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April 16, 1985

TMI Program Office
Attn: Dr. B. J. Snyder
Program Director
US Nuclear Regulatory Commission
Washington, DC 20555

Dear Dr. Snyder:

Three Mile Island Nuclear Station, Unit 2 (TMI-2)
Operating License No. DPR-73
Docket No. 50-320
GPU Nuclear Corporation Seismic Design Criteria

By your letter dated November 28, 1984, you confirmed NRC Staff concurrence "that temporary recovery modifications do not have to meet design basis severe natural phenomena so long as; 1) the structure is temporary (all recovery modifications are not necessarily temporary in the staff's view); 2) a breach of that component by natural phenomena will not cause a radiological release in excess of 10 CFR 100 limits or the failure of that component will not compromise the ability to maintain the reactor in a safe shutdown condition".

In response to your letter, GPU Nuclear undertook the attached analysis for the purpose of demonstrating that those TMI-2 systems installed since the 1979 accident or contemplated for future installation for the sole purpose of supporting TMI-2 recovery activities are not required to meet seismic design requirements. The analysis shows that failure of any of these structures, systems, or components as a result of a seismic event will not result in a radiological release in excess of a small fraction of the guideline values in 10 CFR Part 100 and the failure will not comprise the ability to maintain the reactor in its current safe shutdown condition.

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The above referenced letter also states that "staff concurrence with your design does not relieve you from the necessity of requesting exemption from the code when appropriate". GPU Nuclear has reviewed 10 CFR Part 50, Appendix A, General Design Criterion 2, "Design Basis Protection Against Natural Phenomena" and 10 CFR Part 100 Appendix A, "Seismic and Geological Siting Criteria for Nuclear Power Plants". Since the consequences of failure of post-accident structures, systems and components are bounded by the guidelines of Appendix A, 10 CFR Part 100, it is our understanding that existing designs are adequate.

General Design Criterion 2 states that "structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety function."

10 CFR Part 100, Appendix A, "Seismic and Geological Siting Criteria for Nuclear Power Plants", states that its purpose is to set forth the principal seismic considerations which guide the Commission in its evaluation of the suitability of the plant design bases established in consideration of the seismic and geologic characteristics. The structures, systems, and components which are designed to remain functional for the "Safe Shutdown Earthquake" are those necessary to assure:

1. The integrity of the reactor coolant pressure boundary.

2. The capability to shutdown the reactor and maintain it in a safe shutdown condition, or

3. The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the guideline exposures of this part.

Because of the unique shutdown condition of TMI-2, none of the post-accident systems are required to perform a nuclear safety-related function, as defined above, during a seismic event. Specifically, the first of the criteria clearly is not applicable to TMI-2 in its current condition. Compliance with the second criterion is attested to by the enclosed analysis which concludes that the failure of any post-accident system as a result of a design basis earthquake will not jeopardize the current safe shutdown condition of the TMI-2 reactor. Finally, the enclosed analytical data affirms compliance with the third criterion by substantiating that no post-accident system is required to prevent offsite exposures which would be comparable to 10 CFR Part 100 nor would system-related failures result in such consequences.

Because it has been shown by analysis that post-accident systems do not perform a nuclear safety-related function, as defined by 10 CFR 100, Appendix A, it has been concluded that these systems need not be designed to withstand the effects of a design basis earthquake. It follows, therefore, that 10 CFR Part 50, Appendix A, General Design Criterion 2, as it relates to seismic events, does not apply to TMI-2 post-accident structures, systems, and components and an exemption is not required.

Subsequent to your review of this letter and the attached analysis, please provide comment relative to the conclusions contained herein and the applicability of seismic design requirements to TMI-2 post-accident systems.

Per the requirements of 10 CFR 170, an application fee of $150.00 is enclosed for review of this document.

Sincerely,

for F. R. Standerfer
Vice President/Director, TMI-2

FRS/EDF/eml

Attachment

cc: Deputy Program Director - TMI Program Office, Dr. W. D. Travers

Enclosed: GPU Nuclear Check No. 15454

GPU Nuclear

SAFETY ANALYSIS

SA # 4430-7322-85-1

Rev. # 0

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TITLE

SAFETY EVALUATION JUSTIFYING THE NON-SEISMIC

DESIGN OF TMI-2 "POST-ACCIDENT" SYSTEMS

Originator J. J. Curry Date 4/11/85

CONCURRENCE

RTR Date 4/2/85 SRG Date 4/2/85

Eng. James J. Curry Date 4/2/85 Rad Con Date 4/2/85

Eng. K J Hofstetter 4-9-85

APPROVAL

Mgr. Eng. Section R P Warren Date 4/10/85 Site Ops Director Date 4/2/85

Mgr. Recovery Pgms. Date 4/2/85

FORM 4000-ENG-7310.09-1 (5/83)

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Safety Evaluation Justifying the Non-Seismic Design of TMI-2 "Post-Accident" Systems

1.0 Purpose and Scope

The purpose of this safety evaluation is to provide justification for the non-seismic design of systems that have been installed since the 1979 accident. The scope of this evaluation covers all "post-accident" systems that are currently operational or are planned to be operational to support fuel removal from the reactor vessel. The evaluation demonstrates that the public consequences of radionuclide releases that could occur from system failures due to a design basis earthquake (DBE) are less than the limits of applicable NRC standards. Section 2.0 describes the analytical approach used in the evaluation. Section 3.0 describes the postulated accidents that could result from failure of non-seismic components and their potential offsite consequences. Section 4.0 provides the conclusions of the analysis. Section 5.0 provides an "unreviewed safety question" evaluation as required by 10 CFR 50.59.

2.0 Analytical Approach

The public consequences of an environmental release due to planned defueling activities through "early defueling" have been analyzed (Reference 1). That analysis considered a spectrum of accidents and found that radionuclide releases did not exceed 10CFR Part 100 limitations. Other evaluations have considered the potential consequences of postulated accidents with an open reactor building equipment hatch (Reference 2) or from postulated accidents in other areas of the plant. In general, the analyses of "post-accident" systems and processes have not considered a seismic event as a potential cause of an accident. This evaluation considers the consequences of accidents which may be postulated due to a seismic event. The evaluation was conducted in several phases, as described below.

Phase 1: Identification of Source Terms and Driving Forces

For an accident to result in consequences that exceed those previously analyzed, a larger radionuclide source term and/or a greater driving force to move the radionuclides to the environment must be generated. To identify possible source terms and driving forces a logic diagram was developed and is presented in Section 3.1.

Phase 2: Specification of Seismic Induced Accident Sequences

Each driving force was coupled with each radionuclide source term identified in the first phase of the analysis to define a potential accident sequence resulting from a seismic event. This resulted in approximately 100 combinations of driving forces with source terms. Each combination was examined and those that were physically unrealistic or which could not credibly occur were eliminated from further analysis. Consideration was made as to whether the driving force could be directly associated with the radionuclide source (e.g. a fire in a waste storage area) or whether the driving force simply occurred simultaneously with an increased source term (e.g. an atmospheric pressure differential with an SDS liquid line break). In some cases, such as draining the reactor vessel, the only mechanism for the accident sequence is the failure of a seismically qualified component, e.g., the incore instrument guide tubes. Normally, failure of a seismically qualified component would not be postulated for the design basis earthquake; however, there is the possibility that non-seismically qualified recovery components may have been constructed above seismic category I components since the accident. To take this into account, the failure of a seismically qualified component is postulated unless it has been shown that an "unqualified over qualified" failure is not credible.

The accident sequences are presented in Section 3.2.

Phase 3: Consequence Analysis

An offsite dose assessment was performed for radionuclide releases associated with potential seismically induced accidents. The dose was assessed at the nearest site boundary for two hours as specified in 10CFR Part 100. To conservatively estimate the offsite dose, the 0-1 hour 5th percentile atmospheric dispersion factor (χ/Q) given in Appendix 2D of the Final Safety Analysis Report (Reference 3) was used. The dose conversion factors for radionuclides specified in Regulatory Guide 1.109 (Reference 4) were used; for radionuclides not specified in RG 1.109, NUREG-0172 (Reference 5) factors were used.

The dose limitations in 10CFR Part 100 apply to the whole body and to the thyroid. The dose calculations performed in Reference 1 demonstrate that, for the TMI-2 radionuclide inventory, whole body and thyroid doses are small in comparison to the bone dose. Therefore, doses were calculated for the adolescent bone which is the critical organ for potential TMI-2 releases. To assess the significance of the calculated bone doses, insights from the literature were used.

The assessment considered continuous or "2 hour" releases and instantaneous or "puff" releases as appropriate for a particular sequence. Because GPUN has requested to remove the equipment hatch for limited time periods, and because some building ventilation systems and post-accident structures are not seismically qualified, all releases were assumed to be unfiltered. The fraction of material that may become airborne under the various postulated accident conditions was based on work performed by Sutter et al (References 7, 9, 10) and Chan and Mishima (Reference 8).

3.0 Accident Evaluation

The following subsections identify potential driving forces and accident source terms, determine accident sequences that could result from a seismic event and assess accident consequences.

3.1 Potential Source Terms and Driving Forces

To identify the potential source terms and driving forces that could be associated with a seismic event, the logic diagram shown as Figure 3.1 was developed. The "top event" of the diagram is "Seismically Induced Consequences". This event occurs if both a driving force and a radionuclide source term exist. For the seismically induced accident consequences to have the potential to exceed consequences for previously analyzed conditions, the driving force and/or the source term must be increased over those previously analyzed.

The purpose of Figure 3.1 is to identify source terms and accident sequences that are conceptually possible. The credibility and significance of the events in Figure 3.1 are examined in more detail in Sections 3.1.1 and 3.1.2.

3.1.1 Potential Driving Forces

Potential driving forces that could develop from a seismic event or could credibly occur coincidentally with the event are discussed below.

3.1.1.1 Atmospheric ΔP/Air Exchange (open equipment hatch/open building)

An air pressure differential between the reactor building (RB) and the environment of 1 psig, corresponding to a low pressure weather front, is assumed. This results in a very rapid equilibration of reactor building and outside pressures if the equipment hatch is open. The equilibration results in a loss of about seven percent of reactor building air volume. After this equilibration, the RB and environment are assumed to continue to exchange air at the rate of 2 containment volumes per hour which corresponds to an open equipment hatch with an outside wind of about 20 mph. An air exchange rate of 2 building volumes per hour was also assumed for AFHB releases.

3.1.1.2 Fire ΔP - Combustibles

Combustibles ignite because of shorting of non-seismically qualified electrical components or other unspecified ignition source. The analysis provided in Reference 11 provides a basis for determining the significance of a combustible materials fire driving force. That analysis indicated that a peak pressure of 2 psig might be reached in an enclosed environment using conservative heat transfer assumptions and a large combustion source. The air exchange rate with the outside environment that would be due to a similar fire in an open containment would be less than the exchange rate used in the previous section. Thus, the consequences of postulated accident sequences with the atmospheric driving force in Section 3.1.1.1 would bound those with a combustible fire driving force, given the same radionuclide source. (This assumes that the radionuclide source is not directly involved in the fire; that possibility was considered on a case-by-case basis as indicated in Table 3.1).

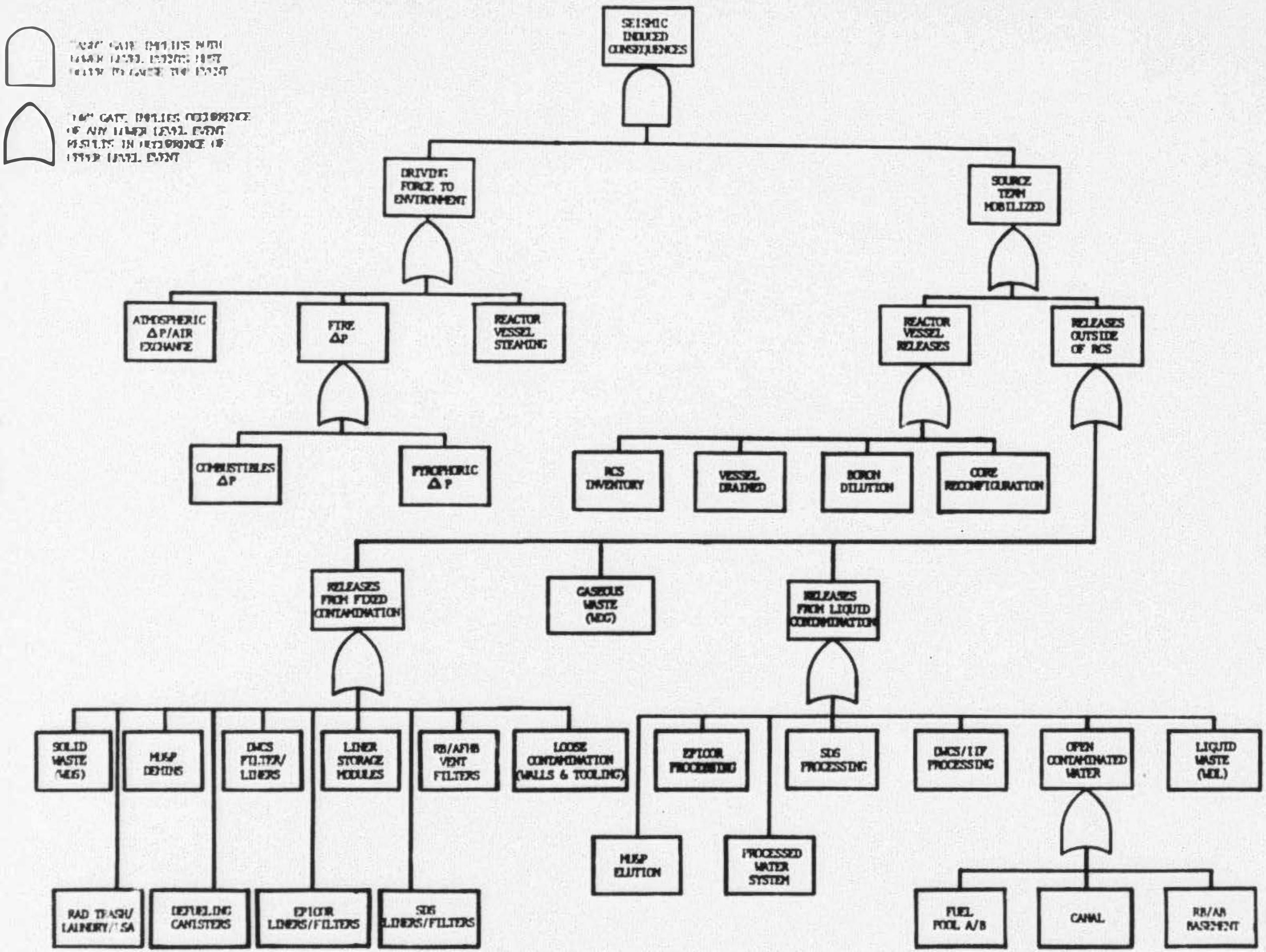

FIGURE 3.1 - POTENTIAL SOURCE TERMS AND DRIVING FORCES

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3.1.1.3 Fire 3P - Pyrophoric

Fire resulting from a pyrophoric reaction has been postulated to result from metal fines that may be associated with the core debris or that may have settled on reactor vessel internals. There are three materials which may now be present in the core and which could undergo a pyrophoric reaction: a uranium-zirconium eutectoid (which is surrounded by an oxygen-stabilized zirconium), zirconium metal and zirconium hydride. Only three occurrences can be postulated that could expose these core materials: (i) uncovering the core by leakage through the guide tubes, (ii) core materials themselves leaking through failed guide tubes or (iii) exposing materials in a defueling canister due to a handling accident.

Considerable analyses have been conducted since the pyrophoric concern was initially raised and are summarized in Reference 12. The analyses indicate that three conditions must exist to initiate and maintain a pyrophoric reaction:

(1) The pyrophoric material must have a high surface to volume ratio of the nature of powder. Experience indicates that moist zirconium fines of less than 10 μm will burn. However, core debris sample analysis indicates only a small fraction (Reference 31 indicated 1.5 percent) of the particulate matter is less than 45 μm. The planned defueling activities are not likely to generate significant additional quantities of fines in the size range of concern.

(2) The pyrophoric material must exist in an oxygen depleted environment and then be suddenly exposed to oxygen. The surface of the core pyrophoric material has been exposed to oxygen in the water since the accident. Thus, oxidation that has already occurred would limit a pyrophoric reaction to material that is freshly exposed. The defueling process is not likely to expose significant quantities of debris in the size range specified in item (1) above. Further, any additional exposure of pyrophoric material due to defueling activities would initially be underwater, where oxidation would again occur at some rate.

(3) The oxidation rate must exceed the heat transfer rate to the surrounding environment. The oxidized debris that will be mixed with any pyrophoric material acts as a diluent and minimizes the potential for ignition and propagation.

In addition to the above considerations, tests have been conducted on a sample of material that was removed from the plenum surface to determine its pyrophoricity (Reference 13). Attempts were made to "pilot ignite" the subject material by conducting a spark test, a strike test, and a flame test. The results found "no pyrophoric characteristic" for the material tested.

In summary, theoretical analysis and experimental data indicate that the characteristics of the material currently in the reactor vessel, or as it may be modified during defueling, is exceedingly unlikely to sustain a pyrophoric reaction. This conclusion is not dependent on continued submergence of the material in water or on a particular postulated seismically induced accident sequence. Thus, it was not considered reasonable to postulate a pyrophoric reaction of exposed fuel debris as a significant driving force for radionuclide transport.

3.1.1.4 RCS Steaming

The reactor coolant system is currently at about 82°F. The only mechanism that could produce steaming would be recriticality or a vessel draindown. Recriticality could be postulated by a core reconfiguration or a boron dilution event. A vessel draindown could occur by leakage through the incore instrumentation guide tubes.

Core reconfiguration

A core reconfiguration can be postulated to result from a seismic event, either due to direct mechanical vibration or due to the collapse into the vessel of seismically unqualified components. To evaluate the potential for criticality due to a seismically induced reconfiguration, the following considerations were made:

(1) The internals indexing fixture (IIF), defueling work platform and, at times, the defueling bridge assembly will be over the open reactor vessel. The IIF is essentially a hollow steel cylinder with a diameter exceeding the vessel diameter. The work platform consists of 17 major structural members, mostly 304 stainless steel, which are bolted together and weigh approximately 34,000 lbs. There are 113,000 pounds of shielding pieces which overlap each other and are supported by the major structural members. The beam end to beam end dimension exceeds the vessel flange opening by about four feet. The auxiliary fuel handling bridge was originally designed as seismic Category I and supports the defueling trolley and canister transfer shield. The auxiliary fuel handling bridge assembly and work platform have been classified "Important to Safety" for structural integrity. No seismic qualification of these components has been demonstrated. However, the design criteria that have been employed and the physical characteristics of the structures provide some measure of assurance that a major structural collapse into the vessel will not occur as a result of a DBE.

(2) An extensive analysis (Reference 14) has been conducted to consider the criticality potential of reconfigurations that could result from defueling operations. The approach used in the analysis was to specify an "infinite poison", i.e., a concentration of soluble boron that would assure subcriticality for all credible reconfigurations. A range of very reactive configurations was analyzed and the boron concentrations required to assure subcriticality were identified. The most reactive of these configurations, which is viewed to bound any credible core reconfiguration, is a concentric lenticular shape with 100% of the batch 3 (2.96% enriched) fuel surrounded by a uniform mixture of 100% of the batch 1 and 2 fuel. Standard fuel pellet size in a conservative geometry with a corresponding optimized fuel/water ratio was assumed with no credit for cladding or solid poison material. The probability of the core being

reconfigured in this manner, even given the collapse into the vessel of non-seismically qualified equipment, is exceedingly small. The referenced analysis indicated that the bounding case core configuration would be subcritical with a boron concentration of 4350 ppm.

(3) The defueling structures above the core are composed largely of stainless steel which is not considered to be a moderating material. Stainless steel may be a slightly better reflector than highly borated water. However, it is very unlikely that the collapse of this material into the core could result in a more reactive configuration than the bounding analysis in Reference 14.

(4) The RCS is maintained at a nominal 5050 ppm (± 100 ppm) to provide an operating margin above 4350 ppm. (Recent samples indicate that the RCS is actually above 5300 ppm.) Thus, an additional measure of criticality safety is provided.

(5) To assure that the core reactivity does not significantly increase by the inadvertent introduction of materials into the RCS, a list of materials which could adversely affect the shutdown margin is being developed. The allowable quantity of materials over the vessel will be defined and procedural controls will be developed to ensure that the limitations on material type and quantity are not violated (Reference 1). Thus, the potential for criticality due to the collapse into the vessel of materials that may be on the work platform is minimized.

Based on the above considerations, criticality due to a seismically induced core reconfiguration is not considered credible. Thus, a reconfiguration will not result in steaming of the RCS inventory.

Boron dilution

A dilution of the RCS boron concentration could be postulated if the seismic event caused the failure of RCS isolation barriers. A detailed examination of potential dilution paths has been performed and at least double isolation barriers are in place for all dilution paths during planned operating conditions (Reference 15). To evaluate the potential for a seismically induced RCS boron dilution, the following considerations were made:

(1) An examination of the isolation barriers and piping required to isolate the RCS during static conditions reveals that only a few dilution paths are not isolated by at least one seismic category I component; the dilution paths that pass through non-seismic isolation are generally associated with chemical addition lines. Thus, the large majority of potential isolations barriers will maintain their integrity during the design basis earthquake.

(2) Dilution through the secondary side is prevented by double isolation of potential dilution sources to the steam generators (Reference 15) and the maintenance of the steam generator water level below the primary side water level.

(3) The major potential sources of unborated water inflow through an open vessel head (assuming collapse of non-seismically qualified IIF/defueling structures) are decontamination activities and the fire service system. Decontamination activities are routinely conducted in the reactor building. Decon water may originate from the PWST, tankage in the chemical cleaning building and other smaller sources. It enters the containment through a penetration that can be isolated by outboard valving. Typical decon flow rates are 25 gpm. A boron dilution to 4350 ppm would require that decon water be directed to the open head and left in place for more than four hours. Thus, dilution of the RCS as a result of decon activities during a seismic event was not considered a credible event.

The fire service system piping is seismic Category I in the reactor building (Reference 3). The system enters the reactor building through a penetration from the fuel handling building (FHB) and is routed to several elevations of the reactor building. There are only two fire service stations at an elevation above the operating deck; these fire stations are above each 'D' ring at approximately el. 371'. The fire service penetration from the FHB enters below the operating deck; containment isolation is normally achieved (unless "hot work" is being performed in containment) by a closed valve in the FHB which is also on seismic Category I piping. Each station has a hose shutoff valve and a nozzle valve. Because of the seismic qualification of the fire service piping, the only postulated failure mechanism for the system that would release unborated water in a design basis earthquake is an "unqualified over qualified" failure. A detailed analysis of this possibility has not been conducted; however, the only major systems above the fire stations at elevation 371' are the HVAC system and polar crane. Both systems were originally seismic Category I and have not been significantly modified in terms of seismic qualification. Therefore, because of the seismic qualification of the fire service system, the seismic qualification of the elevated HVAC system and polar crane, and the ability to isolate the fire service system from outside of the reactor building, dilution of the RCS from a failure in the fire service system was not considered credible.

(4) Prior to the accident, the spent fuel pool gate area contained a weir which regulated water levels in fuel pools A and B. After the accident, a reinforced stainless steel closure plate was welded to the fuel pool B liner, preventing communication between pools A and B. Because the weld is on a vertical surface, the probability of its failure due to the collapse of a seismically unqualified component is minimized. However, because the seismic qualification of the weld was not demonstrated at the time the closure plate was constructed, weld leakage could be postulated during a seismic event. Such leakage could result in dilution of fuel pool A or the refueling canal if the transfer tubes were open and could not be closed. The probability that the closure plate will fail, given that it is reinforced and welded onto a seismic Category I structure, is considered small. Further, additional failures (e.g., crushed canisters, failed fuel transfer system valves) would be required to cause a radionuclide release. An analysis is being conducted to ascertain whether the closure plate will withstand seismic loads. If analysis indicates that the existing barrier between fuel pools A and B will not

withstand a design basis earthquake, appropriate plant modifications will be made to assure isolation of fuel pool B during a DBE. Thus, dilution of the RCS, or fuel pool A, due to leakage from fuel pool B was not postulated.

(5) The relative strengths of non-seismic piping versus the type of isolation barriers that are used to prevent a dilution (e.g., globe, gate valves versus check valves) indicate that it is very unlikely that a double isolation barrier will fail in a manner that will permit inleakage and still leave the non-seismic pipe intact.

(6) A detailed analysis to determine whether seismically unqualified components have been constructed over any RCS isolation barriers has not been conducted. However, the failure of unqualified components is much more likely to cause pipe leakage than to fail double isolation barriers in a manner that would permit a dilution inflow to occur.

(7) There are several seismically qualified makeup paths from the BWST which could provide borated water addition if required to mitigate a dilution event. It is very unlikely that all of these paths will be failed due to the effects of "unqualified over qualified" components.

Based on the above considerations, a significant boron dilution of the RCS due to a seismic event is not considered credible. Thus, recriticality and subsequent RCS steaming cannot be credibly postulated.

Vessel draindown

A thermal hydraulic and criticality analysis of a complete vessel draindown and subsequent refill has been performed (References 16, 17). The analysis used conservative assumptions, for example, (i) an initial RCS temperature of 120°F for draindown, (ii) initial decay heat of 20 kw, (iii) ambient containment temperature of 100°F, (iv) conservative heat transfer coefficients, (v) core reflood temperatures of 180°F, and (vi) a conservative draindown rate which maximizes steaming.

The analysis demonstrated that criticality would not occur during either draindown or reflood. The maximum quantity of steam released during draindown was estimated as less than seven pounds. Thus, reactor vessel steaming during a potential draindown would not be significant driving force for radionuclide movement to the environment.

3.1.2 Potential Source Terms

Potential radionuclide source terms that could develop from a seismic event are described in this section.

3.1.2.1 Reactor Vessel Drained (Core Inventory)

Draining of the reactor vessel would expose the core to the containment atmosphere. The source term for an exposed core was developed by identifying a detailed radionuclide inventory from an ORIGEN computer calculation (Reference 18) assuming decay to August 1984 and accounting for the known radionuclide releases to date (Reference 19). Based on the ORIGEN calculation, the following radionuclides are present in significant quantities and represent the source term used in the analysis:

ISOTOPE	QUANTITY (Ci)	ISOTOPE	QUANTITY (Ci)
H-3	5.9 E+2	Ce-144	2.0 E+5
Co-60	4.7 E+4	U-234	1.2 E+2
Kr-85	2.7 E+4	U-235	4.0 E+0
Sr-90	6.1 E+5	U-238	2.7 E+1
Ru-106	8.1 E+4	Np-237	1.1 E+0
Ag-110m	1.0 E+1	Pu-238	7.9 E+2
Sb-125	3.2 E+4	Pu-239	9.0 E+3
Cs-134	1.6 E+4	Pu-240	2.4 E+3
Cs-137	4.2 E+5	Pu-241	1.2 E+5
Pr-144	2.0 E+5	Am-241	1.2 E+3

Accident sequences involving a drained core could result in airborne particulate material (from exposed core) or an aerosol release (from draining liquid). The size distribution of core particulates has been characterized (Reference 31) and provides the basis for determining the airborne release fractions used in Section 3.2.

3.1.2.2 Boron Dilution

A boron dilution event would not significantly increase the core radionuclide inventory unless it resulted in criticality. From Section 3.1.1.4, a boron dilution due to a seismic event which causes criticality is not credible. Therefore, accident sequences with this source term are not analyzed further.

3.1.2.3 Core Reconfiguration

A core reconfiguration would not significantly increase the core inventory unless it resulted in criticality. From Section 3.1.1.4, criticality due to a reconfiguration from a seismic event is not credible. Therefore, accident sequences with this source are not analyzed further.

3.1.2.4 RCS Inventory

The current radionuclide concentration in RCS water is predominantly due to cesium and strontium. Experience with previous operations within the RCS shows that minor disturbances can result in increased concentrations of a select number of isotopes. Reference 20 provides a typical source term for RCS water that results from disturbances within the RCS. To assure that this analysis will bound RCS conditions throughout defueling, the radionuclide concentrations in Reference 20 were increased by a factor of five. The resulting source term is shown below.

ISOTOPE	RCS CONCENTRATION (μCi/ml)	ISOTOPE	RCS CONCENTRATION (μCi/ml)
H-3	1.8 E-1	U-234	5.5 E-5
Co-60	8.5 E-1	U-235	1.9 E-6
Sr-90	5.0 E+1	U-238	1.2 E-5
Ru-106	1.6 E+0	Np-237	5.5 E-7
Ag-110m	7.5 E-2	Np-239	3.5 E-8
Sb-125	2.5 E+0	Pu-238	2.4 E-5
Cs-134	1.2 E+0	Pu-239	4.2 E-3
Cs-137	2.5 E+1	Pu-240	1.1 E-3
Ce-144	5.5 E+0	Pu-241	7.0 E-2
		Am-241	7.0 E-4

3.1.2.5 Smearable Contamination (Reactor Building walls, tooling)

An estimate for the contamination that could be available for transport is based on the smearable contamination in the reactor building, which is concentrated primarily in the reactor building basement. From Reference 21, there are approximately 25,000 Ci of basement activity, neglecting deep penetration and basement water activity. Assuming 2% (Reference 28) is smearable and the isotopic distribution is similar to that at other elevations yields the following source term:

Sr-90	8.4 E+0 Ci
Sb-125	1.5 E+0 Ci
I-129	3.2 E-4 Ci
Cs-134	4.5 E-1 Ci
Cs-137	4.5 E+2 Ci

The smearable activity on other RB surfaces and in other buildings is small in comparison to this source.

3.1.2.6 Defueling Canisters

Before loading into the shipping cask, there are three locations outside of the reactor vessel for the defueling canisters: (1) in the RB or FHB transfer shield (2) in the deep end of the refueling canal and (3) in fuel pool A. A single defueling canister is transported in a transfer shield and contains, on average, approximately 1/250 of the total core material. Current plans are that the deep end of the refueling canal will normally contain no more than seven defueling canisters at any time (including four filter canisters for DWCS operation and up to three defueling canisters). However, the source term in the deep end of the refueling canal assumes that all eleven available spaces are occupied by canisters. There are 252 spaces for canisters in fuel pool A.

There are two potential source terms associated with a defueling canister: (1) the source term due to the existing fuel and fission product inventory and (2) an increased source term due to criticality.

The potential for an increased source term due to a criticality was evaluated by considering the following:

(1) Each defueling canister includes enough poison material to assure that no single canister or array of canisters, as they will be handled and stored during defueling will have a k_{eff} greater than 0.95. The analysis demonstrating compliance with the k_{eff} criterion used numerous conservative assumptions including: (i) batch 3 fuel only, (ii) no soluble poison or control material, (iii) optimal volume fraction and moderator density, and (iv) complete filling of canister fuel regions (Reference 22).

(2) With the exception of a canister that may be in transport in the transfer shield, all other canisters are submerged in water with a minimum boron concentration of 4350 ppm. The canister storage racks have been designed to withstand the loadings associated with the DBE (Reference 27); thus, direct damage to the stored canisters due to a seismic event is not postulated. It may be postulated that one or more canisters in the deep end of the refueling canal or in spent fuel pool A could be damaged and their contents reconfigured due to the failure of seismically unqualified components above the canisters. However, 4350 ppm has been shown to assure subcriticality for the entire core fuel load in a conservative bounding configuration (Reference 14). Thus, a boron concentration of 4350 ppm will assure subcriticality for any smaller amount of fuel material or for fuel in a less than optimal geometry.

(3) The actual boron concentration that has been maintained in the RCS since head lift is 5050 ppm. It is planned that there will be an operating boron concentration in the refueling canal and fuel pool A in excess of 4350 ppm. Thus, this higher boron concentration provides an additional margin of criticality safety over that provided by the conservative analysis in Reference 14.

(4) The collapse of the defueling trolley into the dry section of the refueling canal may result in a severe reconfiguration of the canister contents. However, from the discussion of the boron dilution potential in Section 3.1.1.4, there is no credible source of unborated water that could act as a moderator for the exposed fuel. From Reference 23, criticality is not possible for unmoderated uranium containing less than about 5 weight percent U-235.

(5) A design basis earthquake could be postulated to cause leakage of the closure plate separating fuel pools A and B, resulting in dilution of pool A water. This event (which would be a concern only if canisters were damaged in a manner that would expose fuel) was not considered credible, as discussed in Section 3.1.1.4 (item (4), boron dilution).

(6) The collapse of the defueling trolley or other seismically unqualified equipment could be postulated to damage the liner in the refueling canal or in the spent fuel pool resulting in a loss of borated water. Undamaged canisters would still contain water borated to at least 4350 ppm and dewatered canisters would have no moderator. It could be postulated that some canisters would be

damaged resulting in exposed fuel. The analyses provided in References 16 and 17 demonstrated that criticality would not occur given a conservative analysis of draindown in the reactor vessel. Thus, by comparison to the analysis of vessel draindown, the criticality of either intact or damaged canisters due to a pool draindown is not considered credible given the canister separation, the presence of additional poison material in each canister, and the lack of moderator.

Thus, criticality was not considered a credible mechanism for generating an increased source term for a defueling canister. The defueling canister source term is due to the existing radionuclide inventory and is defined below:

(1) One percent of the core radionuclide inventory as specified in Section 3.1.2.1 is assumed to be the source term for each canister involved in postulated accidents. This represents the maximum loading of any canister type, as specified in Reference 1.

(2) The form of the core material in the three types of defueling canisters differs. The fuel canister is designed for bulk core materials. The knockout canister is designed for debris ranging in size from 140 μm to whole fuel pellets. The filter canister is designed to remove small debris and particulates down to 0.5 μm. Accident sequences involving a single canister were assumed to involve a filter canister, which will have the largest amount of small particulates. Accident sequences involving multiple canisters are assumed to have an average distribution of core fragment sizes.

3.1.2.7 Sumps

Reactor Building

There are approximately 20,000 gallons of water currently in the RB basement. The most recent sample was taken in November 1984. To define a source term for the RB basement that will bound future activities, the radionuclide concentrations in the November 1984 sample were increased by a factor of five and the amount of basement water was assumed to be the 100,000 gallon administrative limit.

H-3	0.15 μCi/ml
Sr-90	8.0 μCi/ml
Cs-134	1.1 μCi/ml
Cs-137	24.5 μCi/ml

The inventory in the RB sump itself is about 8000 gallons which potentially has a higher radionuclide concentration than the bulk of the water in the RB basement. The sump inventory will be pumped out prior to sludge removal from the sump. The RB sump inventory estimate from Reference 37, which is not expected to be exceeded during defueling, is assumed to be the limiting source term for liquid waste disposal (WDL) system processing. The source term is:

H-3	1.0 E+0 μCi/ml
Sr-90	6.1 E+0 μCi/ml
Cs-134	6.2 E+0 μCi/ml
Cs-137	9.6 E+1 μCi/ml

Auxiliary Building

Contaminated water potentially enters the auxiliary building sump from numerous waste processing and cleanup sources. The most contaminated water source to the auxiliary building sump through defueling is expected to be the Miscellaneous Waste Holdup Tank. To define the source term for this analysis, the highest recent MWHT activity (sample No. 85-01891; Feb. 22, 1985) was increased by a factor of five. The volume in the sump was assumed to correspond to the administrative limit of 9000 gallons. The resulting source term is:

H-3	2.1 E+0 µCi/ml	Sb-125	1.5 E-1 µCi/ml
Co-60	2.1 E-2 µCi/ml	Cs-134	2.9 E-1 µCi/ml
Sr-90	1.2 E+0 µCi/ml	Cs-137	7.0 E+0 µCi/ml
Ru-106	2.4 E-1 µCi/ml	Ce-144	2.1 E-1 µCi/ml
Ag-110m	2 E-2 µCi/ml		

Other Sumps

There are other sumps in the plant that may have some level of contamination. However, the source terms for the reactor and auxiliary building sumps bound radionuclide inventory in other sumps.

3.1.2.8 SDS System

Liner/Filters

The submerged demineralizer system (SDS) liners and filters are located in Fuel Pool B. In the past, the SDS liners have been loaded as high as 60,000 Ci of Cs-137 (152,000 total curies). These liners have been shipped offsite to DOE facilities for disposition. Current and future operations are expected to result in liner loadings less than 5000 Ci. Typical current liner and filter loadings are provided below. For analysis purposes, sequences involving a postulated source term from a single SDS liner or filter were assumed to have 10 times the typical loadings shown below; each liner was assumed to have the typical (or average) loading shown below if a postulated source term involved multiple liners.

SDS ACTIVITY (Ci)

ISOTOPE	SAND FILTER	CUNO FILTER	ZEOLITE LINER
H-3	negligible	negligible	2.7 E-3
Co-60	3.3 E-1	3.9 E-3	3.0 E-1
Sr-90	384.4	8.4	432.8
Ru-106	2.3	2.8 E-2	8.9
Ag-110m	negligible	negligible	4.5 E-1
Sb-125	9.0	3.3 E-2	2.7
I-129	negligible	negligible	2.3 E-5
Cs-134	36.9	1.6	46.1
Cs-137	330.2	15.8	1032.2
Ce-144	8.7 E-1	3.9 E-2	4.3
U-234	4.8 E-5	1.1 E-5	6.9 E-6
U-235	8.0 E-7	6.6 E-8	5.8 E-7

SDS ACTIVITY (Ci)

ISOTOPE	SAND FILTER	CUNO FILTER	ZEOLITE LINER
U-236	1.5 E-6	3.5 E-7	negligible
U-238	1.1 E-5	2.7 E-6	1.2 E-5
Np-237	4.3 E-7	1.1 E-7	negligible
Pu-238	2.1 E-6	6.2 E-5	1.5 E-5
Pu-239	2.5 E-5	6.6 E-4	1.7 E-5
Pu-240	5.4 E-6	1.7 E-4	4.6 E-6
Pu-241	4.7 E-4	1.2 E-2	2.2 E-3
Pu-242	6.7 E-10	1.7 E-8	negligible
Am-241	6.4 E-4	9.5 E-5	1.6 E-5
Cm-242	2.9 E-6	1.1 E-6	9.9 E-9
Cm-244	3.8 E-7	1.0 E-7	negligible
TOTAL	1466.1	63.4	2951.2

Process flow

The radionuclide concentration in the SDS process lines varies according to the process source, e.g., RCS, sumps, MU&P system. These concentrations are specified in the description of the respective sources. The maximum SDS processing rate is 15 gpm. A release from the process stream to the air is postulated through the above-water piping in the fuel handling building.

3.1.2.9 MU&P System

Demineralizers

The makeup and purification system demineralizers are seismic Category I vessels housed in Category I cubicles. No post-accident hardware modifications have been made to those cubicles with the exception of insertion of a one-half inch tygon tube through the 3" resin fill line in each demineralizer. This modification would not compromise the structural integrity of these demineralizers during a seismic event. Thus, a source term from the MU&P system because of the structural failure of the demineralizers during a seismic event is not postulated.

Elution flow

Cleanup of the MU&P demineralizers uses an elution process whose path is from the demineralizers to the neutralizer tanks and then to SDS. Some of the piping is non-seismically designed. The source term assumed for the consequence analysis is the maximum Cs concentration from the MU&P SER for rinse and elution (Reference 25) and concentrations of other radionuclides calculated from a recent neutralizer tank sample (Sample 84-12915; Oct. 19, 1984) results.

Sr-90	8.1 μCi/ml
Cs-134	3.1 μCi/ml
Cs-137	65 μCi/ml
All U	2×10^{-3} μCi/ml
All Pu	10^{-9} μCi/ml
All I	10^{-8} μCi/ml

The system flow rate is 5 gpm to the neutralizer tanks; flow from the neutralizer tanks to SDS may be up to 15 gpm.

Resin Sluicing

The use of a sluicing process is planned for removing contaminated MU&P resins. Final details of the operation have not yet been developed. Basically, however, the operation will involve sluicing the resin from the MU&P demineralizers to the spent resin tanks where it will be processed for ultimate disposition. The source term for the process was developed from a sample of the resin prior to the start of MU&P elution (Reference 36). For this analysis, the source term assumes diluting the solid resin in a 20 to 1 ratio (ratio used for the resin elution process) and processing the mixture at 50 gpm. The resulting source term for the sluicing process is:

ISOTOPE	CONCENTRATION (μCi/ml)	ISOTOPE	CONCENTRATION (μCi/ml)
Co-60	9.4 E-2	U-236	3.7 E-6
Sr-90	1.9 E+1	U-238	1.5 E-5
Sb-125	3.5 E-1	Pu-238	8.4 E-4
Cs-134	3.0 E+1	Pu-239	5.5 E-3
Cs-137	4.3 E+2	Pu-240	1.7 E-3
Ce-144	4.7 E-1	Pu-241	1.4 E-1
U-234	6.2 E-5	Pu-242	1.9 E-7
U-235	2.1 E-6		

3.1.2.10 Contaminated Waste Storage

Reactor building

There are two designated waste storage areas in the reactor building. One is located on the 305' elevation near the personnel airlock; the other is located on the 347' elevation adjacent to the enclosed stairwell. The maximum material in each area is about 6000 lbs. with an isotopic inventory of 2.0 E-1 Ci of Cs-137, 8 E-3 Ci of Cs-134 and 8 E-3 Ci of Sr-90 (Reference 2).

AFHB/Ventilation Filters

There is a maximum capacity of about 22,000 lbs of contaminated waste storage in the auxiliary and fuel handling buildings. This corresponds to about 25 drums and six LSA boxes of compacted waste plus two 50 ft^3 storage bins. A complete HEPA ventilation filter bank would occupy about four LSA boxes; it is assumed that four boxes are occupied by the equivalent of the recently characterized "B" train of the reactor building HVAC system. (The B train had been characterized because of its relatively high activity.) The remaining rad trash storage activity has a typical concentration of 6.3 μCi/lbm. For conservatism, the source term assumed for the AFHB waste storage area (including ventilation filters) is five times the radionuclide inventory associated with the maximum capacity described above. The source term is: 1.4 Ci of Cs-137, 7 E-2 Ci of Cs-134 and 7 E-2 Ci of Sr-90.

Respirator & Laundry Facility

The source term for the Respirator and Laundry Facility is based on the annual inventory which is shipped from that facility. It is conservatively assumed that 10% of the annual inventory would be located in that building at one time. The 1984 inventory of 5000 containers of LSA represented approximately 0.66 Ci; 10% of that inventory is 0.07 Ci which is assumed to have the same isotopic makeup as AFHB rad trash. The resulting source term is 3.1 E-2 Ci of Cs-137, 2 E-3 Ci of Cs-134 and 2 E-3 Ci of Sr-90.

3.1.2.11 EPICOR System

Liners

The highest loading of an EPICOR liner currently on-site is less than 20 Ci. However, GPUN is considering requesting approval of the use of high integrity container (HIC) processing vessels in the EPICOR system, which would allow significantly higher curie loadings of the EPICOR liners. To define a source term for this analysis, the highest EPICOR prefilter loading (Prefilter 16, Reference 38) was used to bound possible future inclusion of high integrity containers in the EPICOR system. (EPICOR prefilters were considered abnormal waste with activities too high to permit commercial disposal. Thus, a prefilter inventory will exceed the expected loadings in an EPICOR HIC which will be disposed of commercially.) The bounding source term for an EPICOR liner is:

Isotope	Activity (Ci)	Isotope	Activity (Ci)
Co-60	1.7 E-1	U-234	3.4 E-5
Sr-90	3.0 E+1	U-235	1.3 E-6
Ru-106	3.0 E-1	U-238	7.4 E-6
Ag-110m	4.5 E-1	Pu-238	2.2 E-6
Sb-125	1.9 E+0	Pu-239	2.5 E-5
Cs-134	9.4 E+1	Pu-240	5.4 E-6
Cs-137	9.5 E+2	Pu-241	4.7 E-4
Ce-144	6.2 E-1	Am-241	1.6 E-5
		Cm-242	7.9 E-8

The EPICOR liners are situated within a cylindrical concrete cask which is surrounded by a lead brick wall. The top of the liner is covered with a portable lead shield and a steel lid. The foundation of the chemical cleaning building, primary concrete walls and structural steel frame are designed to seismic Category I (Reference 26). Thus, rupture of the liners in their processing position due to a seismic event was not considered credible. The source term for the liner failure is due to the postulated drop and rupture of a single liner from the monorail system while in transit.

Filters

There are four main filters currently used in the EPICOR system: (i) a downstream crud filter which could accumulate a concentration of cobalt and (ii) three downstream filters which are designed for resin capture in the event of a liner failure. Samples of the crud filter indicate no cobalt buildup. Thus, the EPICOR filters do not present a significant source term.

Process flow

The EPICOR system is currently used primarily as a polishing system for the SDS. Future operations may allow for direct flow to EPICOR from several contaminated sources. The radionuclide concentration in the EPICOR process lines varies according to the process source. For this analysis, the highest process sources were assumed; these sources are the bounding RB basement and auxiliary building sump terms developed in Section 3.1.2.7.

The EPICOR liquid line break is postulated to occur directly to atmosphere since the upper walls and roof of the EPICOR building are not qualified seismic Category I. The EPICOR system flow rate is 10 gpm.

3.1.2.12 Radwaste Systems

Liquid waste disposal (WDL) system

The WDL system is a pre-accident system and was originally qualified to seismic Category I. However, a detailed analysis has not been conducted to determine if a seismic Category I component failure due to the failure of a non-seismically qualified component is credible. Thus, leakage from that system is postulated using the liquid source term developed in Section 3.1.2.7 for the RB sump.

Solid waste disposal (WDS) system

The major source terms that will be associated with the solid waste disposal system will be due to the use of WDS system piping for MU&P resin sluicing and for radioactive sludge processing. The MU&P resin sluicing operation was described in Section 3.1.2.9. The source term for sludge processing is based on RB basement sludge sample results in Reference 37. It was assumed that the sludge is mixed with sluice water in a 20 to 1 ratio (as is done during MU&P elution). The resulting source term is:

Radionuclide	Concentration (μCi/ml)	Radionuclide	Concentration (μCi/ml)
Co-60	7.6 E-2	Cs-137	3.4 E+2
Sr-90	1.1 E+2	Ce-144	2.5 E+0
Ru-106	1.1 E+0	U-235	1.9 E-4
Sb-125	1.6 E+0	Other U	5.0 E-5
I-129	1.5 E-3	All Pu	1.6 E-2
Cs-134	3.0 E+1		

The process flow rate is assumed to be 50 gpm.

Gaseous waste disposal (WDG) system

Activity levels from the gaseous waste disposal system (WDG) are negligible in comparison to the potential airborne releases from the WDL and WDS systems.

3.1.2.13 DWC System

Process flow

The defueling water cleanup system will process RCS, refueling canal, and/or fuel pool water through ion exchangers and filters. The limiting source term for a process flow leakage is due to RCS processing. Radionuclide concentrations for the RCS are provided in Section 3.1.2.4. Process flow from the RCS is a maximum of 400 gpm. Drainage from the RCS through a broken line by a functioning DWCS pump is limited to 3000 gallons by the elevation of the anti-syphoning holes in the suction line (Reference 34). Releases are postulated to reactor building air.

Liners/filters

There will be four filter canisters in the deep end of the refueling canal which are to be used for DWCS processing of the RCS; there are also four canisters in fuel pool A planned for DWCS cleanup of fuel pool A and the refueling canal. The source term associated with these canisters is as defined in Section 3.1.2.6. There are planned to be three ion exchangers for DWCS processing located in the northwest area of the Fuel Handling Building. Their loading is assumed to be the same as the observed loading for an SDS liner (approximately 3000 Ci) specified in Section 3.1.2.8.

3.1.2.14 IIF Processing

Processing from the internals indexing fixture (IIF) through the SDS may be replaced by DWCS processing for defueling. Potential DWCS processing rates exceed those used in IIF processing. The source term associated with IIF processing is the RCS source term identified in Section 3.1.2.4.

3.1.2.15 Interim Solid Waste Staging Facility (ISWSF)

The maximum inventory of low level waste stored in the ISWSF is approximately eight hundred 55 gallon drums, ninety LSA boxes, and sixty 55 ft^3 liners. The source term taken from Reference 29 is:

Sr-90	8.1 Ci
Cs-134	3.9 Ci
Cs-137	81 Ci

3.1.2.16 Solid Waste Storage Building

The Solid Waste Storage Building ("paint shed") contains small amounts of low specific activity radwaste; there are no plans to significantly increase the radwaste stored in this building. An estimate of transportable radionuclide contamination in the building is provided in Reference 30. In that analysis, a release fraction of $10^{[illegible]}$ was assumed due to potential combustion of flammables in that building. The results of the analysis indicate that potential doses are insignificant compared to doses from other postulated accidents in this report. Thus, no further analysis of this source term was performed.

3.1.2.17 Solid Waste Staging Facility (SWSF)

The Solid Waste Staging Facility is a passive facility for temporary staging of waste prior to ultimate disposition. The facility is composed of concrete modular structures (two modules currently exist); each module is partitioned into 60 cells which may house various quantities of waste. The sump compartment is seismically qualified. There is essentially no combustible material in the SWSF. The source term for this analysis assumes damage to the SWSF from the DBE which results in exposure of 10 SDS liners to the atmosphere. The radionuclide inventory for an SDS liner was provided in Section 3.1.2.8.

3.1.2.18 Processed Water System

The source of water in the processed water storage and recycle system is two 500,000 gallon processed water storage tanks (PWSTs). Tank makeup is the effluent from the EPICOR system; thus radionuclide concentrations in the PWSTs are generally much less than water entering EPICOR. The only exception to this is the concentration of tritium which is not removed by EPICOR. Recent samples (Dec. 31, 1984) indicate a tritium concentration of about 0.5 µCi/ml in each tank. This concentration is not expected to be exceeded during the defueling process and, thus, represents the PWST source term for this analysis.

3.2 Seismic Induced Accident Sequences and Associated Consequences

In this section, the potential accident sequences associated with a seismic event are summarized and tabulated as Table 3.1. In some cases, accident sequences have been postulated because an analysis of the potential mechanism has not been performed. (For example, the postulation of seismic category I component failure because of possible effects of non-seismically qualified components.) The consequences of the potential accident sequences have been estimated using the methods and dose conversion factors specified in Regulatory Guide 1.109, NUREG-0172, and Regulatory Guide 1.4. The major assumptions used in the consequence calculation are summarized below:

(1) Two hour doses at the site boundary were calculated for potential airborne releases. To conservatively estimate the offsite dose, the 0-1 hour fifth percentile X/Q of 6.1×10^{-4} sec/m^3 from Appendix 2D of the TMI-2 FSAR was used. Doses were calculated for the adolescent age group which generally is the maximum dose receptor for the radionuclides of interest at TMI-2.

(2) Organ dose conversion factors were as specified in Regulatory Guide 1.109; for radionuclides not considered in RG 1.109, NUREG-0172 dose conversion factors were used. Because of the current radionuclide inventory, the bone was considered as the critical organ. The dose to this organ was calculated and presented in Table 3.1. The breathing rate of 1.2 m^3/hr was as specified in Regulatory Guide 1.4 (Reference 24).

(3) Insights from NUREG/CR-3535 (Reference 35) and ICRP 26 (Reference 6) were used to assess the significance of the calculated bone doses. Specifically, NUREG/CR-3535 reports significantly higher dose conversion factors for the bone surface than other organs, including bone marrow; ICRP 26 provides equivalent risk weighting factors for bone surface and thyroid. Thus, the 10CFR Part 100 organ (thyroid) dose limitation is assumed to be applicable for situations in which the bone is the critical organ.

(4) Releases to the environment were assumed to be unfiltered. Deposition of airborne activity on walls and surfaces was neglected.

(5) For accident sequences involving spraying liquids or potential leaks under pressure, the fraction of liquid leaking from the system that could become airborne as a transportable liquid aerosol was conservatively assumed to be 10^{-3} (Reference 10).

(6) For accident sequences involving liquid spills, the fraction of liquid that could become a transportable liquid aerosol was conservatively assumed to be 10^{-4} (Reference 10).

(7) For accident sequences involving standing liquid pools that may be subject to a wave action from the seismic event, the airborne release fraction is conservatively assumed to be 3×10^{-5}. This value is the maximum airborne release fraction measured in experiments with a 1 meter liquid free fall (Reference 10).

(8) Some postulated accidents result in particulate material being uncovered and exposed to air. If the particulates are not trapped in filters, the fraction of solid particulates that become airborne is conservatively assumed to be 10^{-3}, based on experiments with free falling powder (Reference 10).

(9) An airborne release fraction of 10^{-4} was assumed for particulates contained on resins and ruptured or crushed filters (Reference 33).

(10) For accident sequences in which the source term is involved in a fire, the airborne release fraction is conservatively assumed to be 10^{-3} (Reference 33).

(11) Particles larger than about 10 μm are predominantly deposited in the nasopharyngeal region and have much less radiological significance than smaller particles which are preferentially deposited in the bronchial system and lung (Reference 10). Furthermore, particles larger than 10 μm deposit rapidly by aerosol deposition mechanisms such as gravitational settling and inertial impaction. A characterization of the particle size distribution (45 - 4000 μm) in a core sample found that about 1.5% of core particulates were less than the smallest size range analyzed, 45 μm (Reference 31).

Based on the observed particle size distribution, it is conservatively assumed that a fraction of 10^{-2} represents the material in the fuel and knockout canisters that is in the size range of concern, 10 µm. This implies that all of the material in those canisters is particulate and all particulates in the range less than 45 µm are actually 10 µm or less.

The filter canisters are designed to collect small debris and particulates as small as 0.5 µm. It is conservatively assumed that all of the material collected in the filter canisters in 10 µm or less.

(12) Accident sequences involving an exposed core could result in transport of particulates from the rubble bed and RV internals surfaces. The total amount of core material that is particulate is uncertain. For this analysis, it is assumed that 10% of the total core mass is particulate which resides on a surface that would be exposed in the event of a vessel draindown.

(13) Coupling assumptions (8) and (11), and assuming that an accident sequence exposes the entire canister contents, yields a total material release fraction of 10^{-5} for postulated accidents involving a direct airborne release from a fuel or knockout canister. From assumptions (9) and (11), the potential material release fraction from a filter canister exposed to air is 10^{-4}. (The release fraction for a filter canister is used for sequences involving the release from a single canister.)

Using the assumptions regarding the amount of exposed particulate core material (12), the distribution of particulate sizes (11), and the fraction that becomes airborne (8), the release fraction for an exposed core is 10^{-6}.

TABLE 3.1 POSTULATED ACCIDENT SEQUENCES DUE TO SEISMIC EVENT [1]

ACCIDENT SEQUENCE					
ID NO.	SOURCE	DRIVING FORCE	MECHANISM	OFFSITE DOSE BONE (rem) [2]	COMMENTS
1	Drained RPV	Atmospheric ΔP	Non-seismically qualified component causes incore instrument guide tube failure.	1.4 E+1	Section 3.1.2.1 source term with airborne release fraction of 10^{-6} for exposed core; Section 3.1.2.4 source term for 50,000 gallons flowing RCS inventory with airborne release fraction of 10^{-4}; Section 3.1.2.7 source term for 100,000 gallons standing RCS sump water with release fraction of 3×10^{-5}. Air exchange rate is 200% of building volume per hour due to open equipment hatch; steaming driving force small compared to air exchange.
2	Drained RPV	Combustible Fire	Coincident in containment fire; source not involved in fire; driving force dominated by atmospheric ΔP (Sequence 1).	Bounded by Sequence 1	
3	Loose Contamination (RB)	Atmospheric ΔP	Mechanical disruption of smearable contamination.	Negligible compared to Sequence 4	Airborne release fraction small compared to fire.
4	Loose Contamination (RB)	Combustible Fire	Heatup from fire releases smearable contamination through open equipment hatch	3.2 E-2	Section 3.1.2.5 source term for particulate releases assuming all RB contaminated surfaces exposed to fire. Airborne release fraction of 10^{-3}; air exchange rate of 200% per hour.

[1] Accident sequences for which a driving force or unanalyzed source term is not credible have not been listed. For example, there are no credible sequences with pyrophoric driving force, no credible sequences with core recriticality.

[2] Two hour site boundary dose to adolescent; unfiltered release to environment.

TABLE 3.1 POSTULATED ACCIDENT SEQUENCES DUE TO SEISMIC EVENT [1]

ACCIDENT SEQUENCE ID NO.	SOURCE	DRIVING FORCE	MECHANISM	OFFSITE DOSE BONE (rem) [2]	COMMENTS
5	Loose Contamination (other Buildings)	All	Mechanical disruption of smearable contamination	Negligible compared to RB sequence	
6A	MU&P Process Flow	Atmospheric ΔP	Non-seismic piping failure.	1.3 E-1	Section 3.1.2.9 source term for elution flow; flow at 15 gpm for 2 hours to SDS. Airborne release fraction of 10^{-3} assumed for aerosol release due to spraying pipe leak; air exchange rate of 200% of building volume per hour through open FHB truck bay door.
6B	MU&P Resin Sluicing	Atmospheric ΔP	Non-seismic piping failure	1.2 E+0	Section 3.1.2.9 source term for resin sluicing; flow at 50 gpm for two hours. Airborne release fraction is 10^{-3} for spraying pipe leak; air exchange rate at 200% building volume per hour.
7	MU&P Process Flow/ Sluicing	Combustible Fire	Non-seismic piping failure with coincident fire in area; source term not involved in fire; driving force dominated by atmospheric ΔP.	Bounded by Sequences 6A, 6B	

[1] Accident sequences for which a driving force or unanalyzed source term is not credible have not been listed. For example, there are no credible sequences with pyrophoric driving force, no credible sequences with core recriticality.

[2] Two hour site boundary dose to adolescent; unfiltered release to environment.

TABLE 3.1 POSTULATED ACCIDENT SEQUENCES DUE TO SEISMIC EVENT [1]

ACCIDENT SEQUENCE ID NO.	SOURCE	DRIVING FORCE	MECHANISM	OFFSITE DOSE BONE (rem) [2]	COMMENTS
8	EPICOR Liner	Atmospheric ΔP	Liner dropped and ruptured due to failure of non-seismically qualified monorail.	Small compared to Sequence 9	Airborne release fraction for resin liner is 10^{-6}.
9	EPICOR Liner	Combustible Fire	Liner ruptures as in Sequence 8; fire occurs due to shorting of unqualified electrical wiring or other sources; combustible materials involve liner.	1.0 E-1	Section 3.1.2.11 EPICOR liner source term; airborne release fraction of 10^{-3} for material involved in fire; entire liner assumed involved; release directly to atmosphere.
10	EPICOR Filters	Atmospheric ΔP/ Fire ΔP	Same as EPICOR Liners. Negligible loading on filters.	Small Compared to Sequence 9	
11A	EPICOR Processing (RB Basement)	Atmospheric ΔP	Break of non-seismic EPICOR process line.	1.1 E-1	Section 3.1.2.11 source term for process flow (RB basement); flow at 10 gpm for 2 hours. Airborne release fraction of 10^{-3} corresponding to spraying leak; release directly to atmosphere.
11B	EPICOR Processing (AFHB Sump)	Atmospheric ΔP	Break of non-seismic EPICOR process line.	1.6 E-2	Section 3.1.2.11 source term for process flow (aux. bldg. sump); other parameters same as Sequence 11A.

[1] Accident sequences for which a driving force or unanalyzed source term is not credible have not been listed. For example, there are no credible sequences with pyrophoric driving force, no credible sequences with core recriticality.

[2] Two hour site boundary dose to adolescent; unfiltered release to environment.

TABLE 3.1 POSTULATED ACCIDENT SEQUENCES DUE TO SEISMIC EVENT [1]

ACCIDENT SEQUENCE ID NO.	SOURCE	DRIVING FORCE	MECHANISM	OFFSITE DOSE BONE (rem) [2]	COMMENTS
12	EPICOR Processing	Combustible Fire	Non-seismic piping failure with coincident fire in area; source term not involved in fire; driving force dominated by atmospheric ΔP.	Bounded by Sequence 11	
13A	SDS Liner	Atmospheric ΔP	Liner rupture occurs under water in spent fuel pool (220,000 gal) due to collapse into pool of non-seismic components.	2.9 E-1	Section 3.1.2.8 liner source term (~30,000 Ci); break in air not credible due to cask loading under water and seismic Category I FHB crane; 100% of liner inventory released into pool; airborne release fraction is 3 E-5
13B	SDS Cuno Filter	Atmospheric ΔP	Filter rupture occurs under water in spent fuel pool (220,000 gal) due to collapse into pool of non-seismic components.	5.6 E-3	Section 3.1.2.8 cuno filter source term (~640 Ci); other parameters same as Sequence 13A.
13C	SDS Sand Filter	Atmospheric ΔP	Filter rupture occurs under water in spent fuel pool (220,000 gal) due to collapse into pool of non-seismic components.	2.5 E-1	Section 3.1.2.8 sand filter source term (~15,000 Ci); other parameters same as Sequence 13A.
14	SDS Liner/ Filters	Fire ΔP	Liner/filter exposure postulated to occur under water; source term not involved in fire; driving force dominated by atmospheric ΔP.	Small compared to Sequences 13	

1 Accident sequences for which a driving force or unanalyzed source term is not credible have not been listed. For example, there are no credible sequences with pyrophoric driving force, no credible sequences with core recriticality.

2 Two hour site boundary dose to adolescent; unfiltered release to environment.

TABLE 3.1 POSTULATED ACCIDENT SEQUENCES DUE TO SEISMIC EVENT [1]

ACCIDENT SEQUENCE ID NO.	SOURCE	DRIVING FORCE	MECHANISM	OFFSITE DOSE BONE (rem) [2]	COMMENTS
15A	SDS Process Flow (RCS)	Atmospheric ΔP	Non-seismic line outside of spent fuel pool breaks when processing RCS water.	7.7 E-1	Section 3.1.2.4 source term; flow at 15 gpm for two hours to SDS. Airborne release fraction of 10^{-3} assumed corresponding to spraying pipe leak; air exchange rate of 200% building volume per hour.
15B	SDS Process Flow (RB Basement)	Atmospheric ΔP	Non-seismic line outside of spent fuel pool breaks when processing RB basement water.	1.2 E-1	Section 3.1.2.7 source term; other parameters same as Sequence 15A.
15C	SDS Process Flow (MU&P)	Atmospheric ΔP	Non-seismic line outside of spent fuel pool breaks when processing MU&P elution.	Same as Sequence 6A	Section 3.1.2.9 elution flow source term; other parameters same as Sequence 15A.
15D	SDS Process Flow (All)	Atmospheric ΔP	Underwater line break	Small compared to Sequences 15A,B,C	Release fraction from standing pools due to potential wave action is 3 E-5.
16	SDS Process Flow (All)	Combustible Fire	Same as Sequence 15; source not involved in fire; driving force dominated by atmospheric ΔP.	Bounded by Sequences 15 A,B,C	

[1] Accident sequences for which a driving force or unanalyzed source term is not credible have not been listed. For example, there are no credible sequences with pyrophoric driving force, no credible sequences with core recriticality.

[2] Two hour site boundary dose to adolescent; unfiltered release to environment.

TABLE 3.1 POSTULATED ACCIDENT SEQUENCES DUE TO SEISMIC EVENT [1]

ACCIDENT SEQUENCE ID NO.	SOURCE	DRIVING FORCE	MECHANISM	OFFSITE DOSE BONE (rem) [2]	COMMENTS
17	DWCS/IIF Process Flow	Atmospheric ΔP	Non-seismically qualified line break; maximum source is RCS process line break before reaching filters and ion exchangers.	1.7 E+0	Section 3.1.2.4 source term; 400 gpm break bounds DWCS and IIF flow rates; RCS drained to elevation of anti-syphoning holes in the sparge line limiting leak to 3,000 gallons. Airborne release fraction of 10^{-3} corresponding to spraying leak; 200% building volumes per hour.
18	DWCS/IIF Process Flow	Combustible Fire	Non-seismic piping failure with coincident in-containment fire; source term not involved in fire; driving force dominated by atmospheric ΔP.	Bounded by Sequence 17	
19A	DWCS Liners	All	Exposure of liner postulated to occur in fuel handling building due to collapse of non-seismic components onto liner.	1.2 E-1	Typical SDS curie loading (~3000 Ci) in Section 3.1.2.8; assumed airborne release fraction of 10^{-4} for resin material; air exchange rate of 200% building volumes per hour.
19B	DWCS Filters	All	Non-seismically qualified components collapse into deep end of canal.	Included in Sequence 23C.	
20	Fuel Pool A/B	All	Airborne release due to mechanical disturbance. Effect small in comparison to sequences involving SDS inventory and defueling canister leaks in pool.	Negligible	

1 Accident sequences for which a driving force or unanalyzed source term is not credible have not been listed. For example, there are no credible sequences with pyrophoric driving force, no credible sequences with core recriticality.

2 Two hour site boundary dose to adolescent; unfiltered release to environment.

TABLE 3.1 POSTULATED ACCIDENT SEQUENCES DUE TO SEISMIC EVENT [1]

ACCIDENT SEQUENCE ID NO.	SOURCE	DRIVING FORCE	MECHANISM	OFFSITE DOSE BONE (rem) [2]	COMMENTS
21	Canal	All	Same as Sequence 20: doses negligible compared to sequences involving releases into canal.	Negligible	
22	RB/Aux Bldg Sumps	All	Airborne release due to mechanical disturbance; bounded by sequences involving spraying pipe when processing.	Bounded by Sequence 15B	Airborne release fraction for standing pools subject to wave action is 3 E-5. Pathway to outside atmosphere is more tortuous than that assumed in Sequence 15.
23A	Defueling Canister (Transport)	Atmospheric ΔP	Canister and/or defueling equipment drops into vessel due to non-seismic handling equipment. Core reconfiguration results, but criticality not credible. Increased source term due to mechanical disturbance. Effect bounded by drained RPV sequence.	Bounded by Sequence 1	
23B	Defueling Canister (Transport)	Atmospheric ΔP	Canister drops over dry refueling canal and is crushed by non-seismic fuel handling equipment.	1.2 E+1	Filter canister assumed which implies that entire canister contents is particulate of of size 10 μm or less; 1% of Section 3.1.2.1 source term assumed. Airborne release fraction is 10^{-4} for materials trapped on filters; 200% per hour air exchange rate corresponding to open equipment hatch.

1 Accident sequences for which a driving force or unanalyzed source term is not credible have not been listed. For example, there are no credible sequences with pyrophoric driving force, no credible sequences with core recriticality.

2 Two hour site boundary dose to adolescent; unfiltered release to environment.

TABLE 3.1 POSTULATED ACCIDENT SEQUENCES DUE TO SEISMIC EVENT [1]

ACCIDENT SEQUENCE ID NO.	SOURCE	DRIVING FORCE	MECHANISM	OFFSITE DOSE BONE (rem) [2]	COMMENTS
23C	Defueling Canister (Transport)	Atmospheric ΔP	Non-seismically designed defueling trolley drops in deep end of canal. Transported canister and ten canisters/filters in deep end are damaged opening source term to 60,000 gallons of water. (If drop also results in leakage of canal, the effect would be bounded by Sequence 1 which postulates draining of RPV.)	4.0 E-1	Each canister holds a maximum of 1% of core material implying 11% of core material exposed to canal water. Average particulate size distribution assumed implying 1% ≤10 μm. Airborne release fraction from water is 3 E-5; air exchange rate is 200% building volumes per hour.
23D	Defueling Canisters (Storage Racks)	Atmospheric ΔP	Defueling equipment falls on storage racks in spent fuel pool A. Contents of 30 canisters are released to water.	1.1 E+0	Number of canisters impacted based on cross sectional area of fuel handling trolley; other parameters same as Sequence 23C
24	Defueling Canister (Transport)	Combustible Fire	Same as Sequence 23A, B, C, D with coincident fire; source term not involved in fire.	Bounded by Sequences 23A,B,C,D	
25	Rad Trash Storage Areas (All locations)	Atmospheric ΔP	Mechanical disturbance of waste storage areas.	Negligible Compared to Sequence 26	

1 Accident sequences for which a driving force or unanalyzed source term is not credible have not been listed. For example, there are no credible sequences with pyrophoric driving force, no credible sequences with core recriticality.

2 Two hour site boundary dose to adolescent; unfiltered release to environment.

TABLE 3.1 POSTULATED ACCIDENT SEQUENCES DUE TO SEISMIC EVENT [1]

ACCIDENT SEQUENCE ID NO.	SOURCE	DRIVING FORCE	MECHANISM	OFFSITE DOSE BONE (rem) [2]	COMMENTS
26	Rad Trash Storage Area	Combustible Fire	Fire in waste storage area ignited by non-seismic electrical equipment or other ignition source.	2.2 E-4	Section 3.1.2.10 source term for auxiliary building contaminated waste; entire 22,000 pounds capacity involved in fire. Airborne release fraction is 10^{-3}; air exchange rate is 200% building volume per hour.
27	WDL System	Atmospheric ΔP	Non-seismic equipment collapses onto Liquid Radwaste Systems piping.	5.5 E-1	Section 3.1.2.12 source term (RB sump); 150 gpm leak corresponding to typical operating flow rate of WDL-P-5A or B; airborne release fraction of 10^{-3} assumed for spraying pipe leak; 200% building volume per hour air exchange rate.
28	WDS System	Atmospheric ΔP	Non-seismic equipment collapses onto Solid Radwaste System piping	5.5 E+0	Section 3.1.2.12 source term (RB basement sludge); 50 gpm flow rate assumed. Airborne release fraction of 10^{-3} assumed for spraying pipe leak; 200% building volume per hour air exchange rate.

[1] Accident sequences for which a driving force or unanalyzed source term is not credible have not been listed. For example, there are no credible sequences with pyrophoric driving force, no credible sequences with core recriticality.

[2] Two hour site boundary dose to adolescent; unfiltered release to environment.

TABLE 3.1 POSTULATED ACCIDENT SEQUENCES DUE TO SEISMIC EVENT [1]

ACCIDENT SEQUENCE ID NO.	SOURCE	DRIVING FORCE	MECHANISM	OFFSITE DOSE BONE (rem) [2]	COMMENTS
29	WDL/WDS Systems	Combustible Fire	Same as sequence 27 with coincident fire; source term not involved in fire; driving force dominated by atmospheric ΔP.	Bounded by Sequences 27, 28	
30	ISWSF	Atmospheric ΔP	Negligible release compared to Sequence 31	Negligible	
31	ISWSF	Fire ΔP	Seismic event causes collapse of building and fire in ISWSF.	2.5 E-2	Section 3.1.2.15 source term; entire inventory involved in fire. Airborne release fraction of 10^{-3}; release directly to atmosphere.
32	Solid Waste Storage Building	All	Negligible release compared to other sequences.	Negligible	
33	SWSF	Atmospheric ΔP	Collapse of non-seismic structure	1.3 E+0	Collapse damages 10 SDS liners and exposes them directly to atmosphere; each liner has a typical loading (~3000 Ci) with a radionuclide distribution specified in Section 3.1.2.8. Airborne release fraction is 10^{-4}.
34	PWST	Atmospheric ΔP	Non-seismic PWST leaks	Negligible [3]	Section 3.1.2.18 source term; 500,000 gallons exposed to atmosphere; airborne release fraction of 10^{-4} assumed for flowing liquids.

1 Accident sequences for which a driving force or unanalyzed source term is not credible have not been listed. For example, there are no credible sequences with pyrophoric driving force, no credible sequences with core r criticality.

2 Two hour site boundary dose to adolescent; unfiltered release to environment.

3 Tritium provides no dose to bone surface; dose to lung was calculated as 3.2 E-6 rem.

4.0 Conclusions

Offsite consequences have been conservatively calculated for a spectrum of accident sequences that could result from a design basis earthquake. The consequences were modeled in terms of a two hour dose at the site boundary, as specified in 10CFR Part 100. The 10CFR Part 100 specifies a limitation of 25 rem external dose to the whole body and 300 rem dose to the thyroid. Doses from postulated seismically induced accident sequences at TMI-2 would be negligible compared to these limits. However, direct comparison to these limits has a marginal value because of the existing radionuclide inventory (e.g., negligible amounts of iodine) and the nature of potential releases (i.e., particulate matter).

To consider the significance of the current radionuclide inventory, doses to the critical organ, the adolescent bone were calculated. Doses to the critical organ did not exceed the critical organ limitation for thyroid in 10CFR Part 100. Insights from recent literature indicate that the 10CFR Part 100 organ dose limitation is an appropriate limitation for the critical organ for postulated TMI-2 accidents.

More restrictive criteria for certain accident sequences have been promulgated in Chapter 15 of the NRC Standard Review Plan. Specifically, the consequences of some accident sequences must be "well within" (less than 25%) or a "small fraction" of (less than 10%) of 10CFR Part 100. The analysis indicates that there is no accident sequence that exceeds these more restrictive guidelines.

5.0 Unreviewed Safety Question Evaluation

The purpose of this SER is to justify the non-seismic design for TMI-2 post-accident systems, structures and components. As part of this evaluation, the criteria specified in 10CFR 50.59 for determining an unreviewed safety question have been examined. A proposed change to a facility involves an unreviewed safety question if the change:

(1) involves an increase in the probability or consequences of an accident previously evaluated,

(2) creates the possibility of a new or different kind of accident from any accident previously evaluated, or

(3) involves a reduction in the margin of safety.

These criteria are considered below.

The accident of concern in this SER is the design basis earthquake (DBE). The probability of the DBE is clearly unchanged by the construction of components, systems and structures at TMI-2. The consequences of a DBE in the current TMI-2 configuration should be compared to the consequences of related accidents analyzed in NUREG-0107 (Reference 32). The accidents described in NUREG-0107 that are related to the TMI-2 recovery are the fuel handling accident and the gas decay tank rupture. The fuel handling accident resulted in a two hour exclusion boundary whole body dose of 3 rem and a thyroid dose of 46 rem; the gas decay tank rupture resulted in a whole body dose of 6 rem. There is no accident sequence postulated for a DBE that exceeds these doses.

The TMI-2 defuellng process is similar to normal fuel handling operations with the exceptions that defueling canisters replace fuel cladding as the primary containment for core materials and the defueling operation is not conducted completely underwater. The defueling path is the same as for normal refuelings, i.e., from the reactor vessel to the spent fuel pool via the fuel transfer system. A broad spectrum of defueling accidents have been analyzed which can be characterized as involving either a release of core material from its primary containment or as a release from a radwaste processing system. The fuel handling and gas decay tank rupture accidents considered in NUREG-0107 are representative of these categories of accidents. Thus, a new type of accident has not been identified for the design basis earthquake.

Existing safety margins at TMI-2 for criticality control and prevention of further core damage due to overheating are conservative and will remain conservative through defueling. The maintenance of both functions is accomplished by passive systems. Specifically, an existing RCS boron concentration assures subcriticality for bounding core reconfigurations and the core is cooled by ambient heat loss to the environment. This SER has demonstrated that there is no credible mechanism associated with a DBE that will result in loss of subcriticality or core heating which could result in further damage. Thus, there is no decrease in safety margins as a result of a seismic qualification exemption for post-accident systems.

In summary, the non-seismic design of post-accident systems does not constitute an unreviewed safety question as defined by 10CFR 50.59.

5.0 References

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18. Memo from R. C. Ashline (Oak Ridge National Laboratory) to J. Greenborg (GPUN),: ORIGEN-2 Calculations for TMI-2 Fuel," dated June 22, 1984.

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21. TPO/TMI-125, "Data Report: Reactor Building Radiological Characterization," Rev. 0, January 1985.

22. TER 15737-2-G03-114, "Defueling Canisters," Rev. C.

23. NUREG/CR-0095, "Nuclear Safety Guide TID-7016 Revision 2," J. Thomas Ed., June 1978 (also referred to as ORNL/NUREG/CSD-6).

24. Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors."

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