ENVIRONMENTAL QUALIFICATION - P/T PARAMETERS

1.1 HYDRAULIC MODEL

Blowdown of the reactor coolant system following an assumed rupture has been simulated by using a modified version of the FLASH code. This code calculates transient flows, coolant mass and energy inventories, pressures, and temperatures during a loss of coolant accident. The code calculates inflow from the emergency cooling strings and calculates heat transferred from the core to the coolant.

Modifications were made to FLASH to make the model more applicable to this system. The changes are as follows:

1) The calculation of reactor coolant pump cavitation was based on the vapor pressure of the cold leg instead of the hot leg water.

2) Core flooding tanks have been added. Water flow from the core flooding tanks is calculated on the basis of the pressure difference between the core flooding tanks and the point of discharge into the reactor coolant system. The line resistance and the inertial effects of the water in the pipe are included. The pressures in the tanks are calculated by assuming an adiabatic expansion of the gas above the water level in the tank. Pressure, flow rate, and mass inventories are calculated and printed out in the computer output.

3) Additions to the water physical property tables (mainly in the subcooled region) have also been made to improve the accuracy of the calculations.

4) A change in the steam bubble rise velocity has been made from the constant value in FLASH to a variable velocity as a function of pressure. The bubble velocity term determines the amount of water remaining in the system after depressurization is complete. For large ruptures, this change in velocity shows no appreciable change in water remaining from that predicted by the constant value in the FLASH code. For smaller ruptures, an appreciable difference exists. The variable bubble velocity is based on data in reference and is adjusted to correspond to data from the LOFT semiscale blowdown tests.

   Test No. 546 from the LOFT semiscale blowdown tests is a typical case for the blowdown through a small rupture area. A comparison of the predicted and experimentally observed pressures is shown in Figure 6B-1, Figure 6B-2 shows the percent mass remaining in the tank versus time, as predicted by the code. At the end of blowdown, the predicted mass remaining is 13 percent. The measured mass remaining is approximately 22 percent.

5) An addition was made which provided for simulation of the vent valves in the reactor vessel core support shield.

The FLASH code describes the reactor coolant system by the use of two volumes plus the pressurizer volume. The system was grouped into volumes on the basis of the temperature distribution in the system as follows:

Volume 1 includes half of the core water volume, the reactor outlet plenum, the reactor outlet piping, and approximately 55 percent of the steam generators. Volume 2 includes half of the core water volume, the reactor inlet plenum and downcomer section, the reactor inlet piping, pumps, and 45 percent of the steam generators.

Volume 3 represents the pressurizer.

The resistances to flow were calculated by dividing the reactor coolant system into 24 regions and calculating the volume-weighted resistance to flow for a given rupture location based on normal flow resistances. For the double-ended ruptures, all of the leak was assumed to occur from the volume in which that pipe appeared.

The reactor core power was input as a function of time as determined by the CHIC-KIN code in conjunction with the FLASH output. Steam generator heat removal was assumed to cease when the rupture occurred. Stored heat in the reactor coolant system was considered in the analysis.

The modified FLASH code has the capability of the simulating injection flow from the core flooding tanks. Reactor vessel filling was calculated by adding the mass remaining in the vessel as predicted by FLASH to the mass injected from the core flooding tanks. This method of calculation is conservative in that condensation of steam by the cold injection water is not taken into account. An analysis using the FLASH code with condensation effects confirms that conservatism is used in this analysis.

Pressure, temperature, mass and energy inventories, and hydraulic characteristics as determined by FLASH, are input into the core thermal code (QUENCH) and the reactor building pressure buildup (CONTEMPT).

1.2 REACTOR BUILDING DESIGN BASIS ACCIDENT

For the LOCA, the Reactor Building pressure was evaluated for a range of rupture sizes with the reactor operating at full power per the digital computer codes described in the previous section 1.1. Reactor Coolant System Accident Simulation were used to perform this analysis. For this analysis, core cooling is provided by two core flooding tanks and a single ECCS string. The ECCS string is assumed to operate on emergency power.

The results of calculations of fluid and heat transport to the reactor building as determined by FLASH and PRIT were used as input to the reactor building pressure analysis program CONTEMPT. FLASH covers the blowdown period and PRIT covers the post-blowdown period. In the PRIT code, as the injected coolant water covers the core, heat is transferred from the core and the reactor coolant system metal to the water. The heat transfer coefficients used in this analysis are shown in Items a through d of Table 6B-1. All heat transferred from the core and the reactor coolant system metal is assumed to generate steam which goes directly to the reactor building atmosphere until the reactor vessel is filled to the nozzle height. Thereafter, all energy is removed by the injection flow of subcooled water, and the energy release to the reactor building atmosphere is terminated.

Both reactor inlet and reactor outlet pipe breaks were analyzed with FLASH. However, a complete reactor building analysis was made only for the reactor outlet pipe breaks. Upon release of hot reactor coolant, the structural steel and concrete act as heat sinks which reduce the reactor building pressure. The heat sink data considered in this analysis are specified in Table 6.6-7. The heat transfer coefficients applied to these surfaces are included in Table 6B-1.

During a LOCA, the reactor building is cooled by two independent safeguard systems (see Section 6). Each system is designed so that it alone can protect the reactor building against over pressure. These systems are initiated by high reactor building pressure and operate on emergency power. Engineered safeguards data are shown in Table 6B-2 and the general parameters used in the Reactor Building pressure analysis are shown in Table 6.6-8.

Figure 6B-3 shows the Reactor Building peak pressure as a function of rupture size. The parameters shown in Table 6.6-7 and 6.6-8 and Figures 6B-1 and 6B-2 were used with the exception that only the air recirculation coolers were used to provide building cooling. Table 6B-3 shows a tabulation of pertinent results for this spectrum of break sizes. Based on the these results, the reactor building design basis accident (DBA) was originally determined to be the 8.5 ft^2 hot leg break.

Figures 6B-4, 6B-5, and 6B-6 show the Reactor Building pressure, temperature and energy inventory as a function of time after rupture for the original DBA for three fan coolers and no spray pumps which is retained for comparative and historical purposes in the FSAR.

Figures 6B-7 through 6B-10 show the Reactor Building pressure as a function of time after rupture for the other rupture sizes.

To demonstrate the effectiveness of Reactor Building spray cooling, the original DBA was analyzed with only 1 of the 2 spray systems operating and with a reduced air recirculation cooling capacity (2 air coolers). The peak pressure for this case (see Table 6B-3 Figure 6B-11) is approximately the same as that for the original DBA since the peak pressure occurs very shortly after safeguards go into operation.

The TMI-I Environmental Qualification Temperature & Pressure Profiles were reanalyzed with the GOTHIC (Generation of Thermal Hydraulic Information for Containments) computer program. GOTHIC is an EPRI sponsored code that was developed by Numerical Application, Inc. for containment analysis. The code was originally developed from COBRA, a fuel heatup code.

The containment pressure and temperature response after a Design Basis LOCA was analyzed in Reference 6.

The results for the LOCA Temperature profile are shown in Figure 6.B-16. The pressure for the LOCA analysis, included for comparative purposes, is shown in Figure 6.B-17. The profile incorporates the assumptions in Table 6B-6. The GOTHIC Analysis was performed with a single failure consistent with the peak pressure analysis documented in Section 6.6. A single failure was assumed in the analysis, only one ECCS train, one Reactor Building Spray train, and one fan cooler are operational. The GOTHIC fan cooler model incorporates a volumetric flow of 25000 cfm for each unit, a cooling water temperature of 95ºF, and cooling water flow of 1450 gpm. This model calculates the fan cooler heat removal rate based on the calculated containment conditions (i.e. pressure, temperature, and humidity). The fan cooler model in the (historic analysis CONTEMPT code) was only a function of temperature. The 7.0 ft^2 cold leg pump suction break was selected because it generated the worst peak pressure with this single failure as shown in Table 6.6-1.

2.0 ENVIRONMENTAL QUALIFICATION PROGRAM

2.1 INTRODUCTION

2.1.1 PURPOSE

The purpose of the Environmental Qualification (EQ) Program for TMI-1 is to provide assurance that specific electrical equipment defined below will perform its intended function. Specifically, the object of the EQ Program is to:

a. Document the qualification of TMI-1 electrical equipment important to safety as required by 10CFR50.49.

b. Establish the maintenance/surveillance required to maintain the qualification of this electrical equipment over the life of the plant.

2.1.2 SCOPE

The equipment within the scope of this program includes the following:

1. Safety related (Class 1E) electrical equipment required for one path to hot shutdown.

2. Non-Class 1E electric equipment whose failure under postulated environmental conditions could prevent satisfactory accomplishment of safety functions performed by Class 1E equipment.

3. Certain postaccident monitoring equipment (RG. 1.97, Rev. 3).

   Only that equipment in the above categories which is contained in a potentially harsh accident environment is within the scope of this program (See 10 CFR 50.49, subparagraph (c)).

2.1.3 PROGRAM ELEMENTS

The foundation of the environmental qualification program is comprised of four elements:

1. The evaluation of environmental service conditions in the plant areas to identify the harsh environments caused by design basis event.

2. The identification of systems and associated components required to mitigate design basis events, and the identification of those components located in the harsh environments.

3. The preparation of environmental qualification documentation for components located in harsh environments.

4. The maintenance and surveillance of the installed components to ensure preservation of environmental qualification.

2.1.4 EQ POSITION ON DOR GUIDELINES, AND R.G. 1.89

The EQ Program for TMI-1 meets the requirements of 10CFR50.49. All equipment within the scope of this program has been evaluated for compliance with either the DOR Guidelines or 10CFR50.49 with guidance from Regulatory Guide 1.89.

TMI-1 was an operating plant when the DOR Guidelines were issued in November 1979. Therefore, installed equipment was required to meet the requirements of the DOR Guidelines.

Replacement parts should be qualified to 10CFR50.49 unless there are sound reasons to the contrary. This NRC position is designed to promote the policy of upgrading the environmental qualification and reliability of installed electric equipment. Situations may arise, however, in which such upgrading will not be feasible or compatible with overall plant safety. Exelon must review each situation on a case-by-case basis to determine that "sound reasons to the contrary" do exist to justify an exception from upgrading. Acceptable "sound reasons to the contrary" are included in NRC Regulatory Guide 1.89, Revision 1, Section C, Paragraph 6, a through g.

The cut-off date of February 22, 1983 is used for determining, in general, whether a component requires qualification to the DOR Guidelines or to 10CFR50.49.

(Reference: Section C, Paragraph 6 of Regulatory Guide 1.89, Rev. 1). Qualifying equipment which was purchased before this date and which is in the warehouse may be accomplished within the criteria of Section C, Paragraph 6 of this Regulatory Guide. After this date, replacement components or subparts will be evaluated to ensure compliance with Regulatory Guide 1.89 - "Criteria For Sound Reasons To The Contrary".

2.1.5 ENVIRONMENTAL QUALIFICATION PROGRAM BACKGROUND

Safety related electrical equipment in nuclear facilities must be capable of performing their safety related functions under all normal, abnormal, and accident conditions. This requirement is embodied in the General Design Criteria 1,2,4 and 23 of Appendix A to 10CFR50, and in 10CFR50.55 a (h) which incorporates by Reference IEEE Standard 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations."

2.2 EQ ORGANIZATION

Implementation of the EQ program at TMI-1 is the responsibility of the Vice President TMI-1.

The evidence of qualification for Class 1E equipment is established through a disciplined program which assures generation and collection of necessary documents in each stage of the process, and maintain them current. The general process of equipment qualification begins with the development of the initial system design and continues through the specification/procurement of the equipment, its installation, operation, maintenance, and replacement.

2.3 EQ MASTER LIST

The TMI-1 EQ Master List, (EQML) is comprised of all TMI-1 components in the Component Record List (CRL) which have a “Y” in the EQ field on the Codes & Classifications page of the CRL entry. The EQML is not maintained as a separate document. All EQML components are electrical equipment or components which must be environmentally qualified for use in a harsh environment. The methodology and bases used to develop the EQ Master List are presented below.

2.3.1 SELECTION OF SYSTEMS WITHIN THE EQ PROGRAM

The first step in defining the Master List of electrical equipment requiring environmental qualification was to develop the lists of systems and associated electrical components required to function during or subsequent to the postulated accidents so as to bring the plant to, and maintain it in, hot shutdown.

The basis for this list was a detailed review of the following documents which describe TMI-1 in detail:

a. TMI-1 FSAR
b. TMI-1 Restart Report
c. TMI-1 Operating Procedures
d. TMI-1 Emergency Procedures
e. TMI-1 System Descriptions
f. SSD, SAD, SSLD
g. Lesson Plans
h. Piping and Instrumentation Drawings
i. Technical Specifications
j. IE Bulletin 79-01B, Appendix A
k. IE Supplement No. 2 to Bulletin 79-01B, Table II

As a check for completeness of the systems included within the scope of the program, a functional systems analysis was conducted. The DOR Guidelines state that the Master List of equipment should include all electrical equipment needed to achieve the following safety functions:

a. Emergency Reactor Shutdown
b. Containment Isolation
c. Reactor Core Cooling
d. Containment Heat Removal
e. Reactor Heat Removal
f. Prevention of Radioactive Material Release to the Environment

2.3.2 IDENTIFICATION OF ELECTRICAL EQUIPMENT

2.3.2.1 Criteria for Selection of Equipment

The EQ program addresses all electrical equipment important to safety which is located in a potentially harsh environment. Electrical equipment important to safety is defined in 10CFR50.49(b) (1), (2) and (3).

For TMI-1 "safe shutdown" is defined as a hot shutdown condition.

Equipment important to safety which would not be exposed to a harsh environment during postulated accident conditions (i.e., mild environment) is not included in the EQ Program.

A mild environment, as defined in 10CFR50.49 (c) "...an environment that would at no time be significantly more severe than the environment that would occur during normal plant operation, including anticipated operational occurrences."

2.3.2.2 Class 1E Equipment and Interfaces

A detailed review of each component included within the EQ Program was performed. This was accomplished by listing each major electrical component (motors, instrument, etc.) and then identifying the auxiliary electrical equipment within the control/power circuit of the component (cable, terminal blocks, splices, etc.). This identification of the auxiliary electrical equipment was carried out through extensive review of the electrical one lines, elementary wiring diagrams, circuit schedules, pull/termination sheets, and by plant walkdowns.

Class 1E equipment which does not have specific Equipment ID Numbers (i.e., cable, electrical accessories) was identified as common equipment.

2.3.2.3 NUREG-0737 and Regulatory Guide 1.97 Equipment

Supplement 1 to NUREG 0737 requires that certain post accident monitoring instrumentation be provided to enable operators to assess plant and environmental conditions during and following an accident. The post accident monitoring instrumentation is selected using the guidance provided by ANSI/ANS 4.5-1980 as endorsed by Regulatory Guide 1.97. R.G. 1.97 Category 1 and 2 instrumentation located in a harsh environment is included in the EQ Program.

2.3.2.4 Equipment Operability Time

The operability duration is the length of time during and following an accident that equipment must maintain its ability to perform its safety function. The safety function includes:

1. The ability to initiate short term protective action.
2. The ability to place the plant in a controlled condition.
3. The ability to keep the plant in a stable condition after the accident and until personnel are able to enter the plant to inspect, repair, or replace equipment.

Components requiring environmental qualification were categorized into one of five time periods assumed for them to accomplish their safety functions.

Operability Time	Basis
1 hour	Equipment which performs its function within the first few minutes of the accident.
19.5 hours	For components inside containment, the containment pressure temperature analysis indicates that containment conditions return to normal or below normal operating conditions within 19.5 hours.
46 hours	Time to reach cold shutdown via natural circulation.

30 days	Regulatory Guide 1.3, Revision 2 evaluates the offsite radiological consequences of a LOCA event for a maximum of 30 days following the accident.
6 months	For the NUREG-0588 review a post-DBE maximum operability of 6 months was utilized. This value was selected as a conservative bounding time for termination of accident effects within the containment. Within a 6 month period the resources of the nuclear industry will be available for modification or repairs.

2.4 ENVIRONMENTAL PARAMETERS

The environmental parameters for each plant area were determined for both normal and accident service conditions.

2.4.1 NORMAL SERVICE CONDITIONS

The plant normal service conditions include all aspects of normal operation, including all levels of power generation, startup, hot standby, hot shutdown, cold shutdown refueling, and any other normally anticipated operational occurrence.

The normal service conditions for a specific component encompass the applicable temperature, pressure humidity, and radiation postulated to occur at the specific component locations during normal operation of the plant. The methodology used to define the normal service conditions is described below.

2.4.1.1 Temperature/Pressure

The temperatures inside the containment were obtained from measurements taken during normal operation. The temperatures in the intermediate and auxiliary buildings were obtained from discussions with plant personnel and are based on conservative plant experience. Atmospheric pressure (14.7 psia) was assumed for all area.

2.4.1.2 Humidity

Per Federal Register Volume 48, No. 15, Page 2732, item (5) Humidity – Paragraph 50.49(e)(2), the Commission agrees that humidity variation during normal operation are difficult to predict. It has not been demonstrated that the time-dependent variations in humidity will produce any differences in degradation of electric equipment. The words “Time-dependent variation of relative” have been deleted from Paragraph 50.49(e)(2).

A 50 percent relative humidity is considered for the Control Building.

2.4.1.3 Radiation

Normal power operation general area radiation dose rate maps for the containment and auxiliary building have been established through the use of TMI-1 health physics dose rate surveys as well as from the radiation shielding design basis for the plant. The intermediate building has negligible radiation levels.

The radiation dose is integrated over a specified normal plant operating time (40 year normal operation plus 6 month accident).

2.4.1.4 Dust

In a nuclear power plant there are typically three major causes of dust inside plant buildings: transport of matter from the outside through ventilation intakes; deterioration of uncoated concrete floors; and residue from maintenance actions. However, dust is not a factor in equipment qualification.

2.4.2 ACCIDENT SERVICE CONDITIONS

The development of accident service conditions for TMI-1 considered the environmental conditions resulting from a postulated Loss of Coolant Accident (LOCA), Main Steam Line Break (MSLB) inside containment, MSLB in the Intermediate Bldg, and an EFW Pump turbine steam line break in the Intermediate Bldg.

The analyses of these postulated accidents address the following environmental parameters:

a. Temperature/Pressure
b. Humidity
c. Chemical Spray
d. Submergence
e. Radiation

The specific analyses performed and their results are discussed in greater detail below.

2.4.2.1 Temperature/Pressure

LOCA inside Containment

The analyses performed (Reference 6) to determine the containment temperature and pressure response to a LOCA is described previously within this Appendix. The time dependent temperature and pressure profiles used in the TMI-I environmental qualification program are shown in Figures 6B-16 and 6B-17. The LOCA is the most severe accident in containment based on total energy released.

The original OTSGs were replaced at the end of Cycle 17. An evaluation of post-LOCA containment response with the replacement OTSGs is contained in Reference16 for qualification of EQ equipment in containment. The analysis credits partial re-vaporization for short-term temperature excursions consistent with the approach included in NUREG-0588. This approach considers the impact of thermal lag which ensures equipment will not heat up significantly during a temperature spike. Consequently, the initial temperature peak can be ignored and the long-term temperature response of the atmosphere, which is essentially saturation temperature at the containment pressure, dominates the temperature response of equipment inside the building.

Main Steam Line Break (MSLB) inside Containment.

Environmentally qualified SSC inside containment must be qualified to the LOCA and MSLB environment. C-1101-823-E610-014, "TMI-1 MSLB Containment Response," determined a significantly higher EQ peak temperature for the MSLB than for the LOCA. Qualification of SSC in containment is based on vendor qualification testing of equipment in excess of the MSLB temperature profile, or “thermal lag” modeling and analysis contained in C-1101-823-E610-014. The MSLB temperature profile is depicted in Figure 6B-15.

Figure 6B-19 depicts the LOCA enveloping curve from Figure 6B-16 with the RB MSLB profile added where it exceeds the LOCA envelope. This is the EQ temperature profile which all EQ components inside containment must meet, either by enveloping or via “thermal lag” analysis. Figure 6B-20 shows the LOCA and RB MSLB pressure profiles, with the enveloping EQ profile.

LOCA outside Containment

A loss of coolant accident pipe break is not postulated to occur outside the containment for environmental qualification of components.

HELBS Outside of Containment

A high energy line break (HELB) can only produce a harsh environment in the Intermediate Building. The methodology, assumptions, and result of analyses which determine the temperature/pressure response to HELBs outside of containment (in the Intermediate Building) is described in EDS Nuclear Report 02-0370-1058, "Pressure and Temperature Conditions Following High Energy Steamline Breaks in the Intermediate Building", Revision 2, June 1981.

This report gives post accident pressure and temperature profiles for two double-ended guillotine pipe breaks in the main steam system at the worst locations from an equipment qualification viewpoint. One break was a 24 inch main steam line break at elevation 322'-0" and the other was an 12 inch main steam to EFW pump turbine line break. The 24 inch main steam line break is the limiting and enveloping break in the Intermediate Building.

Other Line Breaks Outside of Containment

Reference 13 determines the environmental conditions from a crack break in the Auxiliary Steam Line. The environment was analyzed for the auxiliary building, fuel handling building and the control building patio area. Based on further evaluations it was concluded that the elevated temperatures due to the crack do not affect the operability of any electrical equipment and as such does not require any Environmental Qualification program

2.4.2.2 <u>Humidity</u>

The only plant areas where a saturated environment (100 percent RH) could occur during accident conditions are:

a. Containment
b. Intermediate Building

2.4.2.3 <u>Chemical Spray</u>

The Reactor Building Sump is designed to accept a boric acid solution 1.3 percent mixed with enough Trisodium Phosphate (TSP) to maintain a Building Sump pH of at least 7.3. The volume of TSP required to be contained in the baskets will ensure that the final containment recirculation sump pH after injection will be between 7.3 and 8.0.

For equipment qualification purposes, equipment inside containment should be qualified for a pH range as follows:

pH	Time Duration
4.0	0 – 7.5 hours
10.0	7.5 – 24 hours
7.3 – 8.0	24 – 52 hours

Ref. KCI Engineering Consultants Technical Evaluation No. A2037848, E26

2.4.2.4 Submergence

The only plant areas where equipment could be submerged during postulated accident conditions are in the Containment and the Intermediate Building. For all other plant areas, the design of floor drains will prevent submergence of electrical equipment. The maximum allowable flood levels inside the Containment and Intermediate Building (Reference 12) are as follows:

Containment	Elev. 286.85'*
Intermediate Building	Elev. 296.2' (25 min.)

* References 9 and 10 address the elevations of various instruments in the TMI Unit 1 Reactor Building. The lowest point on all instruments addressed is 70.25 inches above the RB floor (elevation 281'0). The maximum flood level inside the containment is lower than this lowest point (Reference 11).

2.4.2.5 Radiation

For equipment qualification purposes, the accident radiation conditions postulated to occur at TMI-1 result from a LOCA inside of containment and were developed based upon the bounding requirements of the DOR Guidelines or NUREG-0588, as applicable to the component being qualified. Thus, qualification cannot be affected by reload core inventory changes.

The accident radiation doses, gamma and beta (if applicable) were integrated over the duration of the accident (which is usually taken as 6 months). This is a conservative approach if the required component operating time is appreciably less than the radiation integration time (e.g., hours versus months, respectively).

2.4.2.5.1 Inside Containment - General areas

The percent of core inventory assumed to be released from the fuel for a LOCA meets the NUREG-0588 requirements of:

100 percent of the Noble Gas Core Inventory; 50 percent of the Iodine Core Inventory; and, 1 percent of the other nuclides in the Core Inventory.

Gamma Dose

The source term, basic assumptions and model used to develop the total gamma dose radiation service condition for Class 1E equipment located in general areas inside of containment, are described in NUREG-0588, Appendix D. Table D-1 of Appendix D gives an estimated value of 1.54×10^7 RADs for the total airborne gamma dose at the center of the containment (6 months to 1 year integrated dose from airborne iodine, airborne noble gases and plateout iodine). Also, the nomogram in 79-01B (DOR Guidelines) for a containment volume of 2×10^6 cuft and reactor power of 2600 MWth yields a 30 day integrated dose of 1.5×10^7 R. The 30 day dose is approximately 92% of the 6 month dose, therefore the TID would be 1.63×10^7 R.

A specific gamma dose was not calculated for TMI-1; therefore, in accordance with the DOR Guidelines, Section 4.1, a total gamma dose radiation service condition of 2×10^7 RADs was utilized.

Beta Dose

The source term, basic assumptions and model used to develop the total beta radiation dose for Class 1E equipment located in general areas inside of containment are described in NUREG-0588, Appendix D.

Table D-2 of Appendix D, gives an estimated value of 1.83×10^8 RADs for the total airborne beta dose at the center point in the containment (6 month integrated dose from airborne iodine and airborne noble gases). A specific beta dose was not calculated for TMI-1. Therefore, in accordance with DOR Guidelines, Section 4.1, a conservative total beta dose radiation source condition of 2.0×10^8 RADs was utilized. This beta surface dose was reduced by an order of magnitude within 30 mils of the surface of electrical cable insulation of unit density (Reference Section 4.1 of the DOR Guidelines). Therefore, for cables with 30 mils or greater insulation, beta radiation is 2×10^7 RADs.

The above radiation service conditions inside of containment are applicable to equipment in the containment vapor space as well as to equipment submerged in the containment sump fluid.

2.4.2.5.2 Outside Containment

The predominant source of radiation outside of containment is considered to be the radioactive recirculating fluids carried in piping. Containment "shine" was calculated to yield a direct gamma dose of less than 100 RADs near the edge of the Reactor Building.

Therefore, containment "shine" was not considered in those areas containing radioactive piping since it would be an insignificant contributor compared to the recirculating fluids. Additionally, for areas not containing radioactive piping, containment "shine" would not yield a harsh radiation environment because it is less than 10^4 RADs.

The following is the description of source terms for the affected plant areas.

Source Terms

For the evaluation of a similar plant, Midland was used to represent TMI. The Midland core was modelled as an equilibrium cycle of 310 EFPD (effective full power days) at a power level of 2552 MWth, and used to develop the source terms in Table 6B-5. A comparison of dose rates using the above power history with dose rates from a 2568 MWth, 661 EFPD core showed that calculated dose rates increased by less than 5%. This TMI Cycle 10 core as modelled is typical of expected future TMI cycles. There are no expected effects on postaccident equipment qualification due to this revised inventory, since this modest increase is accommodated by known conservatisms in the evaluation.

The activity assumed for source term calculations was based on the following:

a. Liquid Containing Systems: 100 percent of the core equilibrium noble gas inventory, 50 percent of the core equilibrium halogen inventory, and 1 percent of all others. In determining the source term for recirculated, depressurized cooling water, it was assumed that the water contains no noble gases (noble gases are assumed to be released to the containment atmosphere upon depressurization).

b. Gas Containing Systems: 100 percent of the core equilibrium noble gas inventory and 25 percent of the core equilibrium halogen inventory.

Two liquid source terms were used in the evaluation. For systems or portions of systems which will contain post accident fluid recirculated from the reactor building sump, the source term was based on diluting the isotopic inventory discussed in item a. above with the minimum expected volume of fluid in the bottom of the reactor building post accident. This volume includes that of the Borated Water Storage Tank (BWST) and the Reactor Pressure Vessel (PV). This is designated as the "Recirculation" source.

For systems which will contain post accident fluid from the reactor coolant system which will not be diluted as noted above, the source term was based on diluting the isotopic inventory in item a. above with the volume of fluid in the reactor coolant system. This is designated as the "Reactor Coolant" source.

The activity assumed for the containment gaseous source term calculations was based on 100 percent of the noble gas core inventory and 25 percent of the halogen core inventory. The containment airborne source term was based on diluting the isotopic inventory of item b. above with the air contained in the containment free volume. This is designated as the "Containment Gas" source. The inventories and source terms discussed above were calculated for the time period immediately after the postulated accident. For other time periods, the decay parameters given in References 4 and 5 were used to adjust the source terms for radioactive decay. The sources were converted to standard shielding source term format as a function of time after the postulated accident and were used as the basic input data to the shielding codes.

2.4.2.6 Duration of Containment Harsh Environment Service Conditions

Harsh environment service conditions exist for the course of the accident. The duration of the harsh environment could be longer or shorter than the equipment operability time. The listing below summarizes the duration of the accident parameters for environmental qualification purposes.

Parameter	Duration
Temperature/ Pressure	As given on the profile for LOCA, MSLB or HELBS, the containment pressure-temperature analysis indicates that containment conditions return to normal operating conditions within 19.5 hours.

Humidity	Assumed to be 100 percent for the duration of the LOCA or HELB profile.
Chemical Spray	For the duration of the LOCA/HELB in containment until the containment pressure goes below 4 psig (sprays can run for a maximum of 30 days for radiological considerations).
Submergence	Increase in flood level based on calculation, then constant level until the water is pumped out.
Accident Radiation	Radiation doses are usually integrated over a period of 6 months.

The duration of the harsh environment parameters need only be as long as the time the equipment is required to perform its safety functions, including, however, considerations of whether equipment failure after that time could adversely affect the safe shutdown condition or could mislead the operator.

If the equipment meets the guidance and requirements of the DOR Guidelines or 10CFR50.49 for a LOCA or HELB, and it can be shown that the failure of a piece of equipment (after it completes its safety function) will not adversely affect any safety related function or mislead the operator after exposure to a particular environmental parameter (e.g., flooding), the equipment could be considered exempt from that portion (i.e., submergence) of qualification.

2.5 QUALIFICATION OF EQUIPMENT

2.5.1 ACCEPTANCE CRITERIA

Electrical equipment was evaluated to ensure that it will function as required after exposure to its normal and postulated accident environments. All qualification conforms to the requirements of either the DOR Guidelines or 10CFR50.49 with guidance from Regulatory Guide 1.89.

2.5.1.1 Accident Environments

Each piece of equipment entered into the TMI-1 EQ Program was evaluated to determine if it would function as required during exposure to postulated accident conditions.

a. Operating Time

   All equipment in this program which is required to function in a harsh environment was qualified for the postulated postaccident duration as specified in the EQ file.

2.5.1.2 Margins

Equipment within the scope of the TMI-1 EQ Program was qualified to accident environmental profiles which enveloped the plant parameters discussed above. The conservatisms included in these profiles are judged to be sufficient to account for uncertainties associated with the analytical techniques, definitions of performance requirements, and variations in commercial production.

2.5.1.3 Connection Interfaces

Equipment which is required to be sealed when exposed to steam conditions coincident with significantly elevated pressure (greater than 1 psig) is provided with a seal. The only plant areas which meet these conditions are Containment and the Intermediate Building. Equipment in all other plant areas will be exposed to slight, if any, pressure increases during accident conditions and are therefore not required to be sealed.

2.5.1.4 Performance Specifications

Included within the TMI-1 EQ Program is an evaluation of equipment to ensure that performance specifications are achieved under conditions existing during and following postulated accidents. This evaluation includes a review of functional requirements, and/or loop accuracy based upon function, location, environment and performance requirement.

2.5.1.5 Voltage and Frequency

Safety related electrical equipment is subject to variations in power supply characteristics such as voltage and frequency. For the AC distribution system, these are comprised of the expected offsite power supply variations, including degraded grid conditions, and the expected variations of the diesel generator if offsite power has been lost. These conditions are addressed in the design of the plant by means of equipment specification and design, protective device settings, etc. While these power supply conditions may affect performance of equipment and must be considered, they are normally not linked with environmental effects. For example, a change in frequency will change the response time of a motor-operated valve, but this change in frequency will not be caused by environmental conditions. Reduced voltage will limit the torque that can be developed by a motor. If voltage or frequency transient must be considered in equipment design, they may be considered independently of environmental effects. For those cases where a relationship between power supply variations and environmental conditions are found, they will be addressed on an individual basis (Reference 8).

2.5.1.6 Synergistic Effects/Phase Changes

Present synergistic effect/phase change information is minimal and not conclusive. The equipment qualification effort did consider synergisms/phase changes as identified below.

a. If the vendor identified a synergistic effect/phase change, it was evaluated.

b. If the reviewer was aware of a synergistic effect/phase change, it was evaluated. As additional synergistic effect/phase change data became available it was evaluated and factored into the program.

c. If neither a. nor b. existed, then no further actions were taken to determine if any synergistic effect/phase changes were known (e.g., a literature research).

2.5.1.7 Field Verification

a. Walkdowns

A baseline field inspection of as-installed Class 1E equipment was performed by walkdowns. The field inspection established (a) a traceable link between the equipment installed at the plant and the equipment that was qualified, (b) a direct verification that any special installation requirements identified in the qualification program were applied, and (c) a verification that external gaskets, seals, protective covers, etc. have been installed.

2.5.2 ENVIRONMENTAL QUALIFICATION (EQ) FILES

2.5.2.1 Assembly of Documentation

The qualification documentation for electrical equipment was reviewed and accepted by GPUN and assembled into EQ Files. An EQ file is usually prepared for a group of components having the same manufacturer and model number, but different plant tag numbers. The EQ File contains all necessary reports, analyses, and correspondence submitted by the vendor to satisfy Purchase Order (P.O.) requirements, thereby establishing a direct link between plant installed equipment and qualification documentation.

For each model Number, the worst-case environment for that equipment was used to evaluate qualification. The Equipment I.D. Numbers and locations pertaining to each model number can be determined from the EQ Master List. Each EQ File was evaluated to ensure the completeness and accuracy of the data presented. The results of this review are documented in Environmental Qualification files based upon the requirements of the DOR Guidelines or 10CFR50.49 with guidance from Regulatory Guide 1.89.

3.0 REFERENCES

1. Margolis, S. G., and Redfield, J. A., FLASH - A program for Digital Simulation of the Loss-of-Coolant Accident, WAPD-TM-534, May 1966.

2. Grenda, R. J., and Patterson, J. F., "The Velocity of Rising Steam in a Bubbling Two-Phase Moisture," Transactions of the ANS, 5, No. 1, p 151, June 1962.

3. Richardson, L. C., Finnegan, L. J., Wagner, R. J, and Waage, J. M., "CONTEMPT," A Computer Program for Predicting the Containment Pressure-Temperature Response to a Loss-of-Coolant Accident, Phillips Petroleum Company, Atomic Energy Division, Idaho Falls, Idaho; AEC Research and Development Report TID-4500, issued June 1967.

4. C. M. Lederer, J. M. Hollander and J. Perlman, "Table of Isotopes," Sixth Edition, John Wiley and Sons, Inc., 1967.

5. A. Tobias, "Data for the Calculation of Gamma Radiation Spectra and Beta Heating from Fission Products (Rev. 3)," RD/B/M2669, CNDC(73)P4, Central Electricity Generating Board, Research Dept., Berkeley Nuclear Laboratories, United Kingdom, June (1973).

6. GPUN Calculation #C1101-823-5450-001, "TMI-1 LBLOCA EQ Temperature Profile Using the GOTHIC COMPUTER CODE", Rev. 9D, dated October 2009.

7. Deleted

8. NRC Generic Letter No. 89-10, "Safety Related Motor Operated Valve Testing & Surveillance - 10CFR50.54(f)", dated June 28, 1989.

9. GPUN Memo 5523-95-228 (CMT 172052) S.T. Plymale to M.C. House, "RB Instrumentation 281' Elevation," September 27, 1995.

10. GPUN Memo 5523-95-234 (CMT 172052) S.T. Plymale to M.C. House, "RB Instrumentation 281' Elevation," October 5, 1995.

11. AREVA Calculation 32-9082236, TMI Unit 1 Reactor Building Maximum Flood Level, Rev. 0.

12. AmerGen Calculation No. C-1101-424-E540-064, “Flooding Due to a Postulated Feedwater Pipe Break in the Intermediate Building”, Rev. 1, p. 4 of 41.

13. AmerGen Calculation No. C-1101-424-E540-064, Rev. 1, dated August 27, 1998.

14. AmerGen Calculation C1101-414-E540-006, Rev. 1, “Gothic Analysis of Auxiliary Building Environment from a Crack in the Auxiliary Steam Line.”

15. AmerGen Calculation C-1101-823-E610-014, Rev. 2A, “TMI-1 MSLB Containment Response.”

16. AREVA NP Inc. Document 51-9007389-002, "TMI-1 EOTSG Post-Accident Containment Environment Summary Report."

TABLE 6B-1
(Sheet 1 of 1)

Heat Transfer Coefficients Applied to the Reactor Coolant System and to the Reactor Building for LOCA Analysis

Item		Btu/h/ft^2-F
a.	Heat Transfer From Fuel Cladding to Water After Blowdown (Pool Boiling)	100
b.	Heat Transfer From RCS Metal During Blowdown	1,000
c.	Heat Transfer From RV Metal and Internals Below Nozzles After Blowdown	100
d.	Heat Transfer From RCS Metal, Other Than Item C. After Blowdown	5
e.	Heat Transfer from RB Atmosphere to Steel Heat Sinks Through Condensate Film	*620/40
f.	Heat Transfer From RB Atmosphere to Concrete Surfaces	40
g.	Heat Transfer From Sump Liquid to Floor	20

* Step change to 40 Btu/h-ft^2-F occurs when heat absorbed equals 110 Btu/ft^2.

References: (1) Proposed Standard for Design Pressure in Pressure Decay Requirements, American Standards Association, Jan. 30, 1962.

(2) Kolflat, A and Chittenden, W. A., A New Approach to Design of Containment Shells for Atomic Power Plants, Proceedings of American power Conference, Vol. XIX, 1957.

Table 6B-2
(Sh 1 of 1)

Engineered Safeguards Data

	Parameter	Value
1.	Number of Air Coolers	3
2.	Design Heat Removal Capacity per Cooler	80.0×10^6 Btu/hr
3.	Delay Time From Actuation for Fans	35 s
4.	Reactor Building Design Temperature	281F
5.	Number of Spray Systems	2
6.	Spray Flow Rate per System (nominal)	1,100 gpm
7.	Spray Water Inlet Temperature (before recirculation)	90F[1]
8.	Maximum Delay Time From Actuation for Sprays	160s

[1] The BWST temperature used in containment EQ Temperature Profile Analysis is 120°F.

Table 6B-3*
(Sh 1 of 1)

Summary of Reactor Building LOCA Pressure Analysis

Rupture Size $ft.^2$	Peak Pressure psig	Time to Reach Peak Pressure, s	Safeguards Operating
14.1	49.6	10 and 50	3 Air Coolers
8.5	50.6	50	3 Air Coolers
5.0	49.9	20	3 Air Coolers
3.0	48.9	35	3 Air Coolers
1.0	45.1	80	3 Air Coolers
8.5	50.5	50	2 Air Coolers and 1 1,500 gpm Spray

* This Table is retained for information only. The actual design basis accident is a 7.0 ft^2 rupture at RCP suction as described in paragraph 1.2. The 7.0 ft^2 rupture results in the highest peak pressure as shown in Table 6.6-1.

TABLE 6B-4
(Sh 1 of 1)

DELETED

TABLE 6B-5
(Sh. 1 of 1)

ISOTOPIC CORE INVENTORY

Isotope	Total Core Inventory In Curies		Isotope	Total Core Inventory In Curies	
Br 84	1.57	+E7	Xe 133	1.27	+E8
Br 85*	2.19	+E7	Xe 135m	3.26	+E7
Kr 83m	9.25	+E6	Xe 135	2.09	+E7
Kr 85m	2.19	+E7	Xe 138	1.17	+E8
Kr 85	5.30	+E5	I 129*	1.80	+E0
Kr 87	4.00	+E7	I 131	7.35	+E7
Kr 88	5.60	+E7	I 132	8.62	+E7
Rb 88	5.64	+E7	I 133	1.28	+E8
Sr 89	7.42	+E7	I 134	1.60	+E8
Sr 90	3.99	+E6	I 135	1.27	+E8
Sr 91	9.72	+E7	Cs 134	1.27	+E6
Sr 92	9.50	+E7	Cs 136	8.02	+E5
Y 90	3.96	+E6	Cs 137	4.99	+E6
Y 91	9.85	+E7	Cs 138	1.23	+F8
Mo 99	1.28	+E8	Ba 137m	4.67	+E6
Ru 106	2.29	+E7	Ba 140	1.25	+E8
Xe 131m	4.38	+E5	La 140	1.27	+E8
Xe 133m	3.07	+E6	Ce 144	7.50	+E7

* Deleted as insignificant for subsequent calculations.

TABLE 6B-6
(Sheet 1 of 6)

ASSUMPTIONS USED IN DEVELOPMENT OF LOCA TEMPERATURE PROFILE

OUTSIDE ATMOSPHERE CONDITIONS:

The air temperature is assumed to change in 24 hour cycles. For the first eight hours of the cycle it is 95ºF, the next 8 hours it is 85ºF and the last 8 hours of the cycle it is 75ºF. This 24 hour cycle repeats throughout the entire transient of approximately 500,000 seconds (5.79 days).

CONTAINMENT PARAMETERS:

The containment volume is 1,999,000 cubic feet.
The initial temperature in the Reactor Building is 130°F.
The initial humidity in the Reactor Building is 100 percent relative humidity.
The initial pressure in the reactor building is 14.7 psia.
The following five passive heat sinks should be modeled:

Heat Conductor	Material Type	Surface Area sq ft	Thickness	Description
1	Paint	81,700	15 mils	Reactor Building
	Steel	81,700	3/8 in	Walls and Dome
	Air Gap	81,700	1/32 in	
	Concrete	81,700	3.4 ft	
2	Stainless St.	6,000	1/4 in	Refuel. Canal
	Air Gap	6,000	1/32 in	Stainless Steel
	Concrete	6,000	3.75 mils	Liner or Inside
	Paint	6,000	15 mils	
3	Paint	106,100	15 mils	Misc. Internal
	Steel	106,100	0.268 in	Steel
4	Paint	117,800	15 mils	Misc. Internal
	Concrete	117,800	1.435 ft	Concrete
5	Paint	11,000	15 mils	Reactor Building
	Concrete	11,000	11.0 ft	Floor

TABLE 6B-6
(Sheet 2 of 6)

The five conductors are represented in one-dimensional slab geometry. The Table below summarizes the volumes exposed to these heat conductors and the types of boundary conditions used for each surface.

CONDUCTORS USED IN TMI-1 GOTHIC MODEL FOR LOCA PROFILE

CONDUCTOR	SURFACE A	BOUNDARY CONDITION A	SURFACE B	BOUNDARY CONDITION B
1	Containment Atmosphere	Uchida Condensation Heat Transfer Coefficient And Natural Convection	Outside Atmosphere	Specified Ambient Temp. with constant Heat Transfer Coefficient of 1 BTU/hr-ft^2
2	Containment Atmosphere	Uchida Condensation Heat Transfer Coefficient And Natural Convection	Containment Atmosphere	Uchida Condensation Heat Transfer Coefficient and Natural Convection
3	Containment Atmosphere	Uchida Condensation Heat Transfer Coefficient And Natural Convection	Insulated	Heat Flux = 0 BTU/hr-ft^2
4	Containment Atmosphere	Uchida Condensation Heat Transfer Coefficient And Natural Convection	Insulated	Heat Flux = 0 BTU/hr-ft^2
5	Sump	Uchida Condensation Heat Transfer Coefficient And Natural Convection	Constant Temperature	Temperature =85°F

TABLE 6B-6
(Sheet 3 of 6)

ASSUMPTIONS USED IN DEVELOPMENT OF LOCA TEMPERATURE PROFILE

BLOWDOWN DATA

The end of blowdown occurs at the end of 36 sec.

The following mass rate and enthalpy to the reactor building should be assumed for a 7.0 sq. ft. break at the pump suction:

Time Interval (second)	Average Mass Flow Rate (lbm/Sec)	Average Enthalpy (Btu/lbm)
0-2	57,300	558.333
2-4	53,350	566.382
4-6	47,035	583.019
6-8	34,195	617.780
8-10	22,187	689.503
10-12	12,904	765.916
12-14	4,768	1,086.838
14-16	4,630	735.501
16-18	5,605	520.250
18-20	5,416	457.903
20-22	2,893	419.165
24-28	889	412.658
28-32	14	298.246
32-36	38	337.748
36-40	0	0.0
40-44	48	333.333
44-48	0	0.0
48-56	79	265.263
56-62	1,427	416.472
62-68	656	1,073.895
68-74	1,477	604.153
74-80	2,688	461.519
80-90	2,479	477.207
90-100	1,959	538.497
100-110	1,131	692.002
110-120	561	888.632
120-140	157	1,133.524
140-160	54	1,180.801
160-180	50	1,182.093
180-200	48	1,148.691
200-240	323	431.353
240-280	673	474.770
280-320	556	438.506
320-360	340	344.001
360-400	413	322.518
400-440	540	316.633
440-480	391	300.288
480-520	383	280.439
520-560	359	275.384
560-600	345	273.663

TABLE 6B-6
(Sheet 4 of 6)

ASSUMPTIONS USED IN DEVELOPMENT OF LOCA TEMPERATURE PROFILE

DECAY HEAT

Following the end of the blowdown data at 600 seconds, decay heat is added to the RCS volume. (The decay heat is included in the blowdown data as is the HPI, LPI and Core Flood Tank inventory.) The RCS is modeled as an overflow volume. The ECCS will fill the vessel to the height of the cold leg where it then spills out of the break. The decay heat will vaporize some to the liquid in the RCS volume resulting in it going to the containment atmosphere out of the break. The ECCS flowing into the vessel is uniformly mixed with the water in the vessel causing it to absorb some of the decay heat before it overflows out of the break and collects in the sump.

The following decay heat values were used:

Time (seconds)	Decay Heat (Btu/Hr)	Time (seconds)	Decay Heat (Btu/Hr)
600.01	2.23686e8	43200.	6.09296e7
700.	2.14138e8	54000.	5.70460e7
800.	2.06197e8	64800.	5.43193e7
900.	1.99437e8	75600.	5.18206e7
1000.	1.93578e8	86400.	4.99072e7
2000.	1.59098e8	97200.	4.81010e7
3000.	1.41850e8	100800.	4.77619e7
4000.	1.30759e8	201600.	3.78446e7
5000.	1.22757e8	302400.	3.26253e7
6000.	1.16584e8	360000.	3.04692e7
7000.	1.11607e8	400000.	2.96000e7
8000.	1.07469e8	500000.	2.77886e7
9000.	1.03945e8	600000.	2.63911e7
10000.	1.00892e8	700000.	2.52646e7
10200.	8.87238e7	800000.	2.43276e7
20400.	7.38976e7	900000.	2.35301e7
30600.	6.65626e7	1000000.	2.28389e7

The decay heat values are based on GE Letter OG8-1081-45, “Decay Heat Data”, from RA Hill, Project Manager, to BWR Owners’ Group (BWROG) Risk Assessment Issues (RAI) Committee, dated November 8, 1988. The data in the aforementioned letter was used with a 1.2 multiplier from 600 seconds, the start of the decay heat in the model, to 10000 seconds. The multiplier was reduced to 1.0 for the time from 10000 seconds until the end of the available data at 360000 seconds. The ANS 1971 equation was used to generate decay heat values for the remainder of the transient.

TABLE 6B-6
(Sheet 5 of 6)

ASSUMPTIONS USED IN DEVELOPMENT OF LOCA TEMPERATURE PROFILE

EMERGENCY CORE COOLING SYSTEM

As a result of a single failure, only one low pressure injection pump with a capacity of 2400 gpm and one high pressure injections pump with a capacity of 500 gpm are available. Prior to 600 seconds, this LPI flow is accounted for in the blowdown data and not explicitly included in the GOTHIC model. The HPI flow is not included in the blowdown data although its effect on the BWST drawdown rate is accounted for in the GOTHIC model. After 600 seconds, (the end of the blowdown data), it is assumed that the HPI is secured and the LPI is injected at a rate of 2400 gpm. This LPI injection is modeled in the GOTHIC analysis.

TABLE 6B-6
(Sheet 6 of 6)

ASSUMPTIONS USED IN DEVELOPMENT OF LOCA TEMPERATURE PROFILE

The LPI injection with total flow of 2400 gpm is modeled to start at 600 seconds, taking suction from the BWST, flowing through the DHR heat exchanger, then into the RCS volume. The ECCS water is uniformly mixed with the RCS liquid and spilled out of the break and into the sump. The GOTHIC model in Reference 6 includes the BWST, and accounts for the drawdown rate of the BWST. When the BWST level reaches 7'4", the plant goes into the recirculation mode. The 7’4” BWST switchover setpoint accounts for instrument inaccuracy. In this mode, the ECCS takes suction from the sump, passes flow through the DHR heat exchanger, and into the RCS.

The decay heat removal heat exchanger is a two pass tube to shell type. The tubes are stainless steel and are 0.049 inches thick. The effective surface area is 90% of design 3350 ft^2 (3015 ft^2). The heat transfer coefficient is 300 BTU/hr-ft^2-°F. Decay Heat Closed Cycle Cooling Water (DHCCCW) is used in the secondary side. The flow rate of DHCCW is 3000 gpm to the DHR heat exchanger and 129 gpm to the remaining equipment heat loads for a total of 3129 gpm. The heat load of the equipment other than the DHR heat exchangers cooled by DHCCW is 271400 BTU/hr.

The Decay Heat Service Cooler has an effective surface area is 90% of design 25,220 ft^2 (22,698 ft^2) and an overall heat transfer coefficient of 224 BTU/hr-ft^2-°F. The flow rate of the river water through the coolers is 6000 gpm and its temperature is 95°F.

REACTOR BUILDING SPRAY SYSTEM

Assuming a single failure, there is only one building spray pump available. The building spray flow rate is 800 gpm and there is a conservatively assumed 160 second start delay after the 47.7 psia actuation signal. The BS acutation signal includes a 3 psi instrument error. Initially, the spray draws suction from the BWST which is evaluated at a high limiting water temperature of 120ºF. Following ECCS Suction Switchover, the RB Spray system draws from the containment sump, which may be significantly warmer.

FAN COOLERS

The Reactor Building Emergency Cooler heat removal rate is calculated using the GOTHIC fan cooler component models contained in the TMI basedeck for the current EQ analysis. The GOTHIC model incorporates a lower volumetric flow of 25000 cfm for each unit, a higher cooling water temperature of 95°F, a lower cooling water flow of 1450 gpm, and a 10% reduction in surface area. The GOTHIC fan cooler model calculates the fan cooler heat removal rate based on the calculated containment conditions (i.e. pressure, temperature, and humidity).

TABLE 6B-7
(Sheet 1 of 4)

ASSUMPTIONS USED IN DEVELOPMENT OF RB MSLB
TEMPERATURE PROFILE

OUTSIDE ATMOSPHERE CONDITIONS:

The air temperature is assumed to change in 24 hour cycles. For the first eight hours of the cycle it is 95°F, the next 8 hours it is 85°F and the last 8 hours of the cycle it is 75°F. This 24 hour cycle repeats throughout the entirety of the MSLB transient.

CONTAINMENT PARAMETERS:

The containment volume is 1,999,000 cubic feet.
The initial temperature in the Reactor Building is 130°F.
The initial humidity in the Reactor Building is 100 percent relative humidity.
The initial pressure in the reactor building is 14.7 psia.
The following five passive heat sinks are modeled:

Heat Conductor	Material Type	Surface Area sq ft	Thickness	Description
1	Paint	81,700	85 mils	Reactor Building
	Steel	81,700	3/8 in	Walls and Dome
	Air Gap	81,700	1/32 in	
	Concrete	81,700	3.4 ft	
2	Stainless St.	6,000	1/4 in	Refuel. Canal
	Air Gap	6,000	1/32 in	Stainless Steel
	Concrete	6,000	3.75 mils	Liner or Inside
	Paint	6,000	85 mils	
3	Paint	106,100	85 mils	Misc. Internal
	Steel	106,100	0.268 in	Steel
4	Paint	117,800	85 mils	Misc. Internal
	Concrete	117,800	1.435 ft	Concrete
5	Paint	11,000	85 mils	Reactor Building
	Concrete	11,000	11.0 ft	Floor

The Table below summarizes the volumes exposed to these heat conductors and the types of boundary conditions used for each surface.

TABLE 6B-7
(Sheet 2 of 4)

ASSUMPTIONS USED IN DEVELOPMENT OF RB MSLB
TEMPERATURE PROFILE

CONDUCTORS USED IN TMI-1 GOTHIC MODEL FOR RB MSLB PROFILE

CONDUCTOR	SURFACE A	BOUNDARY CONDITION A	SURFACE B	BOUNDARY CONDITION B
1	Containment Atmosphere	Uchida Condensation Heat Transfer Coefficient And Natural Convection	Outside Atmosphere	Specified Ambient temp. with constant Heat Transfer Coefficient of 1 BTU/hr-ft^2
2	Containment Atmosphere	Uchida Condensation Heat Transfer Coefficient And Natural Convection	Containment Atmosphere	Uchida Condensation Heat Transfer Coefficient and Natural Convection
3	Containment Atmosphere	Uchida Condensation Heat Transfer Coefficient And Natural Convection	Insulated	Heat Flux = 0 BTU/hr-ft^2
4	Containment Atmosphere	Uchida Condensation Heat Transfer Coefficient And Natural Convection	Insulated	Heat Flux = 0 BTU/hr-ft^2
5	Sump	Uchida Condensation Heat Transfer Coefficient And Natural Convection	Constant Temperature	Temperature =85°F

TABLE 6B-7
(Sheet 3 of 4)

ASSUMPTIONS USED IN DEVELOPMENT OF RB MSLB TEMPERATURE PROFILE

BLOWDOWN DATA

The end of blowdown occurs after 600 seconds.

The following table samples mass flow rates and enthalpies at intervals throughout MSLB blowdown, based on a break at the OTSG:

Time (sec)	Mass Flowrate (lbm/sec)	Average Enthalpy (Btu/lbm)
0	0	1248
0.1	5269	1205
0.5	4446	1254
1	4156	1253
4	3925	1236
4.4	4100	1232
5	3890	1231
8	3515	1230
9	3180	1218
10	1880	1216
15	1738	1221
20	1696	1214
25	1538	1215
30	1489	1215
35	1416	1216
40	1356	1215
45	1282	1216
50	1183	1217
60	1000	1218
70	866	1217
80	773	1213
90	649	1213
100	553	1215
130	133	1256
140	120	1250
145	119	1261
150	119	1244
170	89	1270
200	89	1247
225	88	1261
250	88	1261
275	87	1264
300	88	1250
350	88	1257
400	88	1251
450	87	1253
500	87	1249
550	86	1251
600	86	1246

TABLE 6B-7
(Sheet 3 of 4)

ASSUMPTIONS USED IN DEVELOPMENT OF RB MSLB
TEMPERATURE PROFILE

Time (sec)	Total Mass Release (lbm)	Total Energy Release (BTU)	Enthalpy (BTU/lbm)	Mass Flow Rate (lbm/s)
2.0	8.59150E+03	1.06849E+07	1239.44	4095.00
4.0	1.66030E+04	2.05988E+07	1234.32	4050.00
6.0	2.43900E+04	3.01864E+07	1229.61	3715.00
8.0	3.16200E+04	3.90778E+07	1229.18	3495.00
10.0	3.71760E+04	4.58649E+07	1213.21	1832.00
12.0	4.07140E+04	5.01637E+07	1221.95	1704.00
14.0	4.41350E+04	5.43480E+07	1221.36	1732.00
16.0	4.76140E+04	5.85928E+07	1217.89	1744.00
18.0	5.11000E+04	6.28303E+07	1214.55	1732.00
20.0	5.45310E+04	6.69966E+07	1214.53	1666.00
24.0	6.09480E+04	7.47931 E+07	1215.28	1538.00
28.0	6.70370E+04	8.21930E+07	1215.94	1499.00
32.0	7.29610E+04	8.93938E+07	1215.21	1446.00
36.0	7.86580E+04	9.63193E+07	1215.66	1392.00
40.0	8.41500E+04	1.02996E+08	1215.77	1344.00
48.0	9.44670E+04	1.15540E+08	1215.95	1204.00
56.0	1.03564E+05	1.26604E+08	1217.14	1027.00
64.0	1.11532E+05	1.36315E+08	1219.67	956.00
72.0	1.18743E+05	1.45099E+08	1216.18	828.00
80.0	1.25119E+05	1.52843E+08	1214.38	765.06
90.0	1.32231 E+05	1.61476E+08	1214.85	633.00
100.0	1.38129E+05	1.68639E+08	1216.01	537.00
110.0	1.43089E+05	1.74689E+08	1220.66	426.00
120.0	1.46394E+05	1.78744E+08	1239.82	221.00
160.0	1.51602E+05	1.85250E+08	1270.83	96.00
200.0	1.55177E+05	1.89747E+08	1261.36	88.00
240.0	1.5B699E+05	1.94177E+08	1250.00	88.00
280.0	1.62219E+05	1.98598E+08	1250.00	88.00
320.0	1.65734E+05	2.03007E+08	1257.14	87.50
360.0	1.69244E+05	2.07405E+08	1251.43	87.50
400.0	1.72750E+05	2.11793E+08	1245.71	87.50
450.0	1.771 02E+05	2.17232E+08	1245.71	87.50
500.0	1.81434E+05	2.22638E+08	1248.55	86.50
550.0	1.85738E+05	2.28003E+08	1244.19	86.00
600.0	1.90025E+05	2.33339E+08	1244.44	67.50

TABLE 6B-7
(Sheet 4 of 4)

ASSUMPTIONS USED IN DEVELOPMENT OF RB MSLB
TEMPERATURE PROFILE

NSSS MODELING

The only interaction between the MSLB model and the NSSS is the mass and energy release from the MSLB and the injection of HPI. The mass and energy release is modeled using boundary conditions. HPI is removed from the BWST and is not injected into another volume. Therefore, the NSSS is not included in the RB MSLB model.

REACTOR BUILDING SPRAY SYSTEM

The RB MSLB does not actuate Building Spray because the MSLB accident does not cause enough containment pressure. So, the RB spray does not contribute to containment cooling.

FAN COOLERS

The Reactor Building Emergency Cooler heat removal rate is calculated using the GOTHIC fan cooler component models contained in the TMI basedeck for the RB MSLB. The GOTHIC model incorporates the following conservative assumptions:

- Volumetric flow of 25000 cfm for each unit
- Cooling water temperature of 95^{o} F
- Cooling water flow of 1450 gpm
- 10% reduction in cooler surface area

The GOTHIC fan cooler model calculates the fan cooler heat removal rate based on the calculated containment conditions (i.e., pressure, temperature, and humidity).