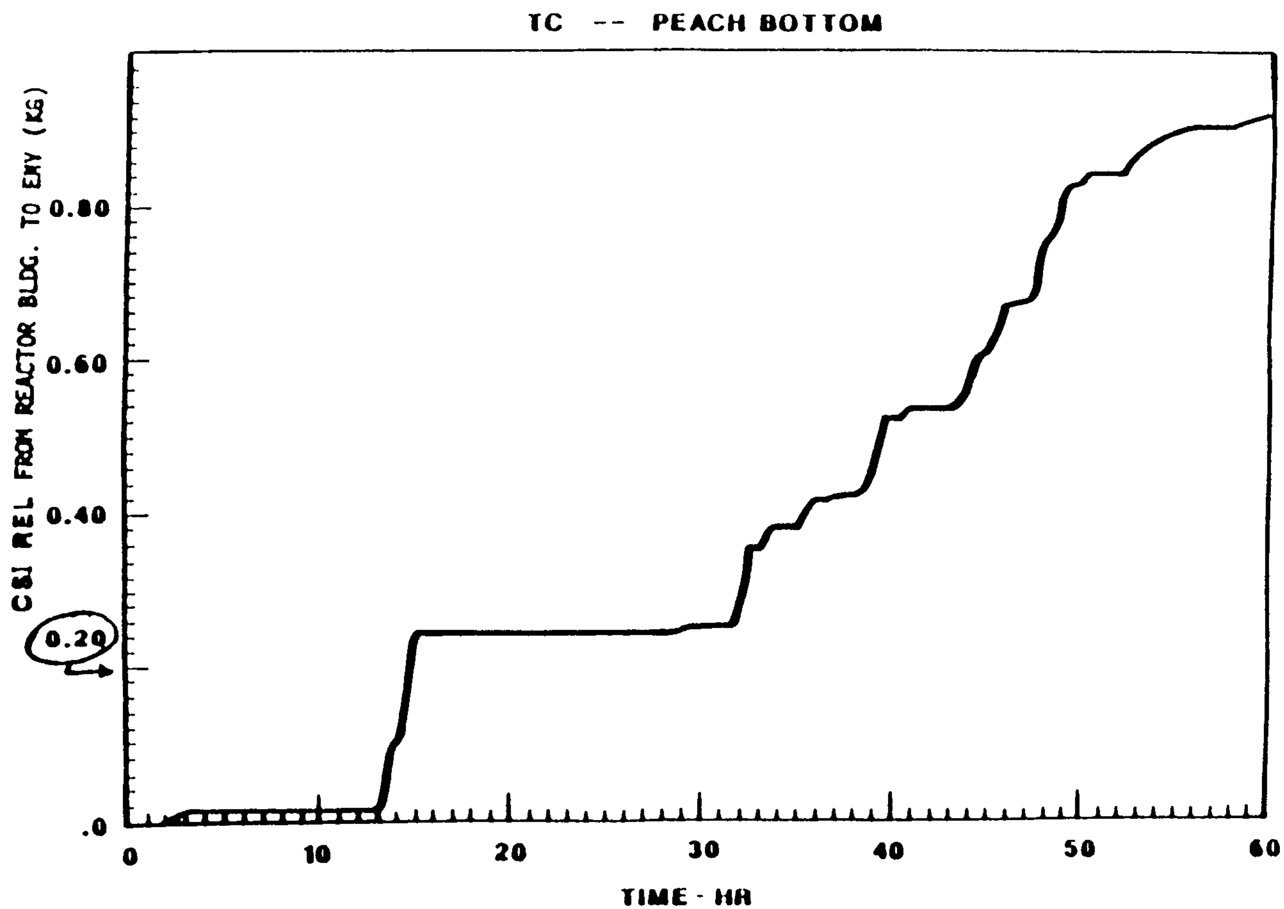

Figure 3-8. Mass of Cesium Iodide Released to Environment

The CsI inventory in the reactor building as a function of time was obtained by subtracting the release to the environment (figure 3-8) from the containment (Figure 3-7).

Between 15 and 30 hr, there is about 6.55 Kg of CsI retained in the reactor building. In order to simplify the analysis in this scoping study, it was assumed that half of this amount is airborne and half is deposited on surfaces. Since roughly half of the mass of CsI is iodine, it was assumed that 1.56 Kg of iodine is airborne.

3.3.3 Committed Dose Rates With Severe Fuel Damage

1. Dose Rate Due to Immersion

The dose rate due to radioactive material airborne in the reactor building was assumed to be dominated by the iodine isotopes. It was assumed 50% of the iodine was airborne and 50% was deposited.

The dose rate due to immersion is given by:

$D_\gamma = 0.25\ E_\gamma X$(rad/sec), from Reg. Guide 1.3 (12)

where:

E_γ = average gamma energy (Mev/dis)
X = airborne concentration (Ci/m^3)

Using the appropriate concentrations at 15 hr, the following immersion dose rates were calculated.

Nuclide	Avg. Energy (MeV/dis)	Concentration (Ci/m^3)	Dose Rate (rem/hr)
I-131	0.375	1.73 + 2	5.8 + 4*
I-132	2.29	2.77 + 0	5.7 + 3
I-133	0.636	2.21 + 2	1.3 + 5
I-135	1.46	7.28 + 1	9.6 + 4
Total			2.9 + 5

*Read as 5.8×10^4, etc.

2. Dose Rate Due to Deposition

It is assumed that 50% of the iodine released into the reactor building plates out. With the reactor building surface area equal to 17,000 m^2 and the deposition dose conversion factors from Regulatory Guide 1.109 (13), the concentrations and doses for the deposited activity at 15 hours are as follows:

Nuclide	Concentration (pCi/m^2)	Dose Conversion Factor (mrem/hr per pCi/m^2)	Dose Rate (rem/hr)
I-131	4.71E + 14	2.80E - 9	1.3E + 3
I-132	8.14E + 12	1.70E - 8	1.4E + 2
I-133	3.96E + 14	3.70E - 9	1.5E + 3
I-135	9.13E + 13	1.20E - 8	1.1E + 3
Total			4.0E + 3

3. Committed Dose Per Unit Exposure Interval Due to Inhalation

With a breathing rate of 1.25 x 10^6cc/hr and the appropriate activities and dose conversion factors for each nuclide, the following concentrations and committed whole body doses per unit exposure interval of 1 hour due to inhalation at 15 hours were calculated.

Nuclide	Half-Life (hr)	Committed Dose per Unit Concentration (Ci/cc)	Internal Exposure @ 15 hr (rem/hr) w/o Respiratory Protection	with SCBA (PF=10,000)
I-131	193	1.73 - 4	5.5 + 5*	5.5 +1
I-132	2.30	2.77 - 6	5.0 + 2	5.0 -2
I-133	21.0	2.21 - 4	1.6 + 5	1.6 +1
I-135	6.72	7.28 - 5	2.9 + 4	2.9 +0
Total			7.4 + 5	7.4 +1

*Read as 5.5 x 10^5, etc.

The exposure due to inhalation would decrease due to aerosol settling between 15 and 30 hr; however, that effect has been neglected in this scoping study. The contribution from I-131 would remain constant given the assumption that half of the CsI would remain airborne. However, radioactive decay would substantially reduce the contribution of the shorter half life isotopes.

As can be seen from figures 3-7 and 3-8, at about 60 hr, the amount of CsI in the reactor building would be roughly twice the amount at 15 hr, due to late releases.

As with the inhalation dose rate, the dose rate due to external exposure is expected to remain fairly constant between 15 and 30 hr, and then increase to about double the 15 hour value at 60 hr, due to the late revaporization of CsI from the RCS.

The external whole body dose rates would preclude human entry into the reactor building under these conditions, thus the inhalation exposures are only of theoretical interest.

3.3.4 Exposure from Unfiltered In-Leakage to Control Room

First order approximation dose rates and doses were estimated in the control room based on the ATWS sequence discussed above. Such estimates are very much dependent on the assumptions used to define the amount of radioactive material which may enter the control room and the time it is assumed to be present prior to removal by the control room exhaust system.

The following assumptions were employed in estimating control room dose rates:

1. The release from the containment is assumed to have occurred at 12.8 hours following initiation of the accident (based on the time of containment venting). In order to simplify the analysis, it was assumed to be a puff (instantaneous) release, although the previous discussion indicates it could be comprised of an initial release between 12.8 and 15 hr followed by a gradual release between 30 and 60 hr.
2. Releases from the containment were 100% of the noble gases and 3% (4, table 6.3) of the iodine core inventory.
3. A conservatively high atmospheric dispersion factor (χ/Q) of 1.0 X 10^{-3}sec/m^3 was assumed.
4. Core inventory was as indicated in Section 3.3.2.
5. An unfiltered air in-leakage of 100 cfm was assumed to occur prior to purge of the control room.
6. A control room purge rate of 10,000 cfm was assumed, followed by a ventilation rate of 2,000 cfm.
7. The control room volume was assumed to be 1.0 X 10^5ft^3.
8. The breathing rate, for personnel in the control room, was taken as 1.25 X 10^6cc/hr (12).
9. Control room walls were assumed to be 2 ft thick concrete in determining the external dose from cloud immersion.

Stone & Webster's DRAGON computer program (19) was utilized to calculate external whole body and thyroid doses in the control room. Whole body dose commitments per unit exposure interval were calculated by combining data from the DRAGON runs with whole body dose conversion factors from Regulatory Guide 1.109 (13).

Table 3-5

WHOLE BODY DOSE COMMITMENT PER HOUR IN CONTROL ROOM WITH ATWS PUFF RELEASE

(Assuming 100 CFM unfiltered in-leakage, puff release of 100% noble gases, 3% halogens, $\chi/q = 10^{-3}$ sec/m^3)

Committed Whole Body Dose Per Unit Exposure Interval (rem/hr)

No Control Room Purge

	External Exposure			Internal Exposure From Inhalation	
Time (hr)	Noble Gases	Halogens	Total	w/o Respiratory Protection	Using Air-Purifying Full Face Mask Respirator**(PF = 50)
12.8	0	0	0	0	0
12.9	28	9	37	205	4.1
13.3	26	8	34	198	3.96
14.8	20	6	26	174	3.48
24.8	6	1	7	82	1.64
48.8	0.5	0.1	0.6	16	0.32
12.1	neg.	neg.	neg.	0.2	neg.

10,000 CFM Purge after 6 Min (a)

	External Exposure			Internal Exposure From Inhalation	
Time (hr)	Noble Gases	Halogens	Total	w/o Respiratory Protection	Using Air-Purifying Full Face Mask Respirator**(PF = 50)
12.8	0	0	0	0	0
12.9	28	9	37	205	4.1
13.3	2.4	0.8	3.2	18	0.36
14.8	0.2	0.1	0.3	1.8	neg.
24.8	neg.	neg.	neg.	neg.	neg.
48.8	neg.	neg.	neg.	neg.	neg.
12.1	neg.	neg.	neg.	neg.	neg.

10,000 CFM Purge after 30 Min (b)

	External Exposure			Internal Exposure From Inhalation	
Time (hr)	Noble Gases	Halogens	Total	w/o Respiratory Protection	Using Air-Purifying Full Face Mask Respirator**(PF = 50)
12.8	0	0	0	0	0
12.9	28	9	37	205	4.1
13.3	26	8	34	198	3.96
14.8	neg.	neg.	neg.	neg.	neg.
24.8	neg.	neg.	neg.	neg.	neg.
48.8	neg.	neg.	neg.	neg.	neg.
12.1	neg.	neg.	neg.	neg.	neg.

*Lifetime dose commitment per hour of inhalation.

**Use of SCBA with PF = 10,000 would result in the internal doses to be negligible

(a) Control room purge begins at 12.9 hr following 6 min of 100 cfm unfiltered in-leakage and continues for 30 min.

(b) Control room purge begins at 13.31 hr following 30 min of 100 cfm unfiltered in-leakage and continues for 90 min.

Table 3-6

DOSE COMMITMENT IN CONTROL ROOM WITH ATWS PUFF RELEASE

(Assuming 100 CFM unfiltered in-leakage, puff release of 100% noble gases, 3% halogens, $\chi/Q = 10^{-3}$ sec/m^3)

	External Dose			Internal Exposure from Inhalation		Thyroid Dose Commitment	
	Noble Gases	Halogen	Total	w/o Respiratory Protection	Using Air-Purifying Full Face Mask Respirator (PF = 50)	w/o Respiratory Protection	Using Air-Purifying Full Face Mask (PF = 50)
No Control Room Purge	220	49	270	2,730	54.6	1.7×10^6	3.4×10^4
10,000 CFM Purge @ 6 min for 30 min (a)	8	3	11	61	1.22	3.6×10^4	7.2×10^2
10,000 CFM Purge @ 30 min for 90 min (b)	18	6	24	134	2.68	7.9×10^4	1.6×10^3

*Lifetime dose commitment from inhalation (rem)

(a) Control room purge begins at 12.9 hr following 6 min unfiltered in-leakage and continues for 30 min.

(b) Control room purge begins at 13.3 hr following 30 min unfiltered in-leakage and continues for 90 min.

The dose rate in the control room due to external exposure from the cloud outside the shielded control room was calculated to be 0.038 rem/hr at 12.8 hr after the start of the accident. Consistent with the assumption of an instantaneous release, the dose from outside the building would be expected to be negligible.

The calculated dose rates due to radioactive material which was assumed to enter the control room are listed in table 3-5 based on the assumptions listed above. Three sets of data are presented. The first assumes no control room purge during the times indicated. The second set assumes a 10,000 cfm purge rate for 30 min starting at 6 min after the puff release. The third assumes a 10,000 cfm purge rate for 90 min starting at 30 min after the release. The cumulative dose commitments for the three sets of conditions considered are summarized in table 3-6.

One important aspect of these data is that: the unfiltered in-leakage was assumed to be 100 cfm and that the NRC's Standard Review Plan 6.4 (7) suggests a 10 cfm value, which would reduce the dose rates and dose by a factor of ten. Since control rooms are equipped with respiratory protection devices and the operators are trained in their use, it would be reasonable expected that the internal exposures would be as indicated in tables 3-5 and 3-6.
If a 10 cfm in-leakage rate were substituted for the 100 cfm assumed in this analysis, the following cumulative committed doses would result:

WHOLE BODY DOSE COMMITMENT (REM)

		Internal Exposure*		
	External Exposure	With Air-Purifying Full Face Mask Respirator (PF = 50)	With SCBA (PF=10,000)	Without Respirators
No Control Purge	27	278	0.03	5.6
10,000 cfm Purge @ 6 min for 30 min	1.1	6.1	neg.	0.12
10,000 cfm Purge @ 30 min for 90 min	2.4	13.4	neg.	0.002

*Lifetime dose commitment from inhalation per unit of exposure interval.

The corresponding thyroid dose commitments under these conditions, i.e., with a 10 cfm in-leakage, would be:

THYROID DOSE COMMITMENT (REM)

	With Air-Purifying Full Face Masks Respirator (PF = 50)	With SCBA (PF=10,000)	Without Respirator
No Control Room Purge	3,400	17	170,000
10,000 cfm Purge @ 6 min for 30 min	72	0.36	3,600
10,000 cfm Purge @ 30 min for 90 min	158	0.79	7,900

Furthermore, no credit was taken for filtered recirculation of control room air. A typical system, recirculating at a rate of 4,000 cfm with a 99% effective iodine filter, would reduce the thyroid dose by an additional factor (2 in the case of purge after 30 min).

A major aspect of a more quantified appraisal of control room doses during severe core damage accidents is the fact that the halogens would not be expected to be released in a puff. Analysis of temporal variations in the release from the containment and the prevailing air intake, treatment, and exhaust in the control room are very important parameters in assessments of control room exposures. Evaluation of such parameters is outside the scope of the present very limited scoping study.

Section 4

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