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EMERGENCY DOSE CALCULATION MANUAL (EDCM) INSTRUCTION MEMO
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1.0 PURPOSE

The purpose of this manual is to provide a summary of the mathematics and assumptions used to perform offsite dose projections during emergency situations. This includes calculating projected on-site and off-site doses from releases of radioactive material to the environment in accident conditions upon implementation of the Emergency Plan. As such, this document describes methods of projecting off-site doses during emergencies or for training purposes. Indications of releases may result from Radiation Monitoring System (RMS) readings, on-site or off-site sample results, or contingency calculations, if RMS and sample results are not available.

2.0 APPLICABILITY/SCOPE

The EDCM is applicable to all qualified TMI Radiological Assessment Coordinator (RAC) personnel involved in the projection of on-site and off-site doses during an emergency. This manual provides the calculational methodologies used in performance of dose projections during emergency situations where radioactive material has been or is predicted to be released to the environment. The contents of the EDCM shall be used as the basis for the computations performed in the Emergency Plan RAC computer program.

3.0 DEFINITIONS

3.1 BUILDING WAKE EFFECTS - When an atmospheric release occurs at, near, or below the top of a building (or any structure) the dispersion of the release is affected by the wake effect of the building. Air flow over and around the structure from the prevailing wind tends to drive the release down to the ground on the downwind side of the structure. This has two effects: it increases on-site concentrations dramatically, while slightly reducing concentrations downwind for a short distance. Far downwind concentrations are affected very little by building wake. Building wake effects are most noticeable for ground level or low flow stack releases such as the condenser off-gas exhaust. Normal plant ventilation usually has a high enough flow that building wake does not affect the plume significantly. Building wake is accounted for as part of the split wake release modeling.

3.2 "CHI over Q" (X/Q) - is the dispersion of a gaseous release in the environment calculated by the split wake dispersion model. Normal units of X/Q are sec/cubic meter. X/Q is used to determine environmental atmospheric concentrations by multiplying the source term represented by Q. Thus dispersion, X/Q (sec/cubic meter) times source term, Q (μCi/sec) yields environmental concentration X (μCi/cubic meter). X/Q is a function of many parameters including wind speed, delta T (change in temperature with height), release point height, building size, and release velocity, among others. The release model takes all these into account when calculating atmospheric dispersion.

3.3 CONTAINMENT ATMOSPHERIC POST-ACCIDENT SAMPLING SYSTEM (CATPASS) - Post accident sampling system capable of providing sample(s) following an accident condition, coincident with a blackout, with limited personnel exposure. The sampling system, located in a post-accident accessible area, provides the capability for obtaining samples of the Reactor Building atmosphere, within one hour after the decision has been made to acquire the sample(s). The samples(s) are then used for radiological and hydrogen analysis. These results will provide an indication of the extent of core damage and provide good data for the Reactor Building source term if a Reactor Building release is possible.

3.4 CONTINGENCY CALCULATION - A source term calculation performed in the absence of sufficient effluent radiation monitoring system readings or post accident sample data. It is a mathematical calculation based upon the most representative physical model of actual accident plant conditions.

3.5 CORE DAMAGE - (Note: This definition to be used in lieu of defective/failed fuel.) A set of core classifications used to address the requirements of the NRC NUREG 0737 Criterion 2(a) upon implementation of the Emergency Plan. Based upon RCS pressure and incore thermocouple readings, an assessment is made of the degree of cladding failure, fuel overheat, and fuel melt.

3.6 DEFECTIVE FUEL/FAILED FUEL - See definition of core damage.

3.7 DOSE RATE CONVERSION FACTOR (DRCF) - A parameter calculated by the methods and models of internal dosimetry, which indicates the Committed Dose Equivalent (CDE) to an organ or Total Effective Dose Equivalent (TEDE) to the whole body per unit activity inhaled or ingested. This parameter is specific to the radionuclide and the dose pathway. Dose conversion factors are commonly tabulated in units of mrem/hr per curie/m^3 inhaled or ingested.

3.8 ELEVATED RELEASE - An airborne effluent plume which is well above any building wake effects so as to be essentially unentrained is termed an elevated release. The source of the plume may be elevated either by virtue of the physical height of the source above the ground elevation and buildings or by a combination of the physical height and the jet plume rise. Semi infinite modeling of elevated releases generally will not produce any significant ground level concentrations within the first few hundred yards of the source. Semi infinite modeling of elevated releases generally have less dose consequence to the public due to the greater downwind distance to the ground concentration maximum compared to ground releases. Elevated releases as used in the EDCM actually means "not at ground" in the split wake plume model. Other definitions of "elevated" with respect to plumes abound in literature.

3.9 EMERGENCY ACTION LEVEL (EAL) - Predetermined conditions or values, including radiation dose rates; specific levels of airborne; waterborne; or surface-deposited contamination; events such as natural disasters or fires; or specific instrument indicators which, when reached or exceeded, require implementation of the Emergency Plan.

3.10 EMERGENCY DIRECTOR (ED) - Designated on-site individual having the responsibility and authority to implement the Emergency Plan, and who will coordinate efforts to limit consequences of, and bring under control, the emergency.

3.11 EMERGENCY DOSE CALCULATION MANUAL (EDCM) - This controlled dose calculation manual is the documentation describing the content and calculational methods of the TMI Radiological Assessment Coordinator (RAC) model.

3.12 EMERGENCY OPERATIONS FACILITY (EOF) - The Emergency Operations Facilities serve as the primary locations for management of the Corporation's overall emergency response. These facilities are equipped for and staffed by the Emergency Support Organization to coordinate emergency response with off-site support agencies and to assess the environmental impact of the emergency. The EOF participates in accident assessment and transmits appropriate data and recommended protective actions to Federal, State and Local agencies.

3.13 EMERGENCY PLANNING ZONE (EPZ) - There are two Emergency Planning Zones. The first is an area, approximately 10 miles in radius around the site, for which emergency planning consideration of the plume exposure pathway has been given in order to assure that prompt and effective actions can be taken to protect the public and property in the event of an accident. This is called the Plume Exposure Pathway EPZ. The second is an area approximately 50 miles in radius around the site, for which emergency planning consideration of the ingestion exposure pathway has been given. This is called the Ingestion Exposure Pathway EPZ.

3.14 ENTRAINMENT - When a release is treated as a wake split release an entrainment factor is applied to specify how much of the release is to be considered elevated and how much is to be considered a ground release. Entrainment factor is related to the building wake effect. The entrainment factor is computed on a case by case basis and is dependent on both the stack exit velocity and the wind speed. At low wind speeds and high exit velocities, building effects are lowest and the entrainment factor selects for elevated release. At high wind speeds and/or low exit velocities the building effect is highest and the entrainment factor results in a ground level release. Intermediate conditions cause entrainment factors which will split the release between ground and elevated. The general form for the application of the entrainment factor (Ef) is:

X/Q(splitwake)=X/Q(ground)*Ef + X/Q(elevated)*(1-Ef).

As can be seen from the formula, when the entrainment factor is one, the release is entirely ground and when the entrainment factor is zero, the release is entirely elevated. When 0 < Ef < 1 then the release is split.

3.15 EPICOR II - Radioactive Liquid Waste Processing Facility located on the east side of TMI-2. This facility is used to process radioactive liquid waste. A Victoreen radiation monitor is located on the ventilation system of this facility. The ventilation system average flow rate is 9000 CFM.

3.16 EXCLUSION AREA (EA) - As defined in 10CFR100.3; "that area surrounding the reactor, in which the reactor licensee has the authority to determine all activities including exclusion or removal of personnel and property from the area". At TMI this is an area with a 2000 ft. radius from the point equidistant between the centers of the TMI-1 and TMI-2 reactor buildings.

3.17 EXIT VELOCITY AND PLUME RISE - Atmospheric dispersion and ground concentrations are in part dependent on release height. Higher release heights will cause lower maximum concentrations at ground and will cause that maximum to occur further downwind than would a lower release height. The effective height of a stack is not only dependent on its physical height, but also on whether the plume rises or not. At high linear flow rates (exit velocity), the release plume behaves much like a geyser and rises in a jet flow above the stack. The height to which the jet flow rises becomes the effective stack height.

3.18 FINITE PLUME MODEL - Atmospheric dispersion and dose assessment model which is based on the assumption that the horizontal and vertical dimensions of an effluent plume are not necessarily large compared to the distance that gamma rays can travel in air. It is more realistic than the semi-infinite plume model because it considers the finite dimensions of the plume, the radiation build-up factor, and the air attenuation of the gamma rays coming from the cloud. This model can estimate the dose to a receptor who is not submerged in the radioactive cloud. It is particularly useful in evaluating doses from an elevated plume or when the receptor is near the effluent source. This model is used by the MIDAS computer program.

3.19 FUEL HANDLING BUILDING ENGINEERED SAFETY FEATURE VENTILATION SYSTEM - The Fuel Handling Building ESF Ventilation System is installed to contain, confine, control, mitigate, monitor and record radiation release resulting from a TMI-1 postulated spent fuel accident in the Fuel Handling Building as described in FSAR, Section 14.2.2.1. Normal operation of the Fuel Handling Building ESF Ventilation System is during TMI-1 spent fuel movements in the Fuel Handling Building. The system design includes adequate air filtration and exhaust capacity to ensure that no uncontrolled radioactive release to atmosphere occurs. The System includes effluent radiation monitoring capability.

3.20 GAUSSIAN PLUME EQUATION - An equation which takes input parameters of plume height, sigma-Y, sigma-Z, and wind speed, which explicitly calculates the straight line Gaussian Plume Dispersion. The Gaussian Plume equation actually averages short term variations to produce a mean effective plume, so short term measurements of the plume may not be duplicated by the Gaussian Plume Model.

3.21 GROUND RELEASE - An airborne effluent plume which contacts the ground essentially at the point of release either from a source actually located at the ground elevation or from a source well above the ground elevation which has significant building wake effects to cause the plume to be entrained in the wake and driven to the ground elevation is termed a ground level release. Ground level releases are treated differently than elevated releases in that the X/Q calculation results in significantly higher concentrations at the ground elevation near the release point. Ground releases also have generally lower X/Qs all the way downwind.

3.22 HYDROGEN PURGE SYSTEM - Post-accident containment purge system is designed to maintain the hydrogen concentration of the post-accident containment atmosphere below the lower flammability limit. The system does this by introducing outside air into the Reactor Building, which allows the displaced containment atmosphere to be discharged in a controlled manner into the normal Reactor Building exhaust duct. In the flow path three release rates exist which can be additive to give flow from 5 to 1250 CFM.

3.23 INTERIM SOLID WASTE STAGING FACILITY (ISWSF) - This facility has no ventilation system or radiation monitor, but has the potential to release radioactive material to the environment. Releases would be assessed using field team data and/or MIDAS.

3.24 LOW POPULATION ZONE (LPZ) - As defined in 10CFR100.3 "the area immediately surrounding the exclusion area which contains residents, the total number and density of which are such that there is a reasonable probability that appropriate protective measures could be taken in their behalf in the event of a serious accident.

3.25 METEOROLOGICAL INFORMATION AND DOSE ASSESSMENT SYSTEM (MIDAS) - This is the acronym for the computer program that can be used by the Environmental Assessment Command Center (EACC) to project off-site doses for routine effluents and releases during emergencies. Some features of MIDAS that are not in the RAC program are ingestion pathway doses, liquid and gas population doses, dose projections at any desired point of interest, and sector dose integration.

3.26 FUEL DAMAGE CLASS - A method of estimating the extent of core damage per NUREG-0737 Criterion 2 (a) under accident conditions requiring implementation of the Emergency Plan. The initial estimate of the degree of reactor core damage is derived from the average of the five highest incore thermocouples and RCS pressure. The assessment is performed utilizing a matrix that consists of ten (10) possible damage categories ranging from "no damage" to "major clad damage plus fuel melting".

3.27 PARTITION FACTOR - (Condenser), see NUREG-0017

3.28 RCS POST-ACCIDENT SAMPLING SYSTEM (PASS) - System used for acquiring a pressurized liquid sample of the RCS during emergency conditions. The post-accident reactor coolant sampling system provides a means of obtaining and analyzing a representative sample of reactor coolant, including dissolved gases, reactor coolant bleed tank contents and reactor containment sump contents, within three hours after the decision to acquire the sample, without excessive operator exposure or compromise of interfacing safety-related systems.

3.29 PLANT SHUTDOWN RADIOIODINE SPIKING -

Radioiodine spiking occurs when the reactor is shutdown, and is caused when the fuel gap activity is washed out of fuel rods with cladding defects.

3.30 PROTECTIVE ACTION GUIDE (PAG) - Projected radiological dose or dose commitment values to individuals of the general population and to emergency workers that warrant protective action before or after a release of radioactive material. Protective actions would be warranted provided the reduction in individual dose expected to be achieved by carrying out the protective action is not offset by excessive risks to individual safety in taking the protective action. The protective action guide does not include the dose that has unavoidably occurred prior to the assessment.

3.31 PROTECTIVE ACTION RECOMMENDATION (PAR) - Those actions taken during or after an emergency situation that are intended to minimize or eliminate the hazard to the health and safety of the general public and/or on-site personnel.

3.32 RADIATION MONITORING SYSTEM (RMS) - The RMS detects, indicates, annunciates, and records the radiation level at selected locations inside and outside the plant to verify compliance with applicable Code of Federal Regulations (CFR) limits. The RMS consists of the following subsystems: area monitoring, atmospheric monitoring, and liquid monitoring.

3.33 RADIOIODINE PLATEOUT - Iodines are very chemically reactive, being members of the halogen family. As such, iodines have a very high probability of reacting with almost any other material they come in contact with. Radioiodine plateout is a generic term for the mechanisms by which radioactive iodines are removed from a waste stream by contact with materials not specifically designed or engineered for radioiodine removal. Examples of potential radioiodine plateout reactions are the removal of iodine from gaseous wastes by adsorption onto interior surfaces of ductwork and piping and on any exposed surfaces of the room or building originating the release.

3.34 RADIOIODINE PROCESSOR STATIONS (MAP-5) - System used for acquiring particulate and iodine samples from the Reactor Building Exhaust, Auxiliary and Fuel Handling Building Exhaust or the Condenser Off-gas Exhaust during emergency conditions. The stations are controlled by solenoid valves which activate on high alarm indications on the low gas channels of the effluent stream. Flow is actuated through (3) parallel filter cartridges per station. The sampling times are adjustable on each local control panel. The filter cartridges must be removed manually for analysis.

3.35 RADIOLOGICAL ASSESSMENT COORDINATOR (RAC) - The RAC is responsible for all on-site radiological assessment activities. Initially, the RAC is responsible for directing the on-site and off-site survey teams. The RAC is relieved of off-site radiological monitoring responsibilities by the Environmental Assessment Coordinator. The RAC performs dose projections, based upon source term estimates and provides information to the EAC. Initially the Group Radiological Control Supervisor assumes the role of the RAC until relieved by the Initial Response Emergency Organization RAC, and RASE.

3.36 RADIOLOGICAL ASSESSMENT SUPPORT ENGINEER (RASE) - Individuals assigned to assist the RAC in performing dose calculations, source term calculations, and overall assessment and control of radiological hazards. Normally one RAC and one RASE are on duty at all times.

3.37 REACTOR COOLANT SYSTEM (RCS) - This system contains the necessary piping and components to provide sufficient water flow to cool the reactor. This system provides for the transfer of thermal energy from the reactor core to the once through steam generators (OTSG) to make steam, acts as a moderator for thermal fission, and provides a boundary to separate fission products from the atmosphere.

3.38 RELEASE DURATION - Release duration refers to the time interval during which radionuclides are released from the nuclear facility. Releases may be monitored, unmonitored, actual, or projected. The time interval used to estimate a release of unknown duration should reflect best estimates of the plant technical staff. In the absence of other information, use eight hours as the expected release duration. For purposes of determining whether to take a protective action on the basis of projected dose from an airborne plume, the projected dose should not include the dose that has already been received prior to the time the dose projection is done.

3.39 RELEASE RATE - This term refers to the rate at which radionuclides are released to the environment. Normally, it will be expressed in curies per second (Ci/sec) or microcuries per second (μCi/sec).

3.40 RESPIRATOR AND LAUNDRY MAINTENANCE FACILITY (RLM) - This facility is used to process clean and maintain laundry and respirators for TMI. This facility's 900 CFM ventilation system is monitored using an AMS-3 radiation monitor. The RLM has HEPA filters installed in the ventilation system. Releases would be assessed using field team data and/or MIDAS.

3.41 RMS RESPONSE FACTOR - Parameter which is used to convert RMS monitor count rates to total microcurie per cubic centimeter of the assumed or measured radionuclide spectrum passing by the monitor. This is different from a meter calibration factor which does the same thing for a single calibration nuclide. These factors are adjusted for changes in mixture decay.

3.42 SEMI-INFINITE PLUME MODEL - Dose assessment model which is based on the assumption that the dimensions of an effluent plume are large compared to the distance that gamma rays can travel in air. If the plume dimensions are larger than the gamma ray range, then the radius of the plume might just as well be infinite since radiation emitted from beyond a certain distance will not reach the receptor. The ground is considered to be an infinitely large flat plate and the receptor is assumed to be standing at the center of a hemispherical cloud of infinite radius. The radioactive cloud is limited to the space above the ground plane. This is the origin of the name SEMI-INFINITE PLUME. The noble gas DAC's were developed on the basis of the semi-infinite plume model.

3.43 SIGMA-Y AND SIGMA-Z - Parameters of the Gaussian diffusion equation which determine horizontal and vertical diffusion. Sigma-Y and Sigma-Z varies by stability class and distance from release point.

3.44 SOURCE TERM - A source term is the activity of an actual release or the activity available for release. The common units for the source term are curies, curies/Sec, or multiples thereof (e.g., microcuries). The term "Source Term" derives from the equations involved in doing dose calculations, since the equations contain many terms (a term being mathematical nomenclature for a portion of an equation), the "Source Term" is that portion of the equation which addresses the activity released. Although the term "Source Term" is used loosely to mean almost any activity for airborne, liquids, and even dose rate calculations in plant, strictly speaking "Source Term" applies only to radioactive material actually released.

3.45 SPLIT WAKE RELEASE - Airborne releases, for purposes of assessing off-site dispersion, must address the elevation of the release since wind speed changes with height, buildings affect dispersion for low releases and even wind direction can be different. Many release points are actually at a height where, given different conditions of release flow rate and meteorology, could either be most accurately described as ground or elevated releases, or some mixture between the two. The purpose of treating a release as a split wake release is to address this problem. When a release point is set up to be treated as a split wake release, the atmospheric dispersion is calculated based on a mixture of elevated and ground releases. Thus at high release flow rates the release may appear to be entirely an elevated release and at very low flow rates it may appear to be entirely a ground level release. In intermediate conditions, the model will "split" the release between ground and elevated as appropriate, so that a release might be 25% ground and 75% elevated from the same release point.

3.46 STABILITY CLASS - Dispersion of an effluent plume in the atmosphere is a function of the amount of mixing occurring between the plume and the atmosphere around the plume. The amount of mixing is related to what is referred to as the stability of the atmosphere. Conditions which create good mixing are unstable and conditions which create poorer mixing are stable. Pasquill stability class is a breakdown of the relative atmospheric stability into seven groups, denoted as A through G, from most unstable to most stable. In the pasquill stability class system, stability is related to the relative change in temperature with height, delta T. The more negative the change in temperature with increasing height, the more unstable the atmosphere. The RAC program uses sensors on the Meteorological tower at 33 feet and 150 feet to determine the delta T. Once the delta T is determined, a stability class is selected based on the delta T and the atmospheric dispersion (X/Q) is calculated based on the selected stability class.

3.47 TERRAIN FACTOR - The terrain factor is the terrain height in meters above plant grade. Terrain factor varies with sector and distance from the release point.

3.48 TWO PHASE RELEASE - Liquid and steam release from the main steam safety relief valves. Following discharge to the environment the steam fraction is calculated assuming there is no change in system entropy and that the OTSG wide range level instrument is indicating that the valve inlet fluid condition is either pure liquid or steam (greater than 600 inches as indicated on the Wide Range Level Indicators on PCL Panel, and the pressure indicators PI-950A and PI-952A).

3.49 WASTE HANDLING AND PACKAGING FACILITY (WHPF) - This facility is used to handle and package radioactive waste. This facility's ventilation is monitored by an AMS-3 radiation monitor, and runs at 7100 CFM. Releases would be assessed using field team data and/or MIDAS.

3.50 WIND SPEED ADJUSTMENTS - Since wind speed varies with height and the wind speed sensors are not at the release height, an adjustment is made to extrapolate the measured wind speed to the wind speed at the release height. The adjustment amount is dependent on the stability class.

4.0 PREREQUISITES

4.1 The following are the prerequisites for performance of TMI projected doses using the methods in the EDCM, and the current TMI RAC model.

4.1.1 The Emergency Plan is being implemented.

4.1.2 The RAC station is manned and functional.

5.0 PROCEDURE

This section of the EDCM is divided into the processes that are contained in the RAC model. Listed below is a table of contents for the procedure section of the EDCM:

5.1 Source Term Calculations

5.2 Release Pathways and Characteristics

5.3 Calculation of Fuel Damage Class and Isotopic Percentages

5.4 Radiation Monitoring System (RMS) Source Term Calculation

5.5 Post Accident Samples Source Term Calculation

5.6 Contingency Calculations Source Term Generation

5.7 Decay Scheme Calculation

5.8 Noble Gas to Iodine Ratio Calculations

5.9 Effluent Release Flow Rates

5.10 Two-Phase Steam Flow Determination

5.11 Source Term Filtration

5.12 Dispersion Model

5.13 Liquid Release Calculation

5.14 Off-Site Air Sample Analysis

5.15 Protective Action Recommendation Logic

5.16 Dose Projection Model Overview

Each part of this section explains what each process does and how it does it.

5.1 Source Term Calculations - The source term portion of the TMI dose assessment model is used to generate the quantity and radionuclide make up of the radioactive material released (or available for release) to the environment. Once the source term is measured or estimated, meteorological and dosimetry models are applied to the assessment.

5.2 Release Pathways and Characteristics

5.2.1 The following are the Release Pathways:

1. OTSG Tube Rupture Release

   - Includes: via the condenser off-gas or directly to atmosphere.

2. Reactor Building Release

3. Station Vent Release

   - Includes: Auxiliary Building, Fuel Handling Building (FHB) and ESF FHB.

5.2.2 The following are the Release Characteristics

1. OTSG Tube Rupture via condenser off-gas

2. OTSG Tube Rupture directly to atmosphere via the Main Steam Reliefs or Atmospheric Dump Valves

3. LOCA in the Reactor Building

4. Fuel Handling Accident in the Reactor Building

5. Fuel Handling Accident in the Fuel Handling Building, including ESF Fuel Handling Building Releases

6. LOCA in the Auxiliary Building

7. Waste Gas Tank Release

5.2.3 The following methods are used for source term generation:

1. Use Post Accident Sample Result
2. Use RMS
3. Use Contingency Calculation

NOTE

The least accurate to the most accurate source term generation methodology is contingency calculations, RMS calculations, and post accident sample result calculations.

5.2.4 Airborne releases from the following pathways are evaluated (See Figure 5.9-4):

A. The OTSG Tube Rupture:

1. RM-A-5 Condenser Off-gas
2. RM-A-5 High-Condenser Off-gas
3. RM-G-25 Condenser Off-gas
4. RM-G-26 Main Steam Reliefs and Atmospheric Dump Valves
5. RM-G-27 Main Steam Reliefs and Atmospheric Dump Valves
6. RM-A-5 MAP-5 Samples
7. Main Steam Release directly to the atmosphere
8. Contingency Calculations without RMS or Post Accident Samples

B. The Reactor Building:

1. RM-A-9 Reactor Building Purge
2. RM-A-9 High-Reactor Building Purge
3. RM-G-24 High High-Reactor Building Purge
4. RM-A-2 Reactor Building Atmosphere
5. CATPASS Samples
6. RM-A-9 MAP-5 Samples
7. Contingency Calculations without RMS or Post Accident Samples

C. The Station Vent:

1. RM-A-4 Fuel Handling Building Exhaust
2. RM-A-6 Auxiliary Building Exhaust
3. RM-A-8 Station Vent (Auxiliary and Fuel Handling Buildings)
4. RM-A-8 High-Auxiliary Building Exhaust
5. RM-A-14 ESF Fuel Handling Building Exhaust
6. RM-A-8 MAP-5 Samples

5.3 Calculation of Fuel Damage Class and Isotopic Percentages - This calculation will determine the mix or percentages of the following fifteen radionuclides.

5.3.1

Ten Noble Gases	Five Radioiodines
1. Kr-85m	1. I-131
2. Kr-85	2. I-132
3. Kr-87	3. I-133
4. Kr-88	4. I-134
5. Xe-131m	5. I-135
6. Xe-133m	
7. Xe-133	
8. Xe-135m	
9. Xe-135	
10. Xe-138	

5.3.2 The Fuel Damage Class Determination

The determination of the Fuel Damage Class is performed using various core temperature regions from Operations Procedure 1210-8, see Figure 5.3-1. The core temperatures used in the model are the average of the five highest incore thermocouples. The curves relating to saturation, and cladding failures are approximated by straight line equations. Fuel damage classes 1 - 10 are based on the different pressure and temperature regions of Figure 5.3-1.

5.3.2.1 Core Temperature Regions - Figure 5.3-1

The region to the left of curve C represents normal RCS activity, Fuel Damage Class-1. The region between curves C and D represents RCS plus a percentage of gap activity, Fuel Damage Class 2 - 4. The region between Curve D and Curve E represents RCS plus all gap activity plus a percentage of noble and volatile fission product release from fuel grain boundaries, (CS, I, Rb), Fuel Damage Class 5 - 7. The region between Curve E and 2550°F incore temperature represents RCS activity plus 100% of the gap activity and 100% of the in vessel melt release assuming NUREG-1228 release fractions, Fuel Damage Class 8 - 10.

5.3.2.2 The curves represented in Figure 5.3-1 have the following equations:

- Curve C: 1400F Tclad Curve

Temperature = 406.1 + (0.34027 * PRESS)-(0.0000538 * $PRESS^2$)

- Curve D: 1800F Tclad Curve

Temperature = 690.92 + (0.37178 * PRESS)-(0.0000554 * $PRESS^2$)

- Curve E: 2200F Tclad Curve

Temperature = 1135.3 + (0.40018 * PRESS)-(0.0000567 * $PRESS^2$)

Where: Temperature = Incore Thermocouple Temperature (F°);

PRESS = RCS Indicated Pressure (Psig)

5.3.2.3 The matrix below shows the theory of fuel damage based on TDR-431.

Degree of Degradation	Fuel Damage Class Minor < 10%	Intermediate 10 - 50%	Major > 50%
No Fuel Damage (RCS)	-No Damage :	Class 1 or Class 1A	
Cladding Failure (GAP)	2	3	4
Fuel Overheat (Fuel Matrix)	5	6	7
Fuel Melt (Fuel Matrix)	8	9	10

5.3.3 Calculation of Radionuclide Mix Percentages Based on Fuel Damage Classification. Once the determination of Fuel Damage Class 1 - 10 has been determined from the Core Temperature Regions the various radionuclide mix percentages can be calculated based on the distribution of the RCS Activity, GAP Activity, and/or Fuel Matrix Activity. Various combinations of activities for each Fuel Damage Class are modeled as follows:

Fuel Damage Class	RCS Activity Fraction	GAP Activity Fraction	Fuel Matrix Fraction
1	1	0.0	0.0
2	1	0.1	0.0
3	1	0.5	0.0
4	1	1	0.0
5	1	1	0.1
6	1	1	0.5
7	1	1	1
8	1	1	1
9	1	1	1
10	1	1	1

5.3.3.1 Therefore, as an example: Fuel Damage Class-6 would consist of 100% RCS Activity, plus 100% of the GAP Activity, plus 50% of the Fuel Matrix Activity.

5.3.3.2 TMI-1 Normal RCS Activity - TMI-1 normal RCS Activity is monitored regularly during operation and is readily available to emergency dose assessment personnel.

5.3.3.2.1 Plant Shutdown Radioiodine Spiking represents the "spiking" of the radioiodines and noble gases. Plant Shutdown Radioiodine Spiking occurs when the reactor is shutdown, and is caused when the fuel gap activity is washed out of fuel rods with cladding defects.

Default spiking factors for current fuel conditions are monitored by Radiological Engineering and Nuclear Engineering, and made available to emergency dose assessment personnel.

The TSC should be requested to provide the actual "spiking" factors for the particular plant shutdown situation. The default factors are to be used if information is not available from the TSC.

This "spiking" of radioiodine and noble gas activities represent an increase in RCS radioactivity due to a plant evolution and do not represent an indication of fuel damage; i.e., Fuel Damage Classes 2-10. Spiking only occurs when fuel defects are present.

5.3.3.3 Gap Activity and Fuel Matrix Activity - Gap activity and Fuel Matrix activity used in the model are determined from TDR-431.

TMI Unit 1					
Isotope	GAP Act. Curies	GAP %		Fuel Matrix Act. Curies	Fuel Matrix %
Kr-85m	4.84E+04	0.45		2.13E+07	2.36
Kr-85	7.48E+04	0.70		8.59E+04	0.01
Kr-87	2.63E+04	0.25		3.90E+07	4.33
Kr-88	6.67E+04	0.63		5.91E+07	6.56
Xe-131m	7.96E+04	0.75		5.40E+05	0.06
Xe-133m	9.30E+04	0.87		3.09E+06	0.34
Xe-133	8.34E+06	78.31		1.28E+08	14.20
Xe-135m	2.72E+04	0.26		3.37E+07	3.74
Xe-135	3.45E+04	0.32		1.59E+07	1.76
Xe-138	0.00E+00	0.00		0.00E+00	0.00
I-131	1.29E+06	12.11		6.37E+07	7.07
I-132	1.85E+05	1.74		9.70E+07	10.76
I-133	2.79E+05	2.62		1.43E+08	15.86
I-134	1.74E+04	0.16		1.67E+08	18.53
I-135	8.83E+04	0.83		1.30E+08	14.42
Sum	1.13E+07	100.0	Sum	9.02E+08	100.0

5.3.3.4 The following tables represent the Fuel Damage Class 2 - 10 mixes, percentages, curies, and concentrations used in the RAC model.

	Damage Class 2			Damage Class 3		
	µCi/cc**	Percent	Curies	µCi/cc**	Percent	Curies
I-131	6.03E+02	12.09	1.29E+05	3.01E+03	12.11	6.45E+05
I-132	8.66E+01	1.74	1.85E+04	4.32E+02	1.74	9.25E+04
I-133	1.31E+02	2.62	2.79E+04	6.52E+02	2.62	1.40E+05
I-134	8.43E+00	0.17	1.80E+03	4.10E+01	0.16	8.76E+03
I-135	4.15E+01	0.83	8.88E+03	2.07E+02	0.83	4.42E+04
Subtotal	8.70E+02	17.45	1.86E+05	4.35E+03	17.46	9.30E+05
KR-85M	2.28E+01	0.46	4.87E+03	1.13E+02	0.45	2.42E+04
KR-85	3.50E+01	0.70	7.48E+03	1.75E+02	0.70	3.74E+04
KR-87	1.25E+01	0.25	2.67E+03	6.16E+01	0.25	1.32E+04
KR-88	3.15E+01	0.63	6.73E+03	1.56E+02	0.63	3.34E+04
XE-131M	3.72E+01	0.75	7.96E+03	1.86E+02	0.75	3.98E+04
XE-133M	4.36E+01	0.87	9.32E+03	2.17E+02	0.87	4.65E+04
XE-133	3.90E+03	78.29	8.35E+05	1.95E+04	78.30	4.17E+06
XE-135M	1.28E+01	0.26	2.74E+03	6.37E+01	0.26	1.36E+04
XE-135	1.72E+01	0.34	3.68E+03	8.17E+01	0.33	1.75E+04
XE-138	1.70E-01	0.00	3.64E+01	1.70E-01	0.00	3.64E+01
Subtotal	4.12E+03	82.55	8.81E+05	2.05E+04	82.54	4.40E+06
Total	4.99E+03	100.00	1.07E+06	2.49E+04	100.00	5.33E+06
Noble Gas To Iodine Ratio	4.73	4.73	4.73	4.73	4.73	4.73

** THESE µCi/cc VALUES ARE BASED ON NORMAL RCS VOLUME OF 66,595 GALLONS

	Damage Class 4			Damage Class 5		
	μCi/cc**	Percent	Curies	μCi/cc**	Percent	Curies
I-131	6.03E+03	12.11	1.29E+06	3.58E+04	7.60	7.66E+06
I-132	8.65E+02	1.74	1.85E+05	4.62E+04	9.81	9.89E+06
I-133	1.30E+03	2.62	2.79E+05	6.81E+04	14.46	1.46E+07
I-134	8.16E+01	0.16	1.75E+04	7.81E+04	16.59	1.67E+07
I-135	4.13E+02	0.83	8.83E+04	6.12E+04	12.99	1.31E+07
Subtotal	8.69E+03	17.46	1.86E+06	2.89E+05	61.44	6.19E+07
KR-85M	2.26E+02	0.45	4.84E+04	1.02E+04	2.16	2.18E+06
KR-85	3.50E+02	0.70	7.48E+04	3.90E+02	0.08	8.34E+04
KR-87	1.23E+02	0.25	2.63E+04	1.83E+04	3.90	3.93E+06
KR-88	3.12E+02	0.63	6.68E+04	2.79E+04	5.93	5.98E+06
XE-131M	3.72E+02	0.75	7.96E+04	6.24E+02	0.13	1.34E+05
XE-133M	4.35E+02	0.87	9.30E+04	1.88E+03	0.40	4.02E+05
XE-133	3.90E+04	78.31	8.34E+06	9.88E+04	20.97	2.11E+07
XE-135M	1.27E+02	0.26	2.72E+04	1.59E+04	3.37	3.40E+06
XE-135	1.62E+02	0.33	3.47E+04	7.59E+03	1.61	1.62E+06
XE-138	1.70E-01	0.00	3.64E+01	1.70E-01	0.00	3.64E+01
Subtotal	4.11E+04	82.54	8.79E+06	1.82E+05	38.56	3.89E+07
Total	4.98E+04	100.00	1.07E+07	4.71E+05	100.00	1.01E+08
Noble Gas To Iodine Ratio	4.73	4.73	4.73	0.63	0.63	0.63

**THESE μCi/cc VALUES ARE BASED ON NORMAL RCS VOLUME OF 66,595 GALLONS

	Damage Class 6			Damage Class 7		
	μCi/cc**	Percent	Curies	μCi/cc**	Percent	Curies
I-131	1.55E+05	7.18	3.31E+07	3.04E+05	7.13	6.50E+07
I-132	2.28E+05	10.55	4.87E+07	4.54E+05	10.66	9.72E+07
I-133	3.35E+05	15.56	7.18E+07	6.70E+05	15.71	1.43E+08
I-134	3.90E+05	18.10	8.35E+07	7.80E+05	18.31	1.67E+08
I-135	3.04E+05	14.11	6.51E+07	6.08E+05	14.26	1.30E+08
Subtotal	1.41E+06	65.50	3.02E+08	2.82E+06	66.07	6.03E+08
KR-85M	5.00E+04	2.32	1.07E+07	9.98E+04	2.34	2.13E+07
KR-85	5.50E+02	0.03	1.18E+05	7.51E+02	0.02	1.61E+05
KR-87	9.12E+04	4.23	1.95E+07	1.82E+05	4.28	3.90E+07
KR-88	1.38E+05	6.42	2.96E+07	2.76E+05	6.49	5.92E+07
XE-131M	1.63E+03	0.08	3.50E+05	2.90E+03	0.07	6.20E+05
XE-133M	7.65E+03	0.36	1.64E+06	1.49E+04	0.35	3.18E+06
XE-133	3.38E+05	15.68	7.23E+07	6.37E+05	14.95	1.36E+08
XE-135M	7.89E+04	3.66	1.69E+07	1.58E+05	3.70	3.37E+07
XE-135	3.73E+04	1.73	7.98E+06	7.45E+04	1.75	1.59E+07
XE-138	1.70E-01	0.00	3.64E+01	1.70E-01	0.00	3.64E+01
Subtotal	7.44E+05	34.50	1.59E+08	1.45E+06	33.93	3.10E+08
Total	2.16E+06	100.00	4.61E+08	4.26E+06	100.00	9.12E+08
Noble Gas To Iodine Ratio	0.53	0.53	0.53	0.51	0.51	0.51

**THESE μCi/cc VALUES ARE BASED ON NORMAL RCS VOLUME OF 66,595 GALLONS

	Damage Class 8			Damage Class 9		
	μCi/cc**	Percent	Curies	μCi/cc**	Percent	Curies
I-131	3.04E+05	7.13	6.50E+07	3.04E+05	7.13	6.50E+07
I-132	4.54E+05	10.66	9.72E+07	4.54E+05	10.66	9.72E+07
I-133	6.70E+05	15.71	1.43E+08	6.70E+05	15.71	1.43E+08
I-134	7.80E+05	18.31	1.67E+08	7.80E+05	18.31	1.67E+08
I-135	6.08E+05	14.26	1.30E+08	6.08E+05	14.26	1.30E+08
Subtotal	2.82E+06	66.07	6.03E+08	2.82E+06	66.07	6.03E+08
KR-85M	9.98E+04	2.34	2.13E+07	9.98E+04	2.34	2.13E+07
KR-85	7.51E+02	0.02	1.61E+05	7.51E+02	0.02	1.61E+05
KR-87	1.82E+05	4.28	3.90E+07	1.82E+05	4.28	3.90E+07
KR-88	2.76E+05	6.49	5.92E+07	2.76E+05	6.49	5.92E+07
XE-131M	2.90E+03	0.07	6.20E+05	2.90E+03	0.07	6.20E+05
XE-133M	1.49E+04	0.35	3.18E+06	1.49E+04	0.35	3.18E+06
XE-133	6.37E+05	14.95	1.36E+08	6.37E+05	14.95	1.36E+08
XE-135M	1.58E+05	3.70	3.37E+07	1.58E+05	3.70	3.37E+07
XE-135	7.45E+04	1.75	1.59E+07	7.45E+04	1.75	1.59E+07
XE-138	1.70E-01	0.00	3.64E+01	1.70E-01	0.00	3.64E+01
Subtotal	1.45E+06	33.93	3.10E+08	1.45E+06	33.93	3.10E+08
Total	4.26E+06	100.00	9.12E+08	4.26E+06	100.00	9.12E+08
Noble Gas To Iodine Ratio	0.51	0.51	0.51	0.51	0.51	0.51

**THESE μCi/cc VALUES ARE BASED ON NORMAL RCS VOLUME OF 66,595 GALLONS

Damage Class 10			
	μCi/cc**	Percent	Curies
I-131	3.04E+05	7.13	6.50E+07
I-132	4.54E+05	10.66	9.72E+07
I-133	6.70E+05	15.71	1.43E+08
I-134	7.80E+05	18.31	1.67E+08
I-135	6.08E+05	14.26	1.30E+08
Subtotal	2.82E+06	66.07	6.03E+08
KR-85M	9.98E+04	2.34	2.13E+07
KR-85	7.51E+02	0.02	1.61E+05
KR-87	1.82E+05	4.28	3.90E+07
KR-88	2.76E+05	6.49	5.92E+07
XE-131M	2.90E+03	0.07	6.20E+05
XE-133M	1.49E+04	0.35	3.18E+06
XE-133	6.37E+05	14.95	1.36E+08
XE-135M	1.58E+05	3.70	3.37E+07
XE-135	7.45E+04	1.75	1.59E+07
XE-138	1.70E-01	0.00	3.64E+01
Subtotal	1.45E+06	33.93	3.10E+08
Total	4.26E+06	100.00	9.12E+08
Noble Gas To Iodine Ratio	0.51	0.51	0.51

**THESE μCi/cc VALUES ARE BASED ON NORMAL RCS VOLUME OF 66,595 GALLONS

5.3.3.5 Calculation of Radionuclide Mix Percentages - Once the total amount of combined activities or curies for a certain Fuel Damage Class has been determined, these curies are then normalized to 100%, i.e., the percentage of each radionuclide in the total mix is calculated.

Exceptions - The above percentages are replaced in cases where an assumed mix is more appropriate. These cases are:

1. Contingency calculations for:

   a. Spent Fuel Accident in the Fuel Handling Building - FSAR mix is assumed.

   b. Fuel Cask Accident in the Fuel Handling Building - FSAR mix is assumed.

   c. Spent Fuel Accident in the Reactor Building - FSAR mix is assumed.

   d. Waste Gas Decay Tank - FSAR mix is assumed.

2. The RCS inputs for these cases are as follows:

RCS ACTIVITY INPUTS (1)

ISOTOPE	(2) WGDT RUPTURE	(3) RB FUEL HANDLING ACCIDENT	(4) FHB FUEL HANDLING ACCIDENT	(5) FUEL CASK ACCIDENT
KR85M	5.56E-03	2.88E-06	9.54E-08	0.00E+00
KR85	3.23E-02	2.10E-02	1.19E-01	1.00E+00
KR87	3.03E-03	0.00E+00	0.00E+00	0.00E+00
KR88	9.79E-03	1.13E-08	0.00E+00	0.00E+00
XE131M	8.68E-03	4.12E-03	8.45E-03	1.07E-04
XE133M	1.01E-02	1.36E-02	5.77E-03	0.00E+00
XE133	9.05E-01	9.53E-01	8.65E-01	2.14E-06
XE135M	3.43E-03	0.00E+00	0.00E+00	0.00E+00
XE135	2.16E-02	1.03E-03	2.62E-04	0.00E+00
XE138	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I131	1.11E-04	6.37E-03	1.72E-03	6.76E-05
I132	1.72E-04	0.00E+00	0.00E+00	0.00E+00
I133	1.32E-04	1.22E-03	4.03E-05	0.00E+00
I134	2.02E-05	0.00E+00	0.00E+00	0.00E+00
I135	7.07E-05	8.06E-06	7.46E-08	0.00E+00
TOTAL	1.00E+00	1.00E+00	1.00E+00	1.00E+00

(1) Normalized isotopic activities released from FSAR analysis to be input into model as RCS activity to get proper monitor response.

(2) From FSAR Table 14.2-21

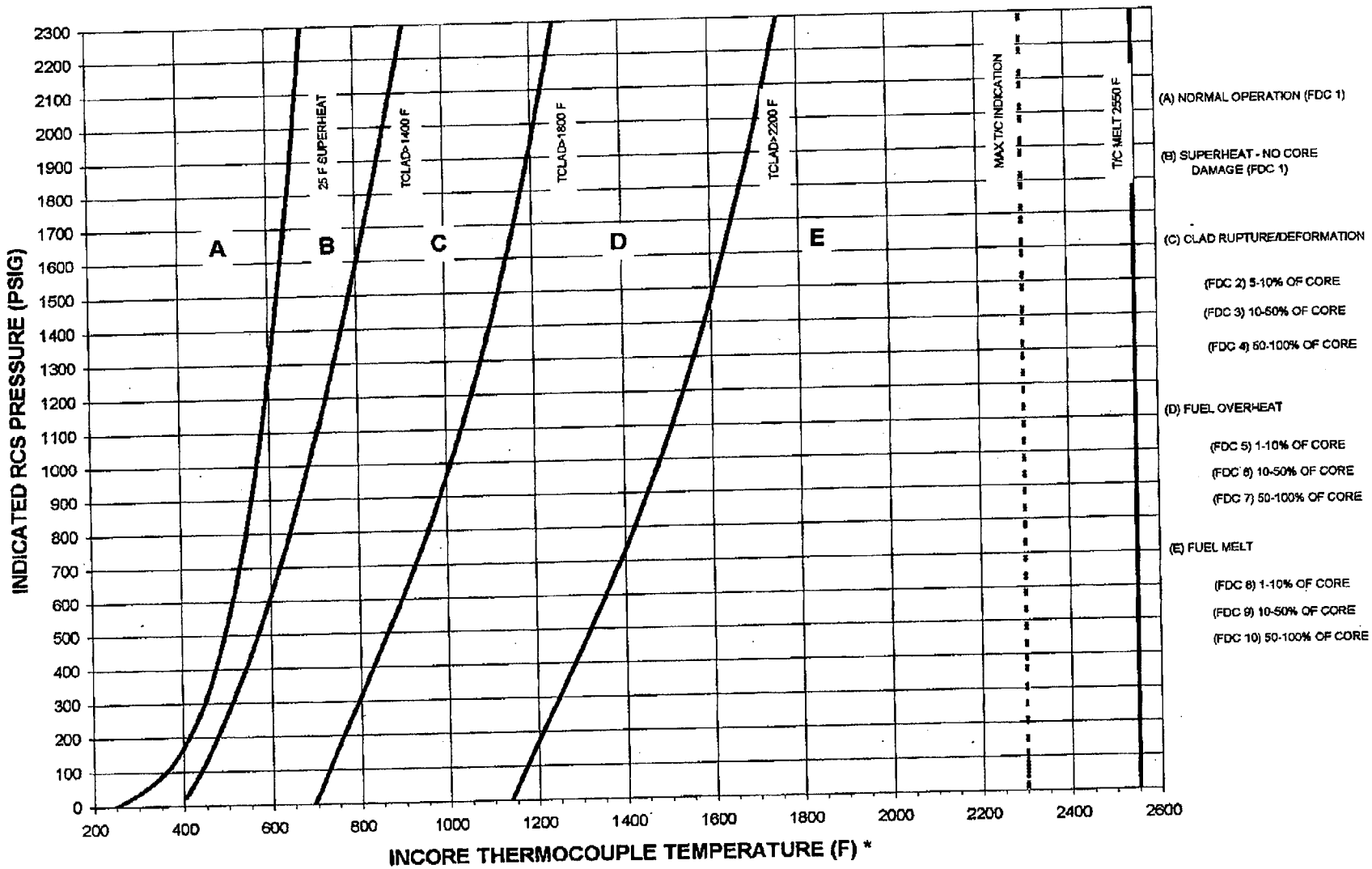
(3) From FSAR Table 14.2-5, Decayed to 72 hours

(4) From FSAR Table 14.2-2

(5) From FSAR Table 14.2-25

FIGURE 5.3-1

CORE DAMAGE REGIONS



* A T/C READING MUST BE IN THE LAST 2/3 OF A GIVEN REGION, OR, IF IN 1ST 1/3, IT MUST REMAIN IN THE REGION FOR AT LEAST 5-10 MINUTES

5.4 Radiation Monitoring System (RMS) Source Term Calculation - This portion of the model allows the user to determine an effluent source term from readings on the TMI-1 Radiation Monitoring System.

5.4.1 Source Term Calculations Using Effluent Monitors

5.4.1.1 To calculate a source term from a RMS reading the following parameters are used:

1. RMS READING: CPM, mR/HR, OR CPM/MIN

2. RMS CHANNEL EFFICIENCY RELATING TO THE CALIBRATION NUCLIDE: CPM/μCI/CC

3. THE METER RESPONSE FACTOR

4. THE FUEL DAMAGE CLASS MIXTURE

5. THE RELEASE FLOW RATE

5.4.1.2 The RMS reading, the particular radiation monitor's efficiency, the monitor response factor, the nuclide fraction from the isotopic percentage section of the program relating to fuel damage class determination, radionuclide mix percentages, and the associated flow rate to the environment are used to calculate a source term. Source terms are identified for the noble gas source term and for the radioiodines.

5.4.1.3 The calculations are performed in the following fashion:

1. First the total monitor response factor is calculated by multiplying the individual nuclide percentages from the fuel damage class determination by the individual nuclide monitor response factors.

$$M = \sum_{1}^{15} P * In$$

Where: M = total monitor response factor

P = individual nuclide percentages from fuel damage class

I_n = individual nuclide monitor response factors

The I_n's for the various RMS detectors are listed as follows:

Individual Mixture Response Factors (I_n)

Nuclide	Scintillation Detectors RM-G26 & RM-G27 (1)	Ion Chamber RM-G24(2)	GM Tubes RM-A5Hi, RM-A8Hi (3), RM-A9Hi, RM-G25	Beta Scint. Detectors RM-A2Lo, RM-A4Lo, RM-A6Lo, RM-A5Lo, RM-A8Lo, RM-A9Lo (4) RM-A15Lo RM-A14Lo
Kr-85m	70.7	212.2	2.35	1.92
Kr-85	1.0	1	0.011	1.98
Kr-87	356	324.39	3.59	9.12
Kr-88	1160	334.15	3.70	2.78
Xe-131m	9.01	4.88	0.054	0.0
Xe-133m	18.6	35.15	0.378	0.0
Xe-133	0.0 (<80keV)	90.24	1	1.0
Xe-135m	193	195.12	2.16	0.0
Xe-135	111	221.95	2.54	2.59
Xe-138	1560	939.02	10.41	4.62
I-131	172	240	2.66	*
I-132	1030	747.8	8.286	*
I-133	274	219.51	2.432	*
I-134	1180	542.44	6.011	*
I-135m	706	341.46	3.784	*

(1) $\frac{\Sigma \text{ MeV*Dis: Nuclide}}{\Sigma \text{ MeV*Dis cal Nuclide}}$ (calibration isotope is Kr-85, threshold set to exclude Xe-133 at 80 keV.)

(2) $\frac{\Sigma\% \text{ Probability Nuclide}}{\Sigma\% \text{ Probability Cal. Nuclide}}$ (calibration nuclide is Kr-85)

(3) $\frac{\Sigma\% \text{ Probability Nuclide}}{\Sigma\% \text{ Probability Cal. Nuclide}}$ (calibration isotope is Xe-133; values for RM-A5Hi, except Xe-133, will be multiplied by 4)

(4) $\frac{\Sigma \text{ Beta decay probability * Beta end-pt. energy nuclide}}{\Sigma \text{ Beta decay probability * Beta end-pt. energy cal. nuclide}}$ (cal. isotope is Xe-133)

* Radioiodines filtered out prior to noble gas channel

2. The noble gas source term in μCi/sec is now calculated using the following equation and input data:

$$Ngst = \left(\frac{1}{M} * ACT * \frac{1}{Me} * Ng/100\right) * (Flow) * 472$$

Where: Ngst = Noble Gas source term in μCi/sec.

M = Total monitor response factor, unitless.

Act = cpm, mR/hr, cpm/min reading from the monitor.

Me = Monitor sensitivity in cpm/μCi/cc, mR/hr/μCi/cc, or cpm/min/μCi/cc.

Flow = Flow rate in CFM

472 = cc/sec/CFM.

Ng = Sum of Noble gas percentages from the selected Fuel Damage classification.

3. The radioiodine source term in μi/sec is then calculated using the noble gas source term and the noble gas to iodine ratio, as discussed in Section 5.8.

$$Rist = Ngst * \frac{Ri}{Ng} * RDF$$

Where: Rist = Radioiodine source term in μCi/sec.

$\frac{Ri}{Ng}$ = The radioiodine to noble gas ratio.

RDF = Iodine reduction factor for the release pathway as discussed in Section 5.11.

4. The noble gas and the radioiodine source terms are then multiplied by the individual isotopic percentages of the fuel damage class mixture to determine the μCi/sec of each of the 15 nuclides, 10 noble gas and 5 radioiodines.

5.4.2 Source Term Calculations Using RM-G-22 and RM-G-23

5.4.2.1 During a Loss of Coolant Accident in the Reactor Building, the high range radiation monitors in the Reactor Building, RM-G-22 and RM-G-23, may be used to estimate the airborne activity in containment. Use of these monitors to estimate the activity released to the containment atmosphere is based on the following assumptions:

1. The response of RM-G-22 and 23 to airborne activity in containment during a LOCA is assumed to be conservatively approximated as a semi-infinite cloud surrounding the detector.

2. It is assumed that the detectors are responding to noble gases only. While some response from airborne iodines would be expected, it is difficult to predict the fraction of iodines released from the RCS that would remain airborne for all LOCA situations. As a result, it is conservatively assumed that all response of the monitors is from noble gases.

3. The isotopic mix of the noble gas activity in containment is assumed to be the same as the distribution in the RCS for a Fuel Damage Class 2. Since the isotopic distribution in the RCS does not dramatically change between Fuel Damage Classes 1-4, this assumption covers a wide range of the most probable core damage conditions. In addition, this mix produces higher μCi/cc/R/hr values than mixes above Damage Class 4, so it is conservative for those conditions.

5.4.2.2 The noble gas source term in μCi/sec is now calculated using the following equation and input data:

$$Ngst = (ACT)(1/65.6)(Flow)(472)$$

Where:

Ngst = Noble Gas source term in μCi/sec

ACT = R/hr reading from the monitor

65.6 = Monitor sensitivity in R/hr/μCi/cc per Reference 7.43

Flow = Flow rate in CFM

472 = cc/sec/CFM

5.4.2.3 The radioiodine source term in μCi/sec is then calculated using the noble gas source term and the noble gas to iodine ratio, as discussed in Section 5.8.

$$Rist = (Ngst)(Ri/Ng)(RDF)$$

Where:

Rist = Radioiodine source term in μCi/sec.
Ngst = Noble gas source term in μCi/sec
Ri/Ng = The radioiodine to noble gas ratio

RDF = Iodine Reduction Factor for the pathway as discussed in Section 5.11.

5.4.2.4 The noble gas and the radioiodine source terms are then multiplied by the individual isotopic percentages of the Fuel damage class mixture to determine the μCi/sec of each of the 15 nuclides, 10 noble gas and 5 radioiodines.

5.5 Post Accident Samples Source Term Calculation - Actual plant effluent sample results may be used to develop the release source terms. This is in fact the preferable method for estimating release quantities if the sample results are available since the method eliminates some of the built in conservatisms of using monitor readings, or contingency calculations. Using sample results also eliminates errors in the source term when the actual release mixture is different from the assumed mix. The use of post accident samples in the RAC model is dependent on the type of post accident sample.

5.5.1 One criteria is the sample station/method to be used. For the Reactor Building three options are presented: 1) CATPASS (Containment Atmosphere Post Accident Sample System), 2) Marinelli/prefilter (marinelli with a particulate and iodine filter upstream), or 3) MAP-5, Radioiodine Processor Station. For the condenser off-gas or the Aux/FHB release pathways, only the Marinelli/prefilter and MAP-5 samples are available.

5.5.2 For MAP-5 sample results each of the identified radioiodine species in the silver zeolite or charcoal cartridge sample from the MAP-5 Processor Station are used for source term generation. The mixture can be decayed from time of shutdown and from time of the sample. NORMALLY THE DECAY CORRECTION WOULD NOT BE APPLIED FROM TIME OF SHUTDOWN SINCE THE ANALYSIS ITSELF ACCOUNTS FOR THAT. Since the MAP-5 only provides information on the radioiodines, the expected ratio between the iodines and noble gases is used to approximate the noble gas activity. The MAP-5 measures radioiodines after they have been acted on by the iodine reduction factors for the release pathway under consideration. These reduction factors are described in Section 5.11. As a result, the noble gas activity will be increased by a factor that is the inverse of the reduction factors described in Section 5.11 to account for reduction of the iodines that is not applicable to the noble gases. Based on the release rate in CFM the isotopic concentrations in μCi/cc are converted to a release rate in μCi/sec for the final source term.

5.5.3 For CATPASS sample results the 10 noble gas and the five radioiodine nuclides identified from air sampling are used for source term generation. The mix can be decayed from the sample to dose projection time. Since the CATPASS only applies to the containment, whether the containment is isolated or not and if the release is proposed or in progress must be considered. Calculations are made using the input activities to develop new isotopic percentages and the activities are changed from μCi/cc to μCi/sec based on the release rate defined to arrive at the final source term.

5.5.4 The marinelli sample results are used if a marinelli/prefilter sample has been collected. This option is available for all three release pathways. Since the production of the source term is based on the measured isotopics, as did the CATPASS program, the marinelli program proceeds in an identical manner.

5.6 Contingency Calculations Source Term Generation - The contingency calculations determine a source term based upon a prioritized set of plant conditions. In this way, credible conservative assumptions, as defined in the FSAR default parameters, are replaced with real-time accident conditions as indicated by plant instrumentation. This will make the calculated source terms more realistic.

5.6.1 In the contingency calculations the previously determined release pathway is utilized to select:

1. Secondary Side Release
2. Reactor Building Release
3. Station Ventilation Releases

The "Secondary Side Release" includes accidents that result in release via: The condenser off-gas, the atmospheric dump valves, the main steam reliefs, and a main steam line rupture.

The "Reactor Building Release" includes accidents that result in a release from the Reactor Building via: the purge duct, when the purge valves are open, or design basis leakage, when the purge valves are closed.

The "Station Ventilation Releases" include accidents that result in a release from the Auxiliary Building, Fuel Handling Building, or ESF Fuel Handling Building.

5.6.2 A "Secondary Side Release" contingency calculation is calculated by identifying four parameters:

1. RCS Activity [D1] μCi/cc
2. Primary to Secondary Leakage [D2] gpm
3. Iodine Reduction Factor (RDF) [D3]
4. Two Phase Release [Tfcf]

5.6.2.1 The "RCS Activity" is determined utilizing:

1. RM-L1 High [A1] cpm, D1 = A1/22.2 μCi/cc
2. RM-L1 Lo [A1] cpm, D1 = A1/1330 μCi/cc
3. Most recent RCS sample, in μCi/cc, or
4. Default to a RCS concentration dependent on the Fuel Damage Class:

Fuel Damage Class	RCS Default μCi/cc
1	*
1A	*
2	4.99E+03
3	2.49E+04
4	4.98E+04
5	4.71E+05
6	2.16E+06
7	4.26E+06
8	4.26E+06
9	4.26E+06
10	4.26E+06

* Damage classes 1 and 1A are variable, based on RCS activity entered from sample data and radioiodine spiking factors used.

5.6.2.2 The "Primary to Secondary Leakage" is determined utilizing:

1. RCS identified leakage [D2] gpm
2. Default to 400 gpm for a double-ended tube shear [D2]

5.6.2.3 The "RDF" [D3] is a function of the release pathway. They are discussed in Section 5.11. The resultant source terms in μCi/sec are calculated by:

Ngst = D1 * D2 * Ng/100 * 63.09

Rist = D1 * D2 * D3 * RI/100 * 63.09 * 1/Tfcf.

Tfcf is described in Section 5.10.

5.6.3 A "Reactor Building Release" contingency calculation is calculated by one of two methods. If the accident type is a LOCA then four parameters are identified:

1. RCS Activity [A2] μCi/cc

2. RCS Leakage to RB [A3] gallons

3. RDF, E4 as discussed in Section 5.11.

4. Release Flow Rate CFM; E3 = flow * 472 to convert CFM to cc/sec

5.6.3.1 The RCS Activity is determined utilizing:

1. RM-L1 High Channel [A1] cpm; [A2] = [A1]/22.2 μCi/cc
2. RM-L1 Low Channel [A1] cpm; [A2] = [A1]/1270 μCi/cc
3. Representative RCS sample results [A2] in μCi/cc
4. Default to a RCS concentration dependent on the Fuel Damage Class:

Fuel Damage Class	RCS Default μCi/cc
1	*
1A	*
2	4.99E+03
3	2.49E+04
4	4.98E+04
5	4.71E+05
6	2.16E+06
7	4.26E+06
8	4.26E+06
9	4.26E+06
10	4.26E+06

* Damage classes 1 and 1A are variable, based on RCS activity entered from sample data and radioiodine spiking factors used.

5. Default mix according to core condition as previously identified. The noble gas and radioiodine released to the Reactor Building is calculated as follows:

   E1 = A2 * A3 * Ng/100 * 3785/5.6E10 μCi

   E2 = A2 * A3 * Ri/100 * 3785/5.6E10 μCi

6. If the accident type is a Fuel Handling Accident in the Reactor Building, then the number of damaged fuel rods is identified by the "user" or an FSAR default condition is used.

   E1 = 1.7 * Num rod/208

   E2 = 0.05 * Num rod/208

5.6.3.2 RCS LEAKAGE - The RCS leakage to the Reactor Building is the "total gallons of RCS leakage into the RB".

5.6.3.3 RDF - This fraction is used to reduce the radioiodines available for release in the Reactor Building, as discussed in Section 5.11.

5.6.3.4 RELEASE FLOW RATE - The release flow rate is determined via flow rate recorder FR-148 if the purge valves are open. If the purge valves are closed the release flow rate is determined via the design basis RB leakrate adjusted for actual RB internal pressure as indicated on PT-291.

5.6.4 A "Station Ventilation Release" contingency calculation is calculated by utilizing the FSAR fuel assembly mix or WGDT mix.

1. Waste Gas Release
2. Fuel Handling Accident in the Fuel Handling Building
3. ESF Fuel Handling Release

5.6.4.1 The extent of the accident is modeled by:

1. Determining the number of damaged fuel rods for a fuel handling accident in the Fuel Handling Building, or for the ESF FHB.

   Ngst = 4.2E6 * Num rods/56 μCi/Second

   Rist = 750 * Num rods/56 μCi/Second

2. Using the worst case FSAR source term for WGDT's of 10,000 curies of noble gas and 5 curies of radioiodine, or using the typical WGDT FSAR mix of 1000 curies of noble gas and 5 curies of radioiodine.

3. Determining the curies released for a Waste Gas Accident. Typical source term based on a typical inventory of:

   Ngst = 1.0E9/Dr μCi/second

   Rist = 1.0E5/Dr μCi/second

   or

   FSAR worst case:

   Ngst = 1.0E10/Dr μCi/second

   Rist = 5E6/Dr μCi/second

   Where: Dr = duration of release.

5.7 Decay Scheme Calculation - The model provides the capability to (1) decay the postulated mixture from the time of reactor shutdown to time of the dose projection or (2) decay sample data from the time the sample is obtained to the time of dose projection. The calculation only decays forward in time. This calculation adjusts the individual nuclide percentages according to the conventional exponential decay equation:

$$A = A_o \exp(-\lambda t)$$

5.7.1 Fifteen isotopes are decayed according to the equation

N(w) = I(w) * EXP (-decay time * f [w])

where:

I(w) = postulated isotopic percentage

decay time = user input time

f(w) = isotopic decay constants

5.7.1.1 The adjusted isotopic percentages N(w) are corrected for Xenon buildup due to iodine decay. For Xe-131m the equations are:

S1 = I(11) - N(11)

N(5) = 0.88 * S1 + N(5)

where:

S1 = amount of I-131 decayed

0.88 * S1 = amount of Xe-131M buildup

5.7.1.2 For Xe-133M the equations are:

S1 = I(13) - N(13)

N(6) = 0.02 * S1 + N(6)

where:

S1 = amount of I-133 decayed

0.02 * S1 = amount of Xe-133m buildup

N(7) = 0.98 * S1 + N(7) calculates the amount of Xe-133 buildup from I-133

5.7.1.3 For Xe-135m and Xe-135 the equations are:

$$S1 = I(15) - N(15)$$

$$N(8) = 0.3 * S1 + N(8)$$

$$N(9) = 0.7 * S1 + N(9)$$

where:

S1 = amount of I-135 decayed

0.3 * S1 = amount of Xe-135m buildup

0.7 * S1 = amount of Xe-135 buildup

5.7.1.4 The isotopic percentages are recalculated as:

$$I(w) = N(w)/Sum\ (N) * 100$$

where:

N(w) = adjusted/corrected postulated isotopic percentages

Sum (N) = sum of the fifteen isotopic percentages

I(w) = final isotopic percentages based upon 100.

5.8 Noble Gas to Iodine Ratio Calculations

5.8.1 Whether performing dose projections based upon RMS readings, post accident samples or contingency calculations, it may be necessary to compute the NOBLE GAS TO IODINE RATIO. The uses of this ratio are discussed below.

An airborne release from a nuclear power plant will primarily consist of noble gases and radioiodines. Except in the most severe and improbable accident scenarios, radioactive particulates are not expected to be important dose contributors. The RAC model was designed to incorporate ten noble gases and five radioiodines.

The 15 isotopes are considered to be the most radiologically significant gaseous isotopes available for release from an operating nuclear power plant. Pertinent radioactive decay parameters such as half life, average gamma energy per disintegration and average beta energy per disintegration for each isotope, and individual isotope source term information are used to determine dose rate conversion factors and dose rates that are specific to the isotopic mixture being released. These calculated quantities can be adjusted to account for radioactive decay during the accident sequence.

5.8.2 The TMI RAC model always projects both thyroid (CDE) and whole body (TEDE) dose rates at specified downwind distances. Consequently an estimate of the isotopic release rate is necessary for both iodines and noble gases. Under normal circumstances the model starts with a core inventory of all fifteen isotopes and traces the progress of each one through various systems or processes until it is released. Depending upon the type and severity of the accident and the engineered safety systems that have been activated, the isotopic ratios can vary widely. There are some circumstances where the release rates of specific isotopes may be zero or negligibly small. But, in general, the model accounts for the fifteen isotopes listed above.

In certain circumstances it is not possible to obtain release rates for all fifteen isotopes individually. For example, some plant effluent monitors have only noble gas channels while others have particulate, iodine and gas channels. The MAP-5 sampling system yields only iodine information, where the CATPASS and the Marinelli gas sampling systems yield information on all fifteen isotopes. For release pathways where information on both noble gases and iodines is not available, the RAC model uses the noble gas to iodine ratio to fill in the missing information. The following example illustrates the use of this ratio:

A certain type of reactor accident has occurred. Based on an assessment of the degree of core damage and the accident type, the model uses a default mixture of 15 nuclides and calculates the fraction of the mix that each isotope represents. The noble gas to iodine ratio is also calculated. Assume that the ratio was equal to 5/1 in this case. Also assume that an iodine sample was taken which indicated a total radioiodine release rate of 5000 μCi/sec. Using the noble gas to iodine ratio in the absence of specific noble gas measurements, the model would calculate a gross noble gas release rate of 25,000 μCi/sec. It would also calculate individual noble gas release rates by using the isotopic fractions from the default mix.

5.8.3 To summarize, the highest quality information available is a quantitative measurement of each nuclide. This type of information is available from RCS, gas Marinelli and prefilter, and CATPASS samples. So there is no need to invoke the noble gas to iodine ratio in these cases. The second best measurement would be one that yielded gross noble gas and gross iodine readings. This situation occurs in the low range radiation monitors which have individual noble gas and iodine channels. Based upon the default mixture fractions, the release is apportioned among the fifteen nuclides to arrive at isotopic release rates. Again, there is no need to use the noble gas to iodine ratio. It is used only in circumstances where either noble gas or iodine measurements are not available, for example, when only noble gas or only iodine information is available.

5.8.4 There are some refinements and subtleties that the model user should be aware of. The noble gas to iodine ratio changes with time because of radioactive decay. The RAC model has the ability to account for radioactive decay and to compute a decay corrected noble gas to iodine ratio. As explained elsewhere in this manual, the model also corrects the Dose Rate Conversion Factor (DRCF) for decay of the isotopes in the mix. When performing dose projections several hours or more after the reactor has tripped, these two decay corrections can significantly alter the resultant projections. The model provides the capability to account for decay between reactor trip and dose projection.

5.8.5 For dose projections based upon RMS readings, the decay correction to the noble gas to iodine ratio is straightforward. A default mixture of the fifteen noble gases and iodines is selected based upon an assessment of core damage. If the mix, is to be decayed, all fifteen isotopes are decayed by the standard exponential decay law using the decay time between reactor trip and dose projection. The noble gases and iodines are totaled separately so that their ratio at dose projection time can be calculated. (Ingrowth of xenon isotopes from decay of iodine is accounted for.) The decay adjusted ratio is then used to fill in the missing noble gas or iodine information, as explained above.

5.8.6 When iodine samples are taken at the MAP-5 stations a two step decay process is used. As above, a default mixture is chosen, based upon the fuel damage classification. For a dose calculation based upon a radioiodine processor sample, if decay correction is desired, the model uses two decay time intervals:

1. The time between sampling and dose projection
2. The time between reactor trip and dose projection

Sample results from the radiochemistry lab are reported as of the sample collection time. When significant time has elapsed between sampling and dose projection, the results should be decayed from sampling time to dose projection time. In order to compute the noble gas portion of the source term, the default mix is first decayed from reactor trip time to dose projection time. The noble gas to iodine ratio is computed for the decayed default mix, and any iodine reduction factors present in the specific release pathway. This ratio, along with the gross radioiodine sample result, is used to compute a gross noble gas source term. Isotopic source terms are calculated from the decayed mixture noble gas fractions. Note that the final source term is a combination of noble gases from a default mix and radioiodines from a sample. Each has been decayed to the dose projection time.

A word of caution should be added at this point. The iodine released in certain types of accidents may be reduced by various chemical and physical processes such as iodine plateout or formation of water soluble iodide salts. The noble gas to iodine ratio, as calculated above, may not completely account for this iodine reduction. As a consequence, the ratio, based upon the default mix, may be too low. This creates the potential for underestimating the noble gas portion of the source term. RAC personnel should be aware of this possibility. A comparison of field team data and the source term dose projections would reveal agreement for thyroid (CDE) doses, but not for whole body (TEDE) doses.

5.9 Effluent Release Flow Rates - Flow rates for effluent releases to the environment are divided into four categories:

1. Normal ventilation flow rates.

2. Reactor Building leakage flow rates.

3. Adjacent momentum plume rise (station vent and reactor purge concurrently releasing).

4. Flow rates for OTSG tube rupture release directly to atmosphere.

   - Buoyant plume rise
   - Source term calculation using RMG-26 or RMG-27
   - Source term calculation using a contingency calculation

These flow rates for accident source terms to the environment are calculated as follows:

5.9.1 TMI-1 Normal Ventilation Flow Rates

The RAC Model provides the option to use the actual ventilation flow rates as read from the flow recorders or to use default flow rate(s). Each normal plant flow path has predetermined flow rate ranges, and assigned flow recorders as follows:

1. Reactor Building Purge
   - FR909 0-20,000 CFM; Low Range
   - FR148B 0-50,000 CFM; High Range
2. Reactor Building Purge and Make-up Exhaust
   - FR148A 0-50,000 CFM
3. Reactor Building Hydrogen Purge System
   - FI282 5-50 CFM
   - FI283 20-200 CFM
   - FI284 100-1000 CFM
   - Total 5-1250 CFM
4. Kidney Filter System
   - AHE-101, AH-F-12 P/I filters 20,200 SCFM
5. Auxiliary Building Exhaust
   - FR150 0-100,000 CFM
6. Fuel Handling Building Exhaust
   - FR149 0-50,000 CFM

7. Auxiliary and Fuel Handling Building Exhausts

   - FR-151 0-150,000 CFM

8. Condenser Off-Gas Exhaust

   - RMR15 Recorder FT-1113 Ch. A0-200 CFM

9. ESF Fuel Handling Building Exhaust

   - No Flow Recorder at this time 0-8000 CFM

Default values are used in the RAC Model when a small value or an unknown value is required as input to a dose projection. The default values are:

5000 CFM - Reactor Building Purge

- Reactor Building Purge and Make-up Exhaust
- Auxiliary Building Exhaust
- Fuel Handling Exhaust
- Auxiliary and Fuel Handling Building Exhausts.

40 CFM - Condenser Off-Gas

7000 CFM - ESF FHB Exhaust

These default values allow the user to continue with dose projections even though a value is small or unknown. Therefore, once the dose projection is complete, the results may be ratioed up or down depending on the situation. For example, if the default value of 5000 CFM was used for a Reactor Building Purge and subsequently an engineering calculation was performed indicating 1000 CFM flow. The dose projection could be ratioed down by one fifth (1/5). Therefore, a dose projection of 10 mrem would then be approximately 2 mrem based on the reduced flow calculation, realizing that the X/Q will also be affected by reducing flow.

5.9.2 Reactor Building Leakage Flow Rate

Another section of the model calculates a leakage flow rate out of the Reactor Building based on Reactor Building pressure. The Reactor Building pressure indicator is PT-291, 0-100 psig, located on control room panel CR. The leakage out of the Reactor Building is based on the amount of pressure in the Reactor Building with all penetrations closed. The following equation is used to calculate the Reactor Building Leak Rate:

$$L_T = L_A * SQRT(P_T/P_A)$$

Where: L_T = Reactor Building Leak Rate in CFM

L_A = Maximum allowable integrated leakage rate at P_A L_A = 6.14 CFM (at atmospheric pressure)

P_A = Peak Reactor Building internal pressure at design basis accident,

P_A = 50.6 psig

P_T = Actual Reactor Building internal pressure in psig

Therefore, the maximum leakage allowed at a design basis accident pressure of 50.6 psig is 6.14 CFM. Leak rates at 0-60 psig can be calculated from the above formula. The default value in the model is 50.6 psig. A graphic representation follows in Figure 5.9-1. This calculated flow rate is used in conjunction with CATPASS results which are corrected to standard temperature and pressure.

5.9.3 Adjacent Momentum Plume Rise (Station Vent and Reactor Purge)

For an isolated stack, either the station vent or the Reactor Purge, the stack gas exit velocity can be calculated from the flow rate according to the following formula:

$$w = \frac{V}{\pi r_2} \qquad (1)$$

Where w = stack gas exit velocity
V = flow rate or volume flux
r = radius of stack

The Station Vent and the Reactor Purge stack are situated close enough together that their plumes will mix as the plumes rise. For two or more adjacent stacks that have different exit velocities, the effect of mixing on the exit velocities of non-buoyant plumes can be given by the following formula:

$$\overline{w} = \frac{\Sigma wV}{\Sigma V} \qquad (2)$$

Where $\overline{w}$ = exit velocity due to mixing

Since airborne activity concentrations calculated by RMS and leakrate calculations are not corrected to atmospheric pressure 1.39 cfm is substituted for the 6.14 cfm in the above equation.

If these adjacent stacks were modeled as a single stack, the radius of the stack would be given by:

$$\bar{r} = \sqrt{\frac{\sum v}{\pi \ \bar{w}}} \qquad (3)$$

At TMI-1, the reactor building stack and station vent are adjacent stacks. For computing plume rise, the stack gas exit velocity and stack radius were calculated according to Eq. (2) and (3) above. A comparison of the adjacent plume rise with the plume rise from the individual stacks is shown in Table 5.9-1.

5.9.4 Flow Rate Calculations for OTSG Tube Rupture Release Directly to the Atmosphere (see Figure 5.9-2 and Figure 5.9-3).

TMI-1 has 22 main steam relief and 2 atmospheric dump valves. Data on the valves are presented in Table 2, which lists the valve identification number, function, manufacturer, pressure set point and flow rate. The set point pressures vary from 200 psig to 1092.5 psig, and the steam flow rate from 70,211 lbs/hr to 824,269 lbs/hr. Note that valves 4A&B are manually operated and do not have a set point pressure. These valves, MS-V-4A/B, can be operated from 0 to 100% open. The valve position openings along with the secondary system pressure relate to a release flow and plume height. The percent open for these two (2) valves can be read at the center control panel under the turbine bypass dump controller for MS-V-4A/B from 0 - 100%.

Each of the 22 valves at TMI-1 has a stack or vent where the steam is ejected into the atmosphere. The location of these stacks is shown in Figure 5.9-2. If a steam generator tube ruptures, each of the 22 valves and stacks acts as a throttle to limit the flow from the steam line to the atmosphere. When a valve opens, the flow through it will be approximately equal to the rated flow, and the flow can be assumed to be approximately constant until the pressure in the steam line drops to the point where the valve reseats. For a stuck open valve, the pressure decreases rapidly with time, and the flow through the valve is only a small fraction of the rated flow. For either a normally operating valve or stuck open valve, if the pressure and temperature in the steam line are known, the conditions just beyond the stack exit can be estimated by assuming expansion of the steam to atmospheric pressure and temperature.

5.9.4.1 Buoyant Plume Rise

When the steam is released into the atmosphere, the rise of the steam plume is initially controlled by its velocity, temperature and cross-sectional area. Depending on these variables and atmospheric conditions, the plume rise can vary from hundreds to thousands of feet. Plume rise is a very important factor in determining maximum ground level doses. For a PWR, plume rise can increase the effective stack height by a factor of 5 to 50. Since maximum ground level dose is roughly proportional to the inverse square of the effective

stack height, a plume rise of 200 feet, for example, gives a ground level concentration 100 times higher than that from a plume rise of 2000 feet.

Modeling of plume rise begins with modeling the steam condition at the valve inlet. Table 5.9-3 outlines the calculational steps required to compute buoyant plume rise, beginning with the valve inlet. The far left side of the table identifies the area for which the calculation applies: valve inlet, top of stack, jet origin, and plume rise. For each area, several quantities must be computed from various inputs, and these are also identified in the table. Buoyant plume rise was calculated according to Briggs (1984). The details of all the calculations are discussed in the Environmental Controls document Potentially Buoyant Releases at TMI-1.

CAUTION

In highly stable atmospheric conditions, the presence of layers of different temperature air can cause thermal boundaries resistant to plume vertical travel. In some conditions, a buoyant plume may penetrate these layers and not come down to the surface as predicted. In other cases the plume may be unable to penetrate the layer and the effective stack height will be reduced to the height of the layer. This may cause ground concentrations to be higher and closer to the plant than predicted. In these conditions (i.e., highly stable meteorology with a buoyant plume) off-site monitoring will provide an indication of the magnitude of the effect. It may also be possible to estimate the effect through visual observation of the plume.

5.9.4.2 Source Term Calculation Using RM-G-26 or RM-G-27 (see Figure 5.9-3)

RMG-26 and RMG-27 are effective in calculating a primary to secondary release source term direct to the atmosphere when:

1. Atmospheric Dump Values (ADV) MS-V-4A or MS-V-4B are open from 0-100%, as indicated on Control Room Panel "CC", and releasing radioactive steam to the environment, and/or,

2. Emergency Feed Pump (EFP) relief valves, MS-V-22A or MS-V-22B are open and releasing radioactive steam to the environment, and/or,

3. EFP is in operation and releasing radioactive steam from the EFP exhaust to the environment.

4. Steam bypass dump to the condenser through MS-V-8A/B.

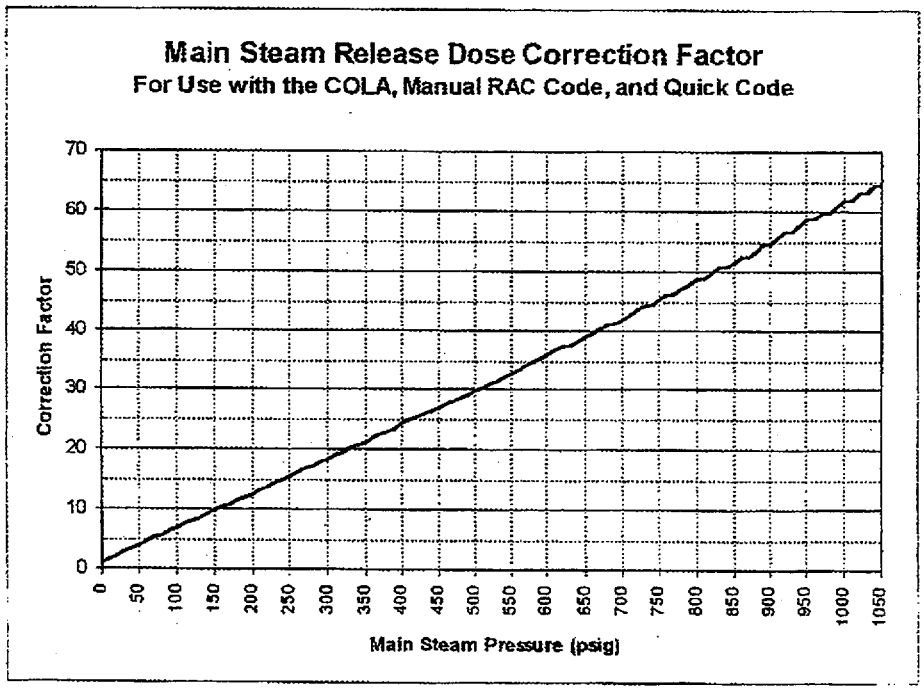
When the model calculates a release from an OTSG tube rupture directly to the atmosphere and uses RM-G-26 or RM-G-27 readings, the calculation Steam Flow Computation is used to determine a release flow rate, depending on which of the valves are open (Table 5.9-2 and Reference 7.44). The mass flow rate from each open valve is added up to give a total flow rate to the environment. A source term is calculated using the mass flow rates and the

RM-G-26/27 readings converted to mass concentration using the monitor efficiencies to give μCi/second.

A correction factor must be applied to results provided by the following dose assessment codes when RM-G-26 and RM-G-27 are used to quantify direct-to-atmosphere steam release (Atmospheric Dump Valve, Main Steam Relief Valve, or Steam Driven Emergency Feed Pump Discharge) source terms and offsite doses:

- COLA Code
- Manual RAC Code
- Quick Code

These codes do not account for main steam pressure when using these monitors to calculate source terms. As a result, they are overly conservative. The following graph provides a correction factor to compensate for this conservatism (Reference RAF 3640-00-011).



To use the graph:

1. On the x-axis of the graph, find the main steam pressure when the dose projection was performed.

2. Determine the appropriate correction factor from the y-axis of the graph.

3. **Divide** the source terms, dose rates and doses produced by the code by this correction factor to provide a more realistic assessment of the release.

NOTE

This correction factor is not to be used for results provided by RAC Spreadsheet.

NOTE

Calculation of a source term using RMS (RM-G-26/27) is dependent on the Atmospheric Dump Valves (ADV) status. If the ADV is open, the calculation is appropriate. If the ADV is closed but plant conditions (OTSG leakrate and core damage) have not changed significantly, or there is other flow past the monitors as noted above, then the use of a RM-G-26/27 peak reading will be appropriate. If the ADV is closed and plant conditions have changed significantly, and there is no other source of flow downstream of MS-V-2A/B, then the contingency calculation applies.

5.9.4.3 Source Term Calculation Using a Contingency Calculation

When the user performs a Contingency Calculation due to the lack of sample results or RM-G-26/27 readings, the flow rate corresponding to the set point pressure is used, if the valve operates normally. This flow rate is given in Table 5.9-2. If, however, the valve sticks open, and the steam generator pressure is less than the set point pressure, then the flow rate is based on the tables supplied by the valve manufacturer. These tables have been incorporated into the RAC model. The flow from all the valves is totaled and modeled as a release to the atmosphere.

TABLE 5.9-1

Adjacent Plume Rise at TMI Reactor Bldg. Stack and Station Vent Stack										
Stability	Actual Flow Characteristics				RAC Model			MIDAS		
	Reac Bld Stack	Stack Diameter	Station Vent	Stack Diameter	Flow Rate	Stack Diameter	Plume Rise	Flow Rate	Stack Diameter	Plume Rise
	(cfm)	(m)	(cfm)	(m)	(cfm)	(m)	(ft)	(cfm)	(m)	(ft)
Unstable	10,000	1.1	10,000	1.7	20,000	1.847	**30.3**	10,000	1.1	**25.4**
Neutral								10,000	1.7	**16.4**
(Class A)	10,000	1.1	120,000	1.7	130,000	1.827	**199.2**	65,000	1.1	**165.4**
								65,000	1.7	**107.0**
Stable	10,000	1.1	10,000	1.7	20,000	1.847	**24.8**	10,000	1.1	**22.1**
								10,000	1.7	**16.5**
(Class F)	10,000	1.1	120,000	1.7	130,000	1.827	**85.8**	65,000	1.1	**75.9**
								65,000	1.7	**57.0**

TABLE 5.9-2

TMI STEAM GENERATOR RELIEF VALVES						
Valve Number	Function	Valve Manufacturer	Valve			Stack Diameter
			Discharge Area	Set Point		
				Pressure	Flow Rate	
(MS-V)			(sq.in.)	(psig)	(lbs/hr)	(inches)
17A-D	Relief Valves, Bank 1	Dresser/ Consolidated	16	1050	792,617	10.02
18A-D	Relief Valves, Bank 2	Dresser/ Consolidated	16	1060	800,065	10.02
19A-D	Relief Valves, Bank 3	Dresser/ Consolidated	16	1080	814,960	10.02
20A&D	Relief Valves, Bank 1	Dresser/ Consolidated	16	1050	792,617	10.02
20B&C	Relief Valves, Bank 4	Dresser/ Consolidated	16	1092.5	824,269	10.02
21A&B	Relief Valves, Small Safety	Dresser/ Consolidated	3.97	1040	194,820	10.02
22A	Safety Relief, Emergency Feed Pump	Lonergan	6.38	200	70,211	13.13
22B	Safety Valve, Emerg. (F.W.P.T. Steam Inlet)	Lonergan	6.38	220	76,795	13.13
4A&B	Relief Valves (manual) to Atmosphere	Fisher	Variable	1010	402,792	13.13

TABLE 5.9-3

	Calculation Steps For Computing Plume Rise			
	Step	Quantity Computed	Input Values Needed	Source of Input
Valve Inlet	1	Steam flow rate thru valve	pressure of steam	Plant Instrumentation
Top of Stack, Below Chamfer	2	Pressure of steam at top of stack, below chamfer (if choked flow)	a. steam flow rate b. internal radius of stack c. enthalphy	Step 1 constant <constant>
	3	Specific volume of steam at to of stack, below chamfer	pressure at top of stack	Step 2
Below Chamfer	4	Temperature of steam at top of stack, below chamfer	pressure at top of stack	Step 2
Iso Jet Origin	5	Velocity of steam at top of stack, below chamfer	a. specific volume of steam b. flow rate of steam c. internal radius of stack	Step 3 Step 1 constant
Jet Origin	6	Density of steam at jet origin (ambient pressure)	a. temperature of steam b. pressure of ambient air	Step 4 <constant>
	7	Jet radius at origin (Needed for MIDAS only; used in RAC but not really needed)	a. density of steam b. velocity of steam c. flow rate of steam	Step 6 Step 5 Step 1
Plume Rise	8	Plume rise	a. jet radius at origin b. density of steam at origin c. velocity of steam at origin d. density of ambient air e. wind speed at origin f. 150'-33' delta T	Step 7 Step 6 Step 5 <constant> Plant Instrumentation Plant Instrumentation

FIGURE 5.9-1

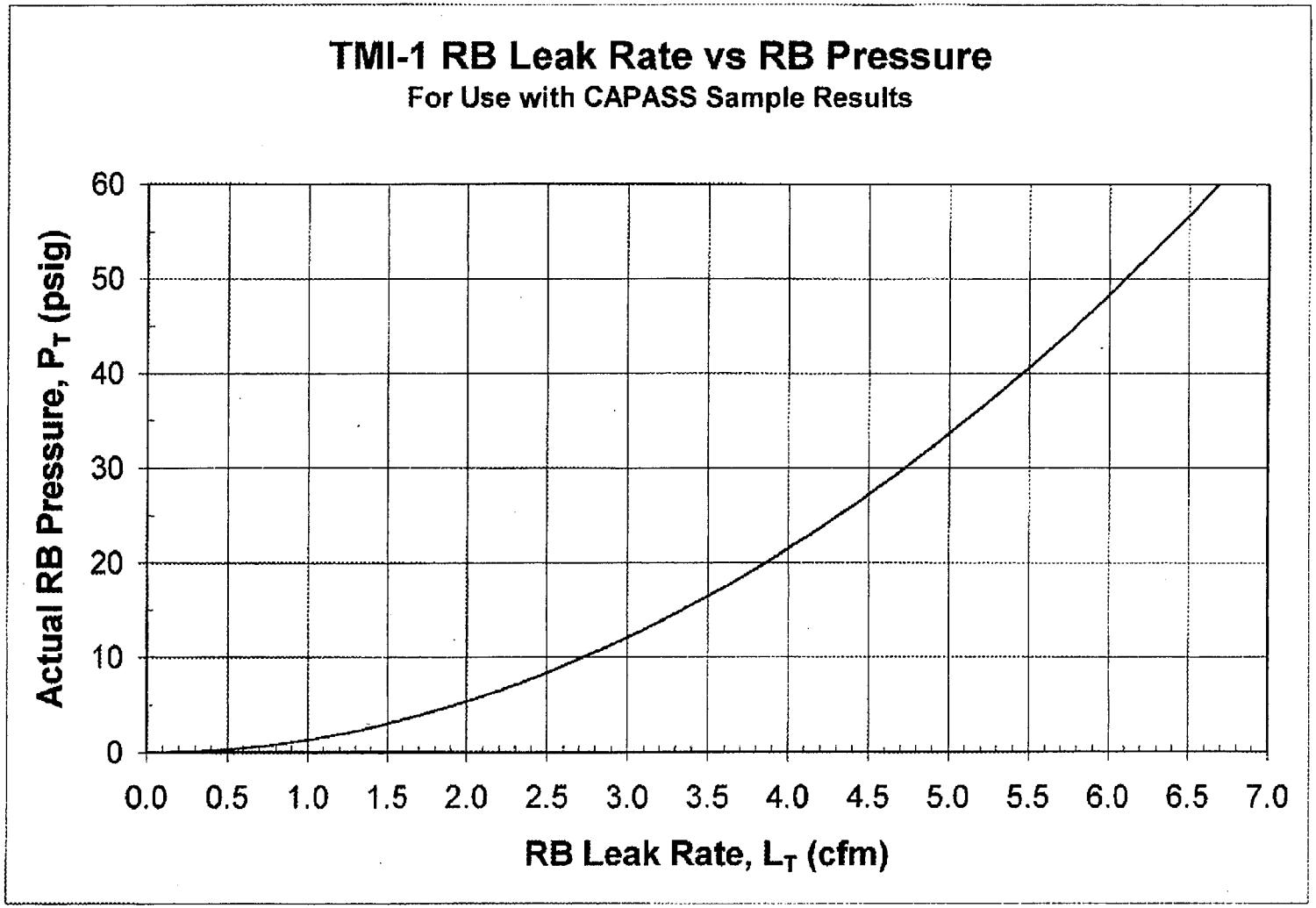
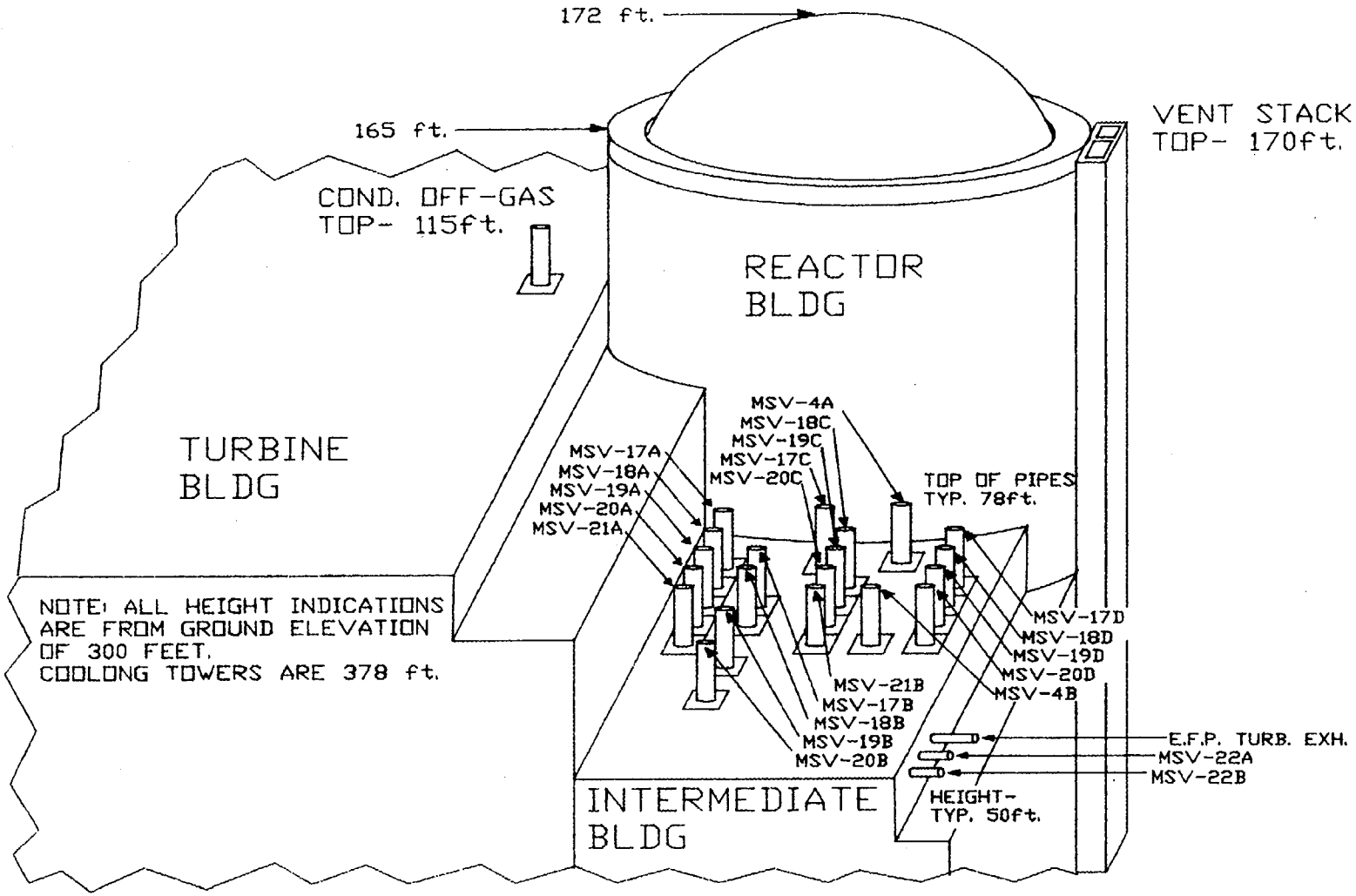

TMI-1 RB Leak Rate vs RB Pressure
For Use with CAPASS Sample Results
Actual RB Pressure, P_T (psig)
0
10
20
30
40
50
60
0.0
0.5
1.0
1.5
2.0
2.5
3.0
3.5
4.0
4.5
5.0
5.5
6.0
6.5
7.0
RB Leak Rate, L_T (cfm)

FIGURE 5.9-2


172 ft.
165 ft.
VENT STACK
TOP- 170ft.
COND. OFF-GAS
TOP- 115ft.
REACTOR
BLDG
TURBINE
BLDG
MSV-4A
MSV-18C
MSV-19C
MSV-17C
MSV-20C
MSV-17A
MSV-18A
MSV-19A
MSV-20A
MSV-21A
TOP OF PIPES
TYP. 78ft.
NOTE: ALL HEIGHT INDICATIONS
ARE FROM GROUND ELEVATION
OF 300 FEET.
COOLONG TOWERS ARE 378 ft.
MSV-17D
MSV-18D
MSV-19D
MSV-20D
MSV-4B
MSV-21B
MSV-17B
MSV-18B
MSV-19B
MSV-20B
E.F.P. TURB. EXH.
MSV-22A
MSV-22B
INTERMEDIATE
BLDG
HEIGHT-
TYP. 50ft.

OTSG LEAK/RUPTURE RELEASE PATHWAYS
(SIMPLIFIED)

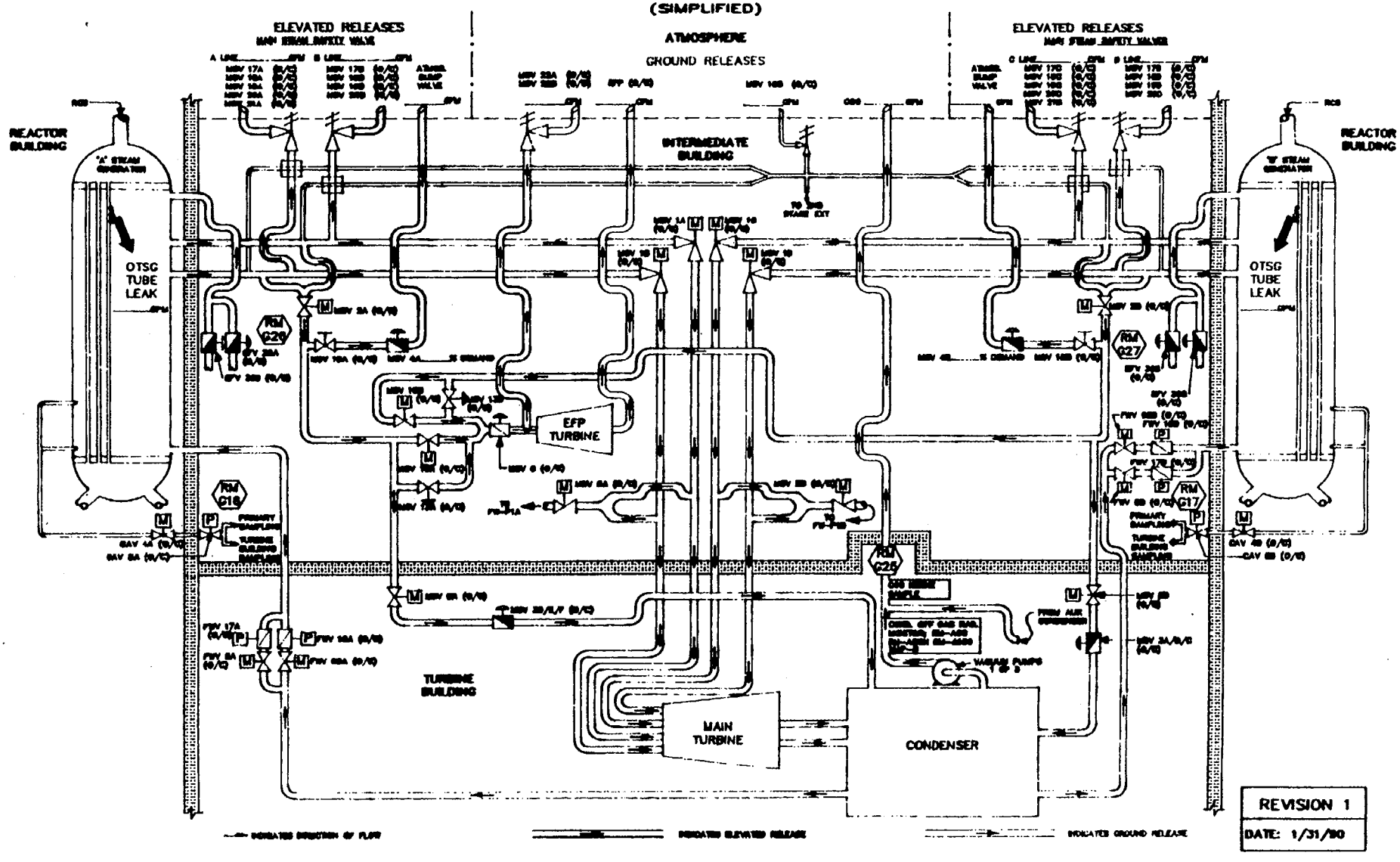
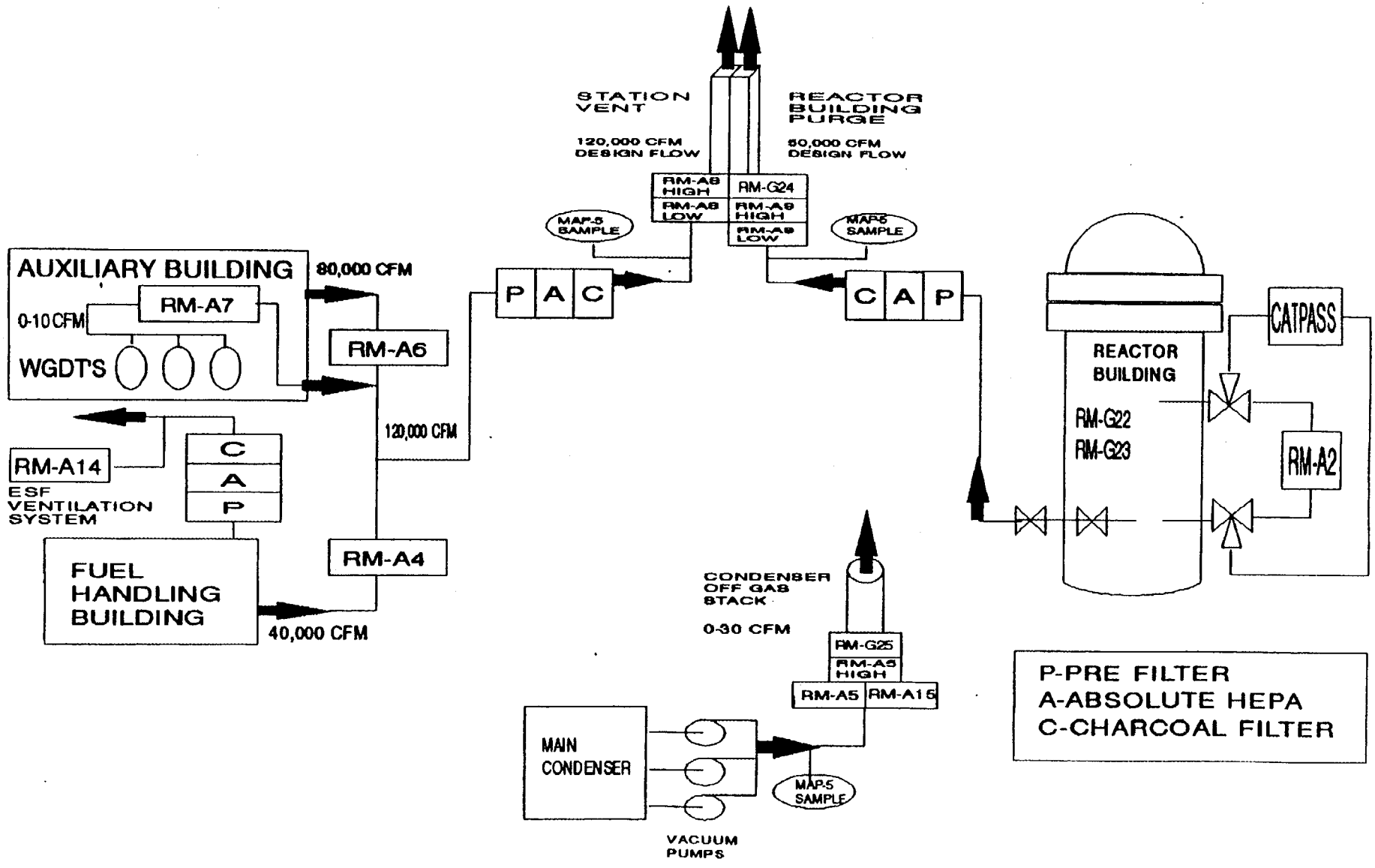


FIGURE 5.9-4


STATION VENT
120,000 CFM DESIGN FLOW
REACTOR BUILDING PURGE
50,000 CFM DESIGN FLOW
RM-A8 HIGH
RM-A8 LOW
RM-G24
RM-A9 HIGH
RM-A9 LOW
MAP-5 SAMPLE
MAP-5 SAMPLE
AUXILIARY BUILDING
80,000 CFM
RM-A7
0-10 CFM
WGDT'S
RM-A6
120,000 CFM
P A C
C A P
CATPASS
REACTOR BUILDING
RM-G22
RM-G23
RM-A2
RM-A14
ESF VENTILATION SYSTEM
C
A
P
FUEL HANDLING BUILDING
RM-A4
40,000 CFM
CONDENSER OFF GAS STACK
0-30 CFM
RM-G25
RM-A5 HIGH
RM-A5
RM-A15
MAIN CONDENSER
MAP-5 SAMPLE
VACUUM PUMPS
P-PRE FILTER
A-ABSOLUTE HEPA
C-CHARCOAL FILTER

5.10 Two-Phase Steam Flow Determination -A two-phase (liquid and gas) release calculation was included for an OTSG tube rupture accident in response to INPO SOER 83-2 (Recommendation #12). INPO SOER 83-2 "Steam Generator Tube Ruptures" was developed based upon the steam generator tube rupture events at R. E. Ginna, Oconee and Rancho Seco. Recommendation #12 states "Emergency Plan Implementation Procedures should . . . ensure that estimates of doses can be made for two-phase or liquid releases through the steam generator safety relief valves." GPUN Corporation is required to respond to all SOER recommendations. The calculational method used to implement this recommendation is based upon the assumptions that the valve inlet fluid condition is either pure liquid or steam (as indicated by the OTSG wide range level instrumentation) and following discharge, the steam fraction is described by assuming that there is no change in total energy content. If the OTSG wide range level instrument is indicating that the valve inlet fluid condition is pure liquid, greater than 600 inches, and the fluid is near saturation for the pressure and temperature, then the fraction of gas vapor present in the release is a function of the OTSG pressure as indicated on the PCL panel, PI950A and 951A, or the console center, SPGA PT1 and 2 or SPGB PT 1 and 2.

5.10.1 The model determines a two-phase correction factor [Tfcf] which is a function of OTSG pressure in psia. This factor is only calculated if the OTSG water level is indicating a liquid release (greater than 600 inches on the wide range level instrument reading). The correction factors are used to account for the radioiodine that would remain in the liquid portion of the resultant two-phase release to the environment.

Using the OTSG pressure in Psia, the model selects a correction factor which is subsequently used with other iodine reduction factors applicable to the release pathway to correct the radioiodine source term. These other iodine reduction factors are described in Section 5.11. The algorithm used by the RAC model to determine Tfcf is as follows:

OTSG pressure ≥ 1200	Tfcf = 2.43
1000 ≤ OTSG pressure < 1200	Tfcf = 2.61
800 ≤ OTSG pressure < 1000	Tfcf = 2.83
600 ≤ OTSG pressure < 800	Tfcf = 3.18
400 ≤ OTSG pressure < 600	Tfcf = 3.77
200 ≤ OTSG pressure < 400	Tfcf = 5.09
OTSG pressure < 200	Tfcf = 7.27

NOTE

Increasing OTSG A/B water level will possibly help cut down the release of radioiodine due to the partitioning effect of the iodine in water. Increasing OTSG level should be discussed with the Emergency Director as a means of reducing off-site doses.

5.11 Iodine Reduction Factors

Radioiodine normally exists in chemical forms which are highly reactive. They readily adsorb onto surfaces and can be scrubbed chemically from the atmosphere. The radioiodine removal methods available are specific to each release pathway and are described in Reference 7.21. The TMI RAC Model provides the capability to include or disregard radioiodine reduction during the movement of these isotopes from the core to the atmosphere. The reduction factors (RDF) used by the TMI RAC Model are those described in Reference 7.21.

5.11.1 Releases from Condenser Offgas During an OTSG Tube Rupture

For a release through the condenser off-gas the, radioiodine RDF is 0.0075. The radioiodine RDF is a product of: The fraction of radioiodine entering the OTSG from the RCS that is a volatile iodine species (.05) and the partition factors for volatile iodine species in the main condenser (.15). Non-volatile iodine species have a partition factor of zero in the condenser off-gas. This is discussed in Reference 7.17.

5.11.2 Releases from Reactor Building Design Basis Leakage

Radioiodines released into the containment building are subject to airborne iodine reduction from either natural processes (e.g., gravitational settling and plateout) or from Reactor Building sprays. The effectiveness of these reduction processes are dependent on the holdup time of the activity in the building prior to the beginning of the release.

For Reactor Building design basis leakage where Reactor Building spray is not on, iodine reduction factors are as follows (Reference 7.21):

$H \geq 24$ hours	RDF = 0.01
2 hours $\leq H < 24$ hours	RDF = 0.04
$H < 2$ hours	RDF = 0.4

For Reactor Building design basis leakage where Reactor Building spray is on, iodine reduction factors are as follows (Reference 7.21):

$H \geq 24$ hours	RDF = 0.002
2 hours $\leq H < 24$ hours	RDF = 0.02
$H < 2$ hours	RDF = 0.03

5.11.3 Direct to Atmosphere Releases During an OTSG Tube Rupture

Iodine reduction factor for direct to atmosphere steam releases during leakage of reactor coolant into the secondary side is 0.5. The RDF is the result of partioning in the steam generator per Reference 7.21.

As described in Section 5.10, if OTSG level is > 600 inches, a two phase correction factor also applies. In these conditions the RDF becomes:

$$RDF=(0.5)(1/Tfcf)$$

Where

0.5 = Steam generator partition factor for B&W type OTSG (Reference 7.21)

Tfcf = Two phase steam flow correction factor described in Section 5.10.

5.11.4 Releases from Auxiliary and Fuel Handling Buildings via Station Vent

Releases from Auxiliary and Fuel Handling Buildings via Station Vent are considered as bypass accidents per Reference 7.21. The reduction factors for this pathway are a function of primary system retention and filtration by the building ventilation system. The RDF for this type of release is as follows:

$$RDF=(0.4)(.01)=0.004$$

Where

0.4 = Primary system retention and plateout reduction (Reference 7.21)

0.01 = Normal reduction for effective ventilation filters (Reference 7.21)

5.11.5 Releases from Reactor Building via Reactor Building Purge

Releases from the Reactor Buildings may occur through the purge ventilation system if the main or hydrogen purge valves are not shut. Radioiodines released into the containment building are subject to airborne iodine reduction from either natural processes (e.g., gravitational settling and plateout) or from Reactor Building sprays. They are also subject to reduction by filtration through the building ventilation system if the release is through the purge exhaust valves. The RDF for this type of release, with no Reactor Building Spray, is as follows:

$$RDF=(0.4)(0.01)=0.004$$

Where

0.4 = reduction by natural processes for minimal hold up time of 0.5 hours (Reference 7.21)

0.01 = Normal reduction for effective ventilation filters (Reference 7.21)

The RDF for this type of release, with Reactor Building Spray on, is as follows:

$$RDF=(0.01)(.03)=0.0003$$

Where

0.01 = Normal reduction for effective ventilation filters (Reference 7.21)

0.03 = Reduction by Reactor Building Spray for minimal hold up time of 0.5 hours (Reference 7.21)

The reduction factor for filtration of 0.01 applies to normal filter conditions. If the filters are known to be degraded (for example, due to moisture) but still partially functioning, this is accounted for in the RAC Model. The filters are either functioning at the full capacity, 99%, or from 0 - 99% due to degradation. If the leakage is not through the purge system filters, the reduction factor component for filtration is 1.

5.11.6 Releases from Fuel Handling ESF Vent System

During a fuel handling accident in the fuel handling building, noble gases and iodines may be released into the fuel pool and subsequently into the atmosphere of the fuel handling building. The ESF ventilation system would then exhaust the atmosphere in the building, through filter banks, to the environment. The RDF for this type of release is as follows:

$$RDF=(0.01)(0.01)=0.0001$$

Where

0.01 = Normal reduction for effective ventilation filters (Reference 7.21)

0.01 = Retention of iodines in the fuel pool water (References 7.21 and Reg. Guide 1.25)

5.12 Dispersion Model

5.12.1 Semi-Infinite Dose Model

The TMI-1 RAC model computes both whole body dose (TEDE) and thyroid dose (CDE) using a semi-infinite model. In the semi-infinite model, the ground is considered to be an infinitely large flat plate and the receptor is assumed to be standing at the center of a hemispherical cloud of infinite radius. The radioactive cloud is limited to the space above the ground plane.

In computing whole body dose, the semi-infinite plume model is based on the assumption that the dimensions of an effluent plume are large compared to the distance that gamma rays can travel in air. If the plume dimensions are larger than the gamma ray range, then the radius of the plume might just as well be infinite since radiation emitted from beyond a certain distance will not reach the receptor. This assumption forms the basis for the Dose Conversion Factors (DCFs) provided in Reference 7.42 which are used by the RAC Model to calculate whole body dose. These DCFs combine doses from external exposure, inhalation from the plume, and exposure for an individual immersed in a plume. The whole body (TEDE) DCFs used in the TMI RAC Model are as follows:

Isotope	DCF (mrem/hr)(μCi/cc)
Kr 85m	9.30E+04
Kr 85	1.30E+03
Kr 87	5.10E+05
Kr 88	1.30E+06
Xe 131m	4.90E+03
Xe 133m	1.70E+04
Xe 133	2.00E+04
Xe 135m	2.50E+05
Xe 135	1.40E+05
Xe 138	7.10E+05
I 131	5.30E+07
I 132	4.90E+06
I 133	1.50E+07
I 134	3.10E+06
I 135	8.10E+06

The TMI-1 RAC model calculates the thyroid dose rate due to inhalation of I-131, I-132, I-133, I-134, and I-135 in the hemispherical cloud. The thyroid dose rate is proportional to X/Q. The thyroid dose conversion factors are calculated using the child breathing rate of 0.42 m^3/hr from Table E-5 of Reference 7.35 and the child inhalation dose factors from Table E-9 of Reference 7.35. The dose is computed by multiplying the dose rate by the expected duration of release. The thyroid DCFs used in the TMI RAC Model are as follows:

Isotope	DCF (mrem/hr)(μCi/cc)
I 131	1.84E+09
I 132	2.21E+07
I 133	4.38E+08
I 134	5.76E+06
I 135	9.00E+07

The radioiodines are decayed during plume travel time. The decay constants for I-131 through I-135 are from the Radiological Health Handbook. They are as follows:

Isotope	Decay Constant (min^{-1})
Kr 85m	2.58E-03
Kr 85	1.23E-07
Kr 87	9.12E-03
Kr 88	4.04E-03
Xe 131m	4.09E-05
Xe 133m	2.20E-04
Xe 133	9.18E-05
Xe 135m	4.43E-02
Xe 135	1.27E-03
Xe 138	4.89E-02
I 131	5.99E-05
I 132	5.06E-03
I 133	5.53E-04
I 134	1.32E-02
I 135	1.75E-03

5.12.2 Calculations Used in the Model

5.12.2.1 X/Q Calculation:

The basis for the X/Q calculation is the Gaussian diffusion equation and a 10 x 7 array of sigma y's and sigma z's. The array of values correspond to sigma y's and sigma z's for 7 stability classes and at 10 fixed downwind distances. For distances other than the fixed downwind distances, the sigma y's and sigma Z's are linearly interpolated before X/Q is computed for that distance. The ten fixed distances are: 200, 500, 1000, 2000, 3000, 6000, 10000, 30000, 50000, and 80000 meters.

5.12.2.2 Compute Building Effect

Returns one of seven pre-computed virtual source distances, depending on stability class. The virtual source distances for each of the seven stability categories are 209,209,209,308,465,770 and 1254 meters, respectively. These values were computed based on the cross-sectional area of the nearest large building. Building wake effects are simulated by adding the virtual source distance, for a particular stability class, to the actual downwind distance for the purpose of computing X/Q. For example, suppose we wanted to know X/Q without building wake effects at 800 meters downwind with stability class D. X/Q would then be computed at 800 meters downwind. With building wake effects, X/Q would be computed at 1108 meters downwind (800 + 308) Thus building wake effect is simulated by computing X/Q at a distance greater than the actual downwind distance and is called only for ground level portion of release.

5.12.3 Other Calculations

5.12.3.1 Compute TMI-1 Emergency Action Level. Declares the emergency action level from highest dose whether whole body (TEDE) or thyroid (CDE).

5.12.3.2 Compute Site Boundary

The whole body and thyroid doses are computed at the site boundary. The distance to the site boundary varies with the compass sector that the wind is blowing to. This distance is in meters.

5.12.3.3 Compute Terrain Factor

Computes terrain height in meters for a given downwind distance. At downwind distances other than the given distances, terrain height is computed by linear interpolation, except at distances closer than 610 meters. Between the plant and 610 meters downwind, the terrain height is set equal to the terrain height at 610 meters. Terrain further from the plant is never lower than terrain closer to the plant due to mathematical approximations.

5.12.3.4 Compute Stability Class

As measured by the TMI Meteorological Tower from the 150 ft minus 33 ft temperature difference. Table 5.13-1 relates the temperature difference to the stability class. The equivalent temperature difference per 100 ft is shown in the last column of the table. Stability class is determined by the measured temperature lapse rate per Reg. Guide 1.21.

5.12.3.5 Adjust Wind Speed

Adjusts wind speed from the anemometer height to the release height. The wind speed is adjusted according to the following equation:

$$u = u_0(h/h_0)^p$$

where the subscript "0" denotes the anemometer height and "u" and "h" are the wind speed and height above ground, respectively. The exponent p is a function of stability: 0.25, 0.33 and 0.50 for unstable, neutral and stable cases, respectively. If the adjusted wind speed is less than 0.5 mph, the adjusted wind speed is set equal to 0.5 mph.

5.12.3.6 Compute Exit Velocity

Computes exit velocity of the released material in feet per second by dividing cubic feet per minute by the stack cross-sectional area.

5.12.3.7 Compute Plume Rise

Computes the plume rise in meters for the elevated portion of a spilt wake release. Two formulas are used to calculate the plume rise; for unstable and neutral conditions jet plume rise, momentum dominated, is calculated from Briggs' Plume Rise, Eq. 4.33; for stable stability, it is calculated using Eq. 4.28 from Briggs' Plume Rise.

5.12.3.8 Compute Entrainment Factor

Computes entrainment factor for wake split flows. A mixed mode release is assumed when: (1) the release point is at the level of or above adjacent solid structures but lower than elevated release points, (2) the ratio of plume exit velocity to horizontal wind speed is between one and five. Specifically, the entrainment factor, E_t, is computed according to the following formulas:

E_t = 1.0 for w_0/u le 1

E_t = 2.58 - 1.58(w_0/u) for 1 gt w_0/u le 1.5

E_t = 0.30 - 0.06(w_0/u) for 1.5 gt w_0/u le 5.0

E_t = 0 for w_0/u gt 5.0

where w_0 is the stack gas exit velocity and u is the wind speed at stack height in miles per hour.

Note that the entrainment factor does not address the case of two adjacent plumes mixing with each other, as would be the case in TMI-1, where it is possible for a clean plume and a contaminated plume to be emitted from adjacent but separate stacks. These plumes are examples of co-located adjacent jets; little is known about the modeling of co-located adjacent jets such as the ones at TMI-1.

5.12.3.9 Release Duration

The release duration is the best estimate of the time that it will take for the plant to be put in a condition that significantly reduces the current release rate of radioactivity from the plant. Significant is generally considered to be one to two orders of magnitude.

The release duration is a key parameter in the dose projection and decision making process, however, it is also somewhat subjective. The most conservative interpretation of release duration would be that point in the future when no further radioactivity will be released from the plant. This definition, however, is not appropriate for the dose projection process. Following a plant event that involves a radiological release, some abnormal level of radioactivity may continue to be released from the plant for days or even weeks following the event. However, there will be a point at which these release rates are insignificant compared to the release rates that were seen in the early stages of the event. The best estimate of the time to reach this point should be considered as the release duration for dose projection purposes.

If a significant radiological release to the environment is occurring, there is a leak (or series of leaks) from a plant component or system that is allowing it to get out. Such leakage could include RCS leakage, main steam releases to direct the environment, containment leakage due to high pressure in the reactor building, or combinations thereof. The release rate of radioactivity from the plant, and resulting offsite dose rates, are proportional to the rate of leakage from the plant component or system. Following the onset of the event, plant operators will take actions to stop the leak or reduce the driving forces that perpetuate the leak. When the leak rate has been reduced by one or two orders of magnitude compared to the leak rate present during the initial stages of the event, offsite dose rates are proportionally reduced. The time it is expected to reach this point should be considered the release duration. To assume that the release duration will be the time it takes to completely isolate the leak makes the assumption that offsite dose rates will remain at their initial levels even though the release rate from the plant will have been dramatically reduced. This can grossly overestimate potential offsite doses and may lead to event over classification and unnecessary protective action recommendations. As a result, the release duration should be the best estimate of the time that it will take for the plant to be put in a condition that significantly reduces the current release rate of radioactivity from the plant. Significant is generally considered to be one of two orders of magnitude.

TABLE 5.12-1

Classification of Atmospheric Stability

Stability Classification	Pasquill Categories	Delta T (150' - 33') (°F)	Delta T (°F/100') (°F)
Extremely Unstable	A	< -1.22	< -1.04
Moderately Unstable	B	≥ -1.22to< -1.09	≥ -1.04to< -0.93
Slightly Unstable	C	≥ -1.09to< -0.96	≥ -0.93to< -0.82
Neutral	D	≥ -0.96to< -0.32	≥ -0.82to -0.27
Slightly Stable	E	≥ -0.32to< +0.96	≥ -0.27to< +0.82
Moderately Stable	F	≥ +0.96to< +2.56	≥ +0.82to< +2.19
Extremely Stable	G	> +2.56	> +2.19

5.13 Liquid Release Calculation - In this section of the model calculations are performed for liquid source term determination, (see Figure 5.13-1 and Tables 5.13-1, 5.13-2) concentrations in the river, travel time to downstream users, and ingestion dose commitment calculations. The methods used to perform the calculations are as follows:

1. The concentrations of the liquid effluents are determined by sampling and analysis of the effluent stream. If sample results are not immediately available, previous Chemistry data may be used to provide a best estimate of the effluent concentrations. Only the four usual iodine isotopes (I-131, I-132, I-133, I-135), H-3, Cs-134, Cs-137, Co-58 and Co-60 are considered in the calculation.

2. The dilution in the river is calculated by first obtaining the river flow rate and inputting the value in the model. The flow rate may be obtained by the following methods:

   Calling the York Haven Dam at 266-3654 or,

   Calling the River Forecast Center at 1-814-234-9861

   The river flow rate is then used along with the discharge flow rate to calculate the concentration in the river. Complete mixing is assumed by the time it reaches the first drinking water station. The concentration in the river is then divided by the effluent concentrations in 10 CFR 20 Appendix B, Table 2, Column 2, to determine the fraction of these concentrations in the river to downstream users.

3. The river concentration is used along with the total discharge time, to calculate the dose commitment to an individual from drinking the river water from one of the downstream intakes. The river concentration is multiplied by the duration of the release, the usage factor, the ingestion dose commitment factor for infants (Reg. Guide 1.109), and the infant usage factor (330 liters/yr) to obtain an estimated dose commitment for the downstream drinker. The infant dose is used because the product of the usage factor and DCF shows that the infant is the maximum age group for the predominant isotopes.

4. A flume arrival time is estimated for each known downstream user. The river volume flow is used in an algorithm based on a model derived from river dye dilution and flow studies conducted from the TMI discharge. The algorithm uses the following equation:

$$\text{Time (hrs)} = \frac{\text{Distance to Downstream User (Miles)}}{[\text{River flow (cfs)} \times .0283\ (m^3/ft^3)]^{0.628} \times 0.019667}$$

5. If the concentration of any nuclides in the river exceeds the concentrations specified in 10 CFR 20 Appendix B, Table 2, Column 2, downstream users must be informed and recommended to curtail usage. Refer to Procedure 1203-44 for downstream users and telephone numbers.

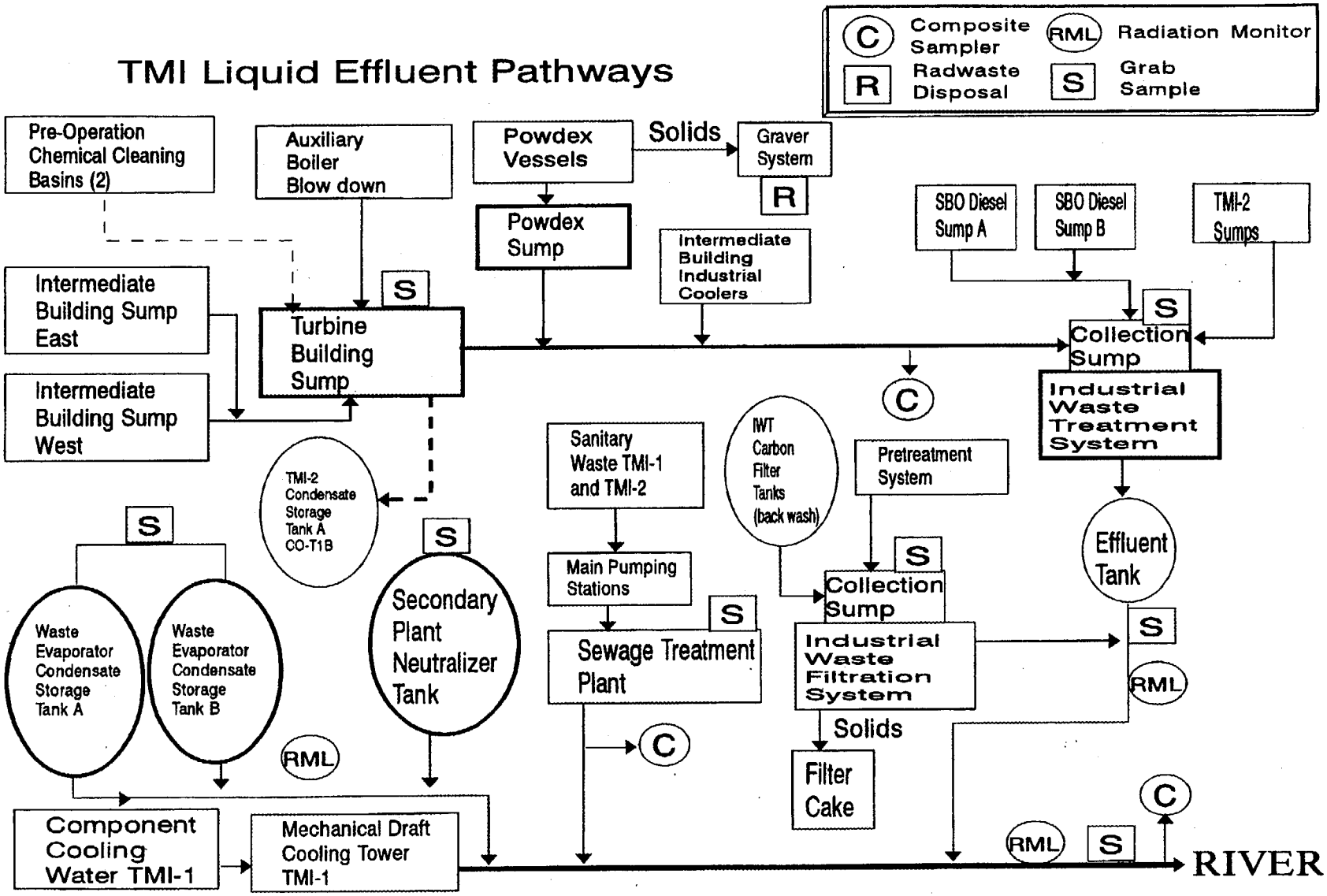

TMI Liquid Effluent Pathways
C Composite Sampler
RML Radiation Monitor
R Radwaste Disposal
S Grab Sample
Pre-Operation Chemical Cleaning Basins (2)
Auxiliary Boiler Blow down
Powdex Vessels
Solids
Graver System
R
Powdex Sump
Intermediate Building Industrial Coolers
SBO Diesel Sump A
SBO Diesel Sump B
TMI-2 Sumps
Intermediate Building Sump East
Intermediate Building Sump West
S
Turbine Building Sump
S
Collection Sump
Industrial Waste Treatment System
C
TMI-2 Condensate Storage Tank A CO-T1B
Sanitary Waste TMI-1 and TMI-2
IWT Carbon Filter Tanks (back wash)
Pretreatment System
Effluent Tank
S
S
Waste Evaporator Condensate Storage Tank A
Waste Evaporator Condensate Storage Tank B
Secondary Plant Neutralizer Tank
Main Pumping Stations
S
Collection Sump
S
Sewage Treatment Plant
Industrial Waste Filtration System
S
RML
RML
C
Solids
Filter Cake
C
Component Cooling Water TMI-1
Mechanical Draft Cooling Tower TMI-1
RML
S
RIVER

TABLE 5.13-1

TMI-2 Sump Capacity

Sump	Total Capacity	Gallons/Inch
Turbine Building Sump	1346 gals	22.43
Circulating Water Pump House Sump	572 gals	10.59
Control Building Area Sump	718 gals	9.96
Tendon Access Gallery Sump	538 gals	9.96
Control to Service Building Sump	1346 gals	22.43
Emergency Diesel Generator		
Sump A/B wet	837 gals	9.96
A/B dry	1200 gals	14.29
Chlorinator House Sump	-----	-----
Water Treatment Sump	1615 gals	22.43
Air Intake Tunnel		
Normal Sump	700 gals	-----
Emergency Sump	100,000 gals	766.00
Condensate Polisher Sump	2617 gals	62.31
Sludge Collection Sump	1006 gals	26.33
Heater Drain Sump	-----	-----
Solid Waste Staging Facility Sump	1476 gals	24.
Auxiliary Building Sump	10,102 gals	~202
Decay Heat Vault Sump	478.5 gals or 957 gals (total)	~ 10
Building Spray Vault Sump	478.5 gals or 957 gals (total)	~ 10

TABLE 5.13-2

TMI-1 Sump/Tank Capacities

Sump	Capacity (Gallons)
Turbine Building Sump (TBS)	10,000
Auxiliary Building Sump (ABS)	10,000
Reactor Building Sump (RBS)	10,000
Intermediate Building Sump West	1,000
Tendon Access Gallery Sump	1,000
Intermediate Building Sump East	1,000
Auxiliary Boiler Sump	2,000
Powdex Sump	40,000
Industrial Waste Treatment System Sump (IWTS)	300,000
Industrial Waste Treatment System Sump (IWFS)	80,000
Tanks	
TMI-2 Condensate B Tank	250,000
TMI-1 OTSG A or B (secondary)	25,000
TMI-1 WECST A or B	8,000
Neutralizer Tank	100,000
BWST	350,000
Condensate Tank A/B	265,000

5.14 Off-site Air Sample Analysis

5.14.1 Introduction

The results provided from field teams can be used to assess thyroid dose commitment.

The method involves collection of an air sample using a low flow (about 50 LPM) sampler with both a particulate filter and an iodine adsorber cartridge. The flow rate of the sampler, the duration of sample collection, the background of the frisker used to count the sample, the gross counts on the particulate filter, and the gross counts on the iodine cartridge are called into the RAC or EACC from the field teams. The RAC or EACC staff then can estimate the off-site dose commitment based on the sample.

5.14.2 Assumptions

A calibrated face loaded iodine cartridge was obtained and was used to determine the actual efficiency of a Eberline E140N with a HP-210/260 type probe to be used for counting in the field. The results of several tests on combinations of different probes and ratemeters showed a consistent 0.0039 (0.39%) counting efficiency. (Reference 7.7, 7.10, 7.11). Since I-131 has a fairly strong beta (0.6 MeV max.), the usual particulate filter counting efficiency of 0.1 (10%) is used. The collection efficiency for both filters for these calculations are assumed to be 1.0.

5.14.3 Calculation

The method first calculates the net counts per minute for the particulate and iodine cartridge. Then, using the given efficiencies separately, it calculates the air concentration of gaseous and particulate iodines. These are then combined for a total air concentration. A child breathing rate and dose conversion factor is then applied along with the estimated duration of exposure to obtain the off-site dose commitment.

Since the plant RAC model normally accounts for five different iodine isotopes, the dose conversion factor (DCF) used is a weighted average of the child DCFs based on the relative abundances of the five isotopes at damage classes of one and five with 100 minutes decay. This accounts for counts on the samples which will be caused by isotopes other than I-131.

5.14.4 Example:

Given an off-site air sample was taken with the following results:

Background = 100 cpm
gross cartridge countrate = 200 cpm
gross particulate countrate = 200 cpm
flow rate through sampler = 50 LPM
sample duration = 10 min.
exposure duration = 1 hour
DRCF = 4.0E8 $\frac{\text{mrem/hr}}{\mu\text{Ci/cc}}$

The RAC model follows the logic below to calculate an off-site thyroid dose commitment for this sample.

a. net particulate countrate = 200 - 100 = 100 cpm

b. net particulate activity = 100/.1 = 1000 dpm

c. net cartridge count rate = 200 - 100 = 100 cpm

d. net cartridge activity = 100/0.0039 = 25600 dpm

e. total activity in sample = 1000 + 25600 = 26600 dpm

f. total microcuries = 26600/2.22E6 = 0.012 μCi

g. sample volume = 50 * 1000 * 10 = 5E5 cc

h. air concentration = 0.012/5E6 = 2.4E-8 μCi/cc

i. dose commitment = 2.4E-8μCi/cc*1 hour*4E8 $\frac{\text{mrem/hr}}{\mu\text{Ci/cc}}$ = 9.6 mrem

5.15 Protective Action Recommendation Logic - The Logic Diagram is designed to enable the user to develop protective actions based primarily upon declaration of a General Emergency and also taking into account plant conditions, release duration and dose assessments. The logic is diagramed in Procedure EPIP-TMI-27, and EPIP-TMI-.02.

6.0 RESPONSIBILITIES

6.1 The RAC is responsible to ensure that dose assessments using the methodology in the EDCM are performed upon implementation of the Emergency Plan.

6.2 The RASE has the responsibility to support the RAC in performance of radiological controls and dose assessment using the methodology in the EDCM.

6.3 The Chemistry Coordinator has the responsibility to support the RAC in the procurement and analysis of in-plant samples required to quantify the accident.

6.4 Radiological Controls has the responsibility of proper review, and evaluation of the EDCM and to assist with the user interface of the RAC program software. Emergency Preparedness is responsible for ensuring that the EDCM and the RAC computer models are current and compatible.

6.5 The Technical Support Center (TSC) has the responsibility to provide the RAC with appropriate fuel damage data and other information pertinent to performing dose calculations.

6.6 The Emergency Preparedness Department and Radiological Engineering have the responsibility to ensure any changes to the RAC program are performed and documented IAW Procedure 1000-ADM-1230.10.

6.7 The Radiological Controls and Emergency Planning Departments have the responsibility to make any changes to the RAC Code, and to compile and distribute new versions of the program.

7.0 REFERENCES

7.1 American National Standard (ANS), ANSI/ANS-18.1 - 1984, Radioactive Source Term for Normal Operations of Light Water Reactors

7.2 APS Source Term Report - Report to the American Physical Society of the Study Group on Radionuclide Releases From Severe Accidents at Nuclear Power Plants, February 1985

7.3 Dose Assessment Manual for Emergency Preparedness Coordinators, February 1986, INPO 86-008

7.4 Efficiency Check using an Air I-131 Source Cartridge and a Ba-133 Source Cartridge, Memorandum 9502-88-0139, September 28, 1988

7.5 EPA 520/1-75-001 - Manual of Protective Action Guides and Protective Actions for Nuclear Incidents

7.6 EPIP-TMI-.07 - Off-site/On-site Dose Projections

7.7 Evaluation of a Front Loaded Iodine Cartridge using Various Survey Equipment, Memorandum 9100-88-0194, May 12, 1988

7.8 Field Measurements of Airborne Releases of Radioactive Material, Memorandum 9502-88-0098, May 25, 1988

7.9 FSAR, TMI-1 Chapter 11, Radioactive Waste and Radiation Protection

7.10 FSAR, TMI-1 Chapter 14 - Safety Analysis

7.11 TMI-2 PDMS Safety Analysis Report

7.12 TMI Emergency Plan, 1000-PLN-1300.01

7.13 ICRP-23 - Report of the Task Group on Reference Man, 1981

7.14 INPO SOER 83-2 - "Steam Generator Tube Ruptures"

7.15 Introduction to Health Physics, Herman Cember, 2nd Edition, 1985

7.16 NRC-BNL Source Term Report

7.17 NUREG-0017 PB-251 718 - PWR - GALE Code; Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from PWR, Revision 1, 1985

7.18 NUREG-0133 - Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants, October 1978.

7.19 NUREG-0654 - Revision I - Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants

7.20 NUREG-0737 - Clarification of TMI Action Plant Requirements, U.S. Nuclear Regulatory Commission, November 1980, Generic Letter 82-33, Supplement 1 to NUREG-0737 - Requirements for Emergency Response Capability, U.S. Nuclear Regulatory Commission, Washington, D.C., December 1982

7.21 NUREG-1228 - Source Term Estimation during Incident Response to Severe Nuclear Power Plant Accidents, October 1988

7.22 NUREG/CR-3011 - Dose Projection Considerations for Emergency Conditions at Nuclear Power Plants

7.23 N1830 - Post Accident Reactor Coolant Sampling

7.24 N1831 - Post Accident Atmospheric Sampling

7.25 N1832 - Post Accident Sample Analysis

7.26 Deleted

7.27 1210-10 - Abnormal Transients Rules, Guides and Graphs

7.28 1210-8 - RCS Super Heated

7.29 1202-12 - Excessive Radiation Levels

7.30 Operational Quality Assurance Plan, 1000-PLN-7200.01

7.31 Proprietary Midas User Documentation, Pickard, Lowe, and Garrick

7.32 Radioactive Decay Data Tables, David C. Kocher, ORNL, DOE/TIC-11026, 1981

7.33 Radiological Health Handbook, Revised Edition Jan. 1970, US Dept. HEW

7.34 Reg. Guide 1.21 Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water Cooled Nuclear Power Plants, Rev. 1, June 1974.

7.35 Reg. Guide 1.109 - Calculation of Annual Doses to Man From Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10CFR Part 50, Appendix I, October 1977, Rev. 1

7.36 SER-419628-003, Rev. 7, Instrument Calibration Facility, Feb. 12, 1988

7.37 TDR-390 - TMI-1; Primary-to-Secondary OTSG Leakage and its On-site/Off-site Radiological Impact, April 1983

7.38 TDR-405 - TMI-1; Evaluation of Plant Radiation Release and its 10CFR50, Appendix I Conformance for Different Operating Conditions

7.39 TDR-431 - Method for Estimating Extent of Core Damage Under Severe Accident Conditions

7.40 WASH-1400 - 1975 Nuclear Safety Study WASH-1400 (also known as Rasmussen Report)

7.41 Sutron Report No. SCR-358-82-063, Study of Travel Time and Mixing Characteristics for the Susquehanna River Below Three Mile Island

7.42 EPA 400-R-92-001, Manual of Protective Action Guides and Protective Actions for Nuclear Incidents

7.43 RAF 6612-97-019, "Source Term Algorith for RMG-22/23

7.44 RAF 3640-98-19, "Documentation of Calculation of Direct-to-Atmosphere Mass Flow Rates"

8.0 EXHIBITS

Exhibit 1 - Emergency Dose Assessment User's Manual

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