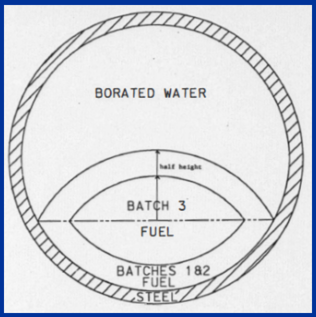
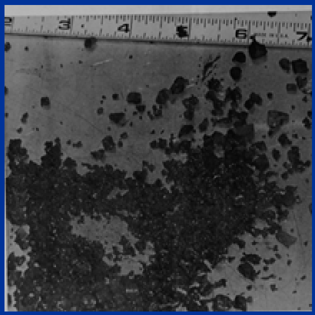
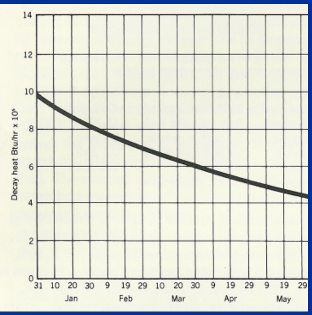
U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment



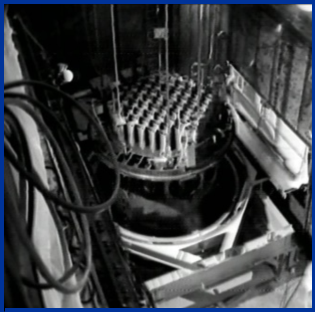
Criticality

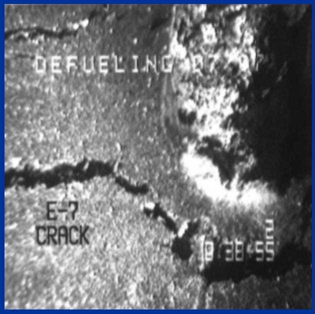

Hydrogen


Pyrophoricity



Decay Heat


Load Drop



Vessel Integrity

Occupational Exposure


Radiological Release

Other Concerns

Three Mile Island Accident of 1979 Knowledge Management Digest

Cleanup Safety Evaluations

1979–1993

NUREG/KM-0001, Supplement 3

AVAILABILITY OF REFERENCE MATERIALS IN NRC PUBLICATIONS

NRC Reference Material

As of November 1999, you may electronically access NUREG-series publications and other NRC records at the NRC's Library at www.nrc.gov/reading-rm.html. Publicly released records include, to name a few, NUREG-series publications; *Federal Register* notices; applicant, licensee, and vendor documents and correspondence; NRC correspondence and internal memoranda; bulletins and information notices; inspection and investigative reports; licensee event reports; and Commission papers and their attachments.

NRC publications in the NUREG series, NRC regulations, and Title 10, "Energy," in the *Code of Federal Regulations* may also be purchased from one of these two sources:

1. **The Superintendent of Documents**
   U.S. Government Publishing Office
   Washington, DC 20402-0001
   Internet: www.bookstore.gpo.gov
   Telephone: (202) 512-1800
   Fax: (202) 512-2104

2. **The National Technical Information Service**
   5301 Shawnee Road
   Alexandria, VA 22312-0002
   Internet: www.ntis.gov
   1-800-553-6847 or, locally, (703) 605-6000

A single copy of each NRC draft report for comment is available free, to the extent of supply, upon written request as follows:

Address: **U.S. Nuclear Regulatory Commission**
Office of Administration
Digital Communications and Administrative Services Branch
Washington, DC 20555-0001
E-mail: distribution.resource@nrc.gov
Facsimile: (301) 415-2289

Some publications in the NUREG series that are posted at the NRC's Web site address www.nrc.gov/reading-rm/doc-collections/nuregs are updated periodically and may differ from the last printed version. Although references to material found on a Web site bear the date the material was accessed, the material available on the date cited may subsequently be removed from the site.

Non-NRC Reference Material

Documents available from public and special technical libraries include all open literature items, such as books, journal articles, transactions, *Federal Register* notices, Federal and State legislation, and congressional reports. Such documents as theses, dissertations, foreign reports and translations, and non-NRC conference proceedings may be purchased from their sponsoring organization.

Copies of industry codes and standards used in a substantive manner in the NRC regulatory process are maintained at—

The NRC Technical Library
Two White Flint North
11545 Rockville Pike
Rockville, MD 20852-2738

These standards are available in the library for reference use by the public. Codes and standards are usually copyrighted and may be purchased from the originating organization or, if they are American National Standards, from—

American National Standards Institute
11 West 42nd Street
New York, NY 10036-8002
Internet: www.ansi.org
(212) 642-4900

Legally binding regulatory requirements are stated only in laws; NRC regulations; licenses, including technical specifications; or orders, not in NUREG-series publications. The views expressed in contractor prepared publications in this series are not necessarily those of the NRC.

The NUREG series comprises (1) technical and administrative reports and books prepared by the staff (NUREG–XXXX) or agency contractors (NUREG/CR–XXXX), (2) proceedings of conferences (NUREG/CP–XXXX), (3) reports resulting from international agreements (NUREG/IA–XXXX),(4) brochures (NUREG/BR–XXXX), and (5) compilations of legal decisions and orders of the Commission and the Atomic and Safety Licensing Boards and of Directors' decisions under Section 2.206 of the NRC's regulations (NUREG–0750).

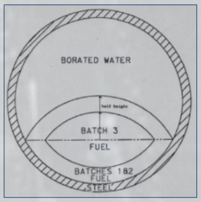
ABSTRACT

After the accident at Three Mile Island Generating Station, Unit 2 (TMI-2) in 1979, safety evaluations were required by the licensee and NRC for most cleanup activities and new cleanup systems. This supplement (Supplement 3) details the safety evaluations of 64 cleanup activities that were performed during the 1979-1993 period and has been organized into the following groups: data collection; pre-defueling preparations; defueling tools and systems; defueling operations; and liquid waste management systems.

The safety evaluations for each of the cleanup activities are discussed by safety topics which include: criticality; boron dilution; decay heat removal; fire protection; hydrogen generation; industrial safety; instrument interference; impact on Unit 1 activities; load drop; occupational exposure, including internal and external exposures; pyrophoricity; radiation protection/as low as reasonably achievable practice; radiological release; reactor vessel integrity; seismic hazard; radiation shielding; and vital equipment protection.

Supplement 3 is part of the NUREG/KM-0001 series and provides complimentary details of the TMI-2 cleanup activities which can be found in its preceding supplements. Supplement 1 provided summary descriptions of programs, activities, systems and tools that were long involved in the decade-long cleanup campaign of the damaged reactor core and severely contaminated equipment and buildings. The Digital Versatile Discs (DVDs) accompanying Supplement 1 contain most of the references cited in Supplement 3. Whereas Supplement 2 consolidated many of the experiences and lessons during the TMI-2 cleanup that had been recorded in numerous reports and papers. Contents of the DVDs are available at the Idaho National Laboratory Web site (https://tmi2kml.inl.gov).

Criticality

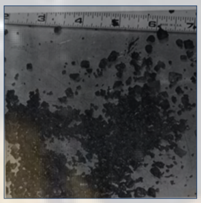


Lower reactor vessel head criticality model assumed the most reactive spherical configuration. Assumptions included highest enriched fuel in the center, surrounded by lower enriched fuel, no neuron poisons in fuel regions, and surrounded by steel neutron reflector with borated water.

Hydrogen

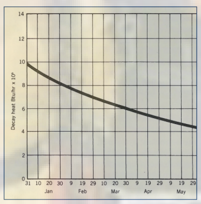
Debris canisters contained catalyst beds to recombine radiolytic hydrogen and oxygen to prevent buildup of combustible mixtures of gases. Porous metal catalyst packages were incorporated into the upper and lower (shown) heads that recombined gases at various orientations.

Pyrophoricity



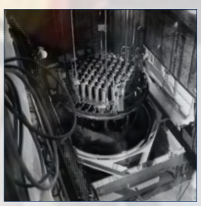
Finely divided metallic Zircaloy, which was fragmented during the accident, caused concern over a pyrophoric reaction (and fire) if core debris were exposed to air. Samples of fine core debris that were subjected to ignition tests resulted in no pyrophoric reactions.

Decay Heat



The main concern with decay heat removal during defueling and its preparations involved the lowering of reactor coolant level in the reactor coolant system and reactor vessel. Evaluations supported the Quick Look video inspection and reactor vessel head removal.

Load Drop



Plenum being raised from the reactor vessel through the water filled internals indexing fixture. It was transferred to its storage stand in the partially flooded deep end of the fuel transfer canal. A six-foot-high dam allowed the deep end to be flooded to provide radiation shielding.

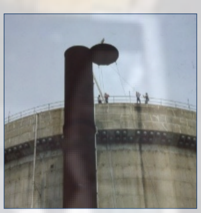
Vessel Integrity

Tears were found in the cladding of the lower reactor vessel head around melted instrumentation penetration, E7. Cracks were attributed to stresses associated with the thermal gradient in the thick-walled carbon steel vessel during the heating and cooling phases of the accident.

Occupational Exposure

A radiation work permit was required to access contaminated ceiling areas above the clean lower areas of the auxiliary building main corridors. Lowers walls and floor were painted to shield beta radiation. Piping and cable trays hanging from the ceiling remained contaminated.

Radiological Release



TMI-2 ventilation stack being uncapped for purging accident-generated radioactive krypton gas from the containment. The ventilation stack was previously capped to redirect ventilation from the auxiliary and fuel handling building to the temporary supplementary air filtration system.

Other Concerns

Other safety considerations included: fire protection; industrial occupational safety; control room instrument interference; seismic hazard; Unit 1 safety and operations impact; and vital equipment protection of critical safety functions.

TABLE OF CONTENTS

ACKNOWLEDGMENTS

The editors of this knowledge management supplement greatly appreciate the contributions from many individuals who assisted in the research, drafting, fact checking, and reviews of this supplement about the safety evaluations of the many activities associated with the cleanup of Three Mile Island Nuclear Station, Unit 2 (TMI-2). This supplement details safety evaluations of over five dozen cleanup activities that examined over a dozen safety considerations. Most of these evaluations by the licensee and U.S. Nuclear Regulatory Commission (NRC) were revised numerous times to account for incremental experience gained during the challenging cleanup campaign that spanned 13 years. The contributors to this supplement were either involved with the writing or reviews of the original TMI-2 safety evaluations or were subject matter experts who reviewed portions of this supplement.

Knowledge contributors included the following (alphabetically):

Don Algama, NRC nuclear criticality expert
Lake Barrett, first responder and executive manager of the NRC onsite TMI-2 project office
James Byrne, former TMI-2 licensing manager and corporate senior official
Michel Call, NRC nuclear criticality expert and co-editor of this supplement
Anthony Huffert, NRC health physics expert
Don Marksberry, former TMI-2 quality assurance auditor and NRC co-editor of this supplement
Michael Masnik, former NRC TMI project office member and TMI-2 project manager
Donald Mueller, retired nuclear criticality expert from the Oak Ridge National Laboratory
Charles Negin, former TMI-2 cleanup planning consultant
Joseph Shepherd, hydrogen expert from the California Institute of Technology
Gordon Skillman, first responder and first TMI-2 defueling manager
Theodore Smith, NRC TMI-2 project manager
James Tarpinian, former TMI-2 radiological engineering manager
Thomas Wellock, NRC historian

Their contributions to this NUREG/KM should not be considered as endorsements of the historical contents of this report.

The editors also recognize the important contributions of the technical editors from QTE who made this epic document readable. Further, the contribution of the NRC co-editor Alice Chung greatly improved the technical understanding of the legacy safety evaluations from the 1980s, and her editing of the current text and terminology should improve computer translations into foreign languages. Thanks to Karen Dickey and the NRC publications team whose diligence and friendly assistance make these NUREGs world-class reports.

Most NRC legacy records of the TMI-2 docket (No. 05000320) and other license dockets are now digitally available from the Web-based public Agencywide Documents Access and Information System (ADAMS) (nrc.gov). Special thanks go to the NRC records management team who performed this monumental task during the COVID-19 pandemic. Their accomplishments are ultimate knowledge management.

ABBREVIATIONS

$\Delta\rho$	change in reactivity
AECL	Atomic Energy of Canada Limited
ADAMS*	**Agencywide Documents Access and Management System (NRC)**
AFHB	auxiliary and fuel handling building
AIGIH	American Conference of Governmental Industrial Hygienists
ALARA*	**as low as reasonably achievable (exposure reduction practices)**
amu	monomeric molecular weight
ANSI	American National Standards Institute
APSR	axial power shaping rod
ASME	American Society of Mechanical Engineers
B_4C	boron carbide
B&W	Babcock & Wilcox
BNL	Brookhaven National Laboratory
BTU	British thermal unit
BWST	borated water storage tank
cc/kg	cubic centimeter per kilogram
cfm	cubic feet per minute
CFR*	***Code of Federal Regulations***
CLDS	canister loading decontamination system
CMAA	Crane Manufacturers Association of America
CPS	canister positioning system
CRDM	control rod drive mechanism
CTS	canister transfer shield
DE	diatomaceous earth
degrees C	degrees Celsius
degrees F	degrees Fahrenheit
DF	decontamination factor
DHR	decay heat removal
DOE*	**U.S. Department of Energy**
DOT	Department of Transportation
DVD	Digital Versatile Disc
DWCS	defueling water cleanup system
DWS	demineralized water system
EPICOR	EPICOR II water cleanup system (not an abbreviation)
EPRI	Electric Power Research Institute
FCSR	fuel canister storage rack

FHB	fuel handling building
FHBVS	fuel handling building ventilation system
FTC	fuel transfer canal
g/cm^3	gram per cubic centimeter
GDC	general design criterion/criteria
gpm	gallon per minute
GPU*	**GPU Nuclear Corporation (the licensee)**
GWTS	gaseous waste treatment subsystem
H_2O_2	hydrogen peroxide
HEPA	high-efficiency particulate air (filter)
HIC	high-integrity container
IIF	internals indexing fixture
INEL*	**Idaho National Engineering Laboratory (currently INL)**
k_{eff}	effective neutron multiplication (factor)
$k_{infinity}$	infinite neutron multiplication (factor)
LCSA	lower core support assembly
LRVOS	liner recombiner and vacuum outgassing system
LWR	light-water reactor
MDM	metal disintegration machining
MHC	mini hot cell
MPC	maximum permissible concentration
NA	not applicable
NEPA	National Environmental Policy Act
NO	nitrous oxide
NO_2	nitrous dioxide
NRC*	**U.S. Nuclear Regulatory Commission**
NuPac	Nuclear Packaging, Inc.
NUREG*	**Nuclear Regulatory (NRC report)**
NUREG/KM*	**Nuclear Regulatory/Knowledge Management (NRC report)**
ORNL	Oak Ridge National Laboratory
OSHA	Occupational Safety and Health Administration
OTSG	once-through steam generator
PA	plenum assembly
PEIS*	**Programmatic Environmental Impact Statement**
PNL	Pacific Northwest Laboratory

ppm	parts per million
psi	pounds per square inch
psia	pounds per square inch absolute
psig	pounds per square inch gauge (pressure)
PSLDS	pressurizer spray line defueling system
RCBT	reactor coolant bleed tanks
RCS	reactor coolant system
RG	regulatory guide
RIS	regulatory issue summary
RV	reactor vessel
RVLH	reactor vessel lower head
scf	standard cubic feet
SCSB	single canister support bracket
SDS	submerged demineralizer system
SER	safety evaluation report
SF	safety factor
SFP-A	spent fuel pool "A"
SFP-B	spent fuel pool "B"
Si(Li)	silicon lithium
SPND	self-powered neutron detector
SRST	spent resin storage tank
SSTR	solid state track recorders
TER	technical evaluation report
TLD	thermoluminescent dosimeters
TMI-1	Three Mile Island Unit 1
TMI-2*	**Three Mile Island Unit 2**
TRVFS	temporary reactor vessel filtration system
UCSA	upper core support assembly
UHP	ultrahigh pressure
UO_2	uranium dioxide
vdc	volts direct current
ZrO_2	zirconium dioxide

* Note: Abbreviations marked with an asterisk and in bold are used throughout this NUREG/KM supplement. All others are first spelled out in each numbered section or subsection.

1 INTRODUCTION

TABLE OF CONTENTS

1.1 Background

This supplement is the third report in this NUREG/KM series about the accident at Three Mile Island Unit 2 (TMI-2) and its recovery and cleanup. NUREG/KM-0001, "Three Mile Island Accident of 1979 Knowledge Management Digest, Overview," (1) issued June 2016, was the first in the U.S. Nuclear Regulatory Commission's (NRC's) knowledge management series of reports whose purpose is to preserve knowledge of the important historical events and research that have shaped the NRC's regulatory programs for present and future generations. The main report presents an overview of the accident: emergency response, investigations, regulatory implications, and accident recovery. Supplement 1, "Recovery and Cleanup," (2) issued June 2016, expounds on the technical details of recovery and cleanup activities: management and oversight, plant stabilization, worker protection, data acquisition and analysis, waste management, decontamination, defueling, and after defueling.

Revision 1 of the main overview report (3) and Supplement 1 include document collections that were derived from correspondence between the licensee (a, 4) and the NRC and from the results of research activities sponsored by GPU Nuclear (the licensee), Electric Power Research Institute, the NRC and the U.S. Department of Energy (DOE). (b) The accompanying Digital Versatile Discs (DVDs) contain about 100,000 pages in over 4000 documents, 500 photographs and diagrams, and three NRC video presentations about the accident and recovery activities. A hypertext markup language (HTML)-based interactive guide is provided on the DVDs to help navigate through the historical records from both of these reports.

Supplement 2, "The Cleanup Experience: A Literature Review," issued December 2020, (5) catalogs the many experiences and insights documented in numerous reports and papers spanning the 1979 to 1993 cleanup period at TMI-2. The experiences described in the second supplement focus on those aspects of TMI-2 relating to long-term plant stabilization, cleanup, and defueling. Descriptions of these experiences were based on an extensive review of a wide range of reports, papers, presentations, and interviews with personnel formerly from the key organizations involved in the cleanup.

This newest supplement, "Cleanup Safety Evaluations, 1979–1993," provides extensive details of the evaluations performed by the licensee and the NRC to assess the safety of cleanup activities at TMI-2. Safety evaluations were required for most cleanup activities and new cleanup systems.

The groups of cleanup activities include: (●) data collection; (●) pre-defueling preparations; (●) defueling tools and systems; (●) defueling operations; and (●) liquid waste management

[a] The GPU Nuclear Corporation was added as the licensee for TMI-2 and replaced Metropolitan Edison Company as the licensee authorized to operate TMI-2, effective January 1, 1982.

[b] The Idaho National Engineering Laboratory (INEL is now call INL) was tasked by DOE to manage and coordinate the DOE's TMI Information and Examination Program, which was jointly sponsored by GPU Nuclear (the licensee), EPRI, the NRC, and the DOE and collectively called GEND. This program published most research and development reports (called GEND reports) for the nuclear industry and the public. Other reports were also published by each organization.

systems. This supplement discusses the safety evaluations of 64 activities and cleanup systems.

Safety topics discussed for each activity include: (●) criticality; (●) boron dilution; (●) decay heat removal; (●) fire protection; (●) hydrogen generation; (●) industrial safety; (●) instrument interference; (●) impact on TMI Unit 1 activities; (●) load drop; (●) occupational exposure; including internal and external exposures; (●) pyrophoricity; (●) radiation protection/as low as reasonably achievable (ALARA) practice; (●) radiological release; (●) reactor vessel integrity; (●) seismic hazard; (●) radiation shielding; and (●) vital equipment protection.

The remainder of this introduction briefly overviews the NRC's safety evaluation review process and information for navigating through this supplement.

1.2 Safety Evaluations

The NRC reviewed and approved the implementation of most of the cleanup activities at TMI-2. Each request that was submitted by the licensee was evaluated by the NRC's onsite TMI-2 Project Office. The NRC's evaluations ensured that all applicable regulatory and license requirements were met to protect the public's health and safety and to minimize worker exposure. To the extent applicable, evaluations of safety and environmental impacts of the proposed activity were based on the evaluations of cleanup alternatives discussed in the NRC's "Programmatic Environmental Impact Statement Related to Decontamination and Disposal of Radioactive Wastes Resulting from March 28, 1979 Accident Three Mile Island Nuclear Station, Unit 2" (called the PEIS). [6, 7]

This section summarizes the NRC's safety evaluation process as documented in NUREG-0698, [8, 9, 10] "NRC Plan for Cleanup Operations at Three Mile Island Unit 2," as revised.

- ***Licensee Proposals***. The licensee prepared a safety evaluation report (SER) [c] for each major cleanup activity and a technical evaluation report for each new system. In addition to many of the licensee's contractors and consultants, the DOE and its national laboratories were major resources for technical analyses that provided the basis for many evaluations. The licensee's SERs included four general parts: (●) basic description of the activity or system design/operations; (●) safety analysis that covered a wide range of safety considerations; (●) a review under Title 10 of the *Code of Federal Regulations* (10 CFR) 50.59, "Changes, tests and

[c] Called a "safety analysis report" in some TMI-2 submittals. Today, the licensee submits a safety analysis report (SAR), and the NRC reviews the SAR and issues a safety evaluation report documenting the results of the agency's review.

experiments,” [d] that determined if a change to the TMI-2 recovery technical specifications [e] was needed (or an unreviewed safety question [f] was involved); and (●) environmental assessment to demonstrate that the SER was bounded by the PEIS.

The SER of the proposed request was submitted to the NRC’s onsite project office for review and approval.

- ***Proposal Approval: NRC Staff***. The NRC Commissioners’ policy statement [11] that endorsed the PEIS gave the staff the authority to approve most cleanup activities. The Commissioners stated in April 1981 that, as the licensee proposed specific major decontamination activities, the NRC staff [g] would determine whether these proposals, and the associated impacts that were predicted to occur, were within the scope of those already assessed in the PEIS. Except for the disposition of processed accident-generated water, which the Commissioners wanted to decide on later, the staff was allowed to act on each major cleanup activity without the Commissioners’ approval if the activity and the associated impacts were within the scope of those assessed in the PEIS. [12]

- ***Proposal Approval: NRC Commissioners***. In the year following the accident, the NRC Commissioners approved radiological effluent criteria for the interim period, before the issuance of the PEIS, for radiological releases based on data-gathering and maintenance operations. Following the issuance of the PEIS, if a cleanup task was evaluated to be outside the scope of the PEIS, then the NRC’s project office would recommend to the NRC Commissioners either their approval of the task or the development of a supplement to the PEIS, as required by the National Environmental Protection Act. This action was not necessary during the 13-year cleanup campaign. The licensee never submitted a proposal for a cleanup activity that the NRC determined would result in an environmental impact outside the scope of the PEIS and its supplements. [13]

---

[d] The regulation in section 10 CFR 50.59 permits the holder of an operating license to make changes to the facility or perform a test or experiment, provided that the change, test, or experiment does not meet any of the criteria in section 10 CFR 50.59 that require application for a license amendment.

[e] Within a year following the accident, the NRC issued an order that established the new “recovery technical specifications” (Appendix A to the facility operating license). These specifications considered the condition of plant systems at that time. Requirements were modified as cleanup progressed. Appendix B to the facility operating license, called the “environmental technical specifications,” which established limitations on effluent releases and discharges; these limitations were unchanged and where to remain in effect except as provided in the order and subsequent modification of requirements.

[f] The term “unreviewed safety question” is no longer used in today’s regulations (section 10 CFR 50.59 was revised to provide safety criteria in plain English). In 1986, the term meant a change that involved any of the following: (1) the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report could be increased; or (2) a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report could be created; or (3) the margin of safety as defined in the basis for any technical specification was reduced.

[g] The five NRC Commissioners are usually called collectively the “Commission” and everyone else is “staff.” This NUREG/KM uses the terms “Commissioners” instead of the Commission and “NRC” instead of NRC staff, unless both are being discussed.

- ***Safety Evaluations***. The NRC and the licensee conducted safety evaluations to ensure that the proposed cleanup activity could be implemented without significant risk to the health and safety of the public. The NRC's safety reviews evaluated the safety concerns applicable to the proposed request. In general, safety evaluations by the licensee and NRC addressed similar safety considerations to ensure worker and public safety that involved the control of radioactive materials under normal, anticipated, and accident conditions. One significant advantage at TMI-2 was an intact containment building [h] to contain radioactive materials. Technical specification requirements ensured containment building integrity, as needed, for cleanup operations.

General safety concerns that were typically addressed in the licensee's and NRC's SERs, as applicable, included: (●) criticality control; (●) boron dilution; (●) radiological releases during normal operations and postulated accident conditions; (●) hydrogen evolution and control; (●) pyrophoricity of debris fines; (●) decay heat removal; (●) heavy load drops (impacting reactor vessel integrity); (●) reactor vessel and primary system integrity (draining); (●) fire protection; (●) occupational exposure to radiation; (●) impact on other plant activities and operations; (●) instrument interference (disruption of instrumentation required by technical specification); and (●) submerged combustion (burning and cutting operations damaging the reactor vessel wall).

- ***Cleanup Activities and Systems***. The term "cleanup" is used in this NUREG/KM to mean actions taken to decontaminate the plant (surfaces and water); defuel the reactor vessel, reactor coolant system and auxiliary systems; and dispose of radioactive waste. Safety evaluations were required for most cleanup activities and new cleanup systems. The categories of cleanup activities included: (●) data collection, such as video inspections in the reactor vessel, reactor vessel underhead characterization, and core bore sample examination; (●) pre-defueling preparations, such as reactor vessel head removal, heavy load handling, and reactor vessel plenum removal; (●) defueling tools and systems development, such as defueling water cleanup system, defueling canisters, plasma arc torch, and core bore machine; (●) defueling operations, such as bulk defueling of the central core region, defueling lower reactor vessel core support structures, defueling reactor vessel lower head, and defueling upper core support structure behind the core former baffle plates; and (●) liquid waste management systems, such as EPICOR II and the submerged demineralizing system.

This supplement discusses the safety evaluations of 64 activities and systems. Supplement 1 provides summary descriptions of these activities and systems. Detailed descriptions can be found in the cited SERs and supporting correspondence.

1.3 How To Use This Report

This report summarizes the most notable safety evaluations of TMI-2 postaccident cleanup activities. Cleanup activities are defined in this NUREG/KM supplement to include

[h] To eliminate confusion with other nuclear power plant designs, the term "containment building" is used throughout this NUREG/KM instead of "reactor building," which is the standard terminology at TMI-2 for the pressure suppression function.

decontamination, examinations, preparations for defueling, and defueling. The report describes the many safety evaluations performed, enabling the reader to understand the thinking of the evaluators at the time, the expectations and the reality, uncertainties in data, and measurement and mitigation methods. It also presents a high-level chronology of cleanup activities. The reader is cautioned to refer to the cited safety evaluation and supplemental correspondence, if any, for the complete record of the safety evaluation at a particular point in time. A safety evaluation preceded essentially every major undertaking in the TMI-2 cleanup. However, this NUREG/KM may describe safety evaluations for tasks that were never undertaken. [i] Not all tasks for which a safety evaluation was prepared were executed. The discussions of safety evaluation reports (SERs) presented here are not official NRC records or technical positions.

- ***Scope***. Safety topics that are summarized in this NUREG/KM supplement include: (●) criticality, including boron dilution; (●) decay heat removal; (●) fire protection; (●) hydrogen generation; (●) industrial safety; (●) instrument interference; (●) impact on TMI Unit 1 activities; (●) load drops; (●) occupational exposure, including internal and external exposures; (●) pyrophoricity; (●) radiation protection/ALARA; (●) radiological releases; (●) reactor vessel integrity; (●) seismic hazard; (●) radiation shielding; and (●) vital equipment protection.

- ***Not in Scope***. This supplement does not include safety evaluations of activities associated with (●) shipping and transportation of radioactive wastes or fuel debris; (●) offsite storage of such wastes; (●) changes to technical specifications; (●) NRC orders; (●) containment building purge of krypton-85 in 1980; (●) disposal of decontaminated accident-generated water through evaporation; or (●) post-defueling monitoring storage. Safety evaluations of recovery activities to stabilize the plant during the early weeks and months following the accident were typically not documented in formal SERs; therefore, this supplement does not include these early activities.

- ***Report Organization***. This NUREG/KM supplement is structured to conveniently group related sections of all safety evaluations for each safety topic. This should allow technical reviewers to collectively review the safety evaluations associated with their specialized field in the designated chapter, instead of having to search through the hundreds of SERs for numerous cleanup activities.

Each chapter of a safety topic contains: (●) table of contents; (●) introduction that gives the scope of the chapter; (●) list of key studies and technical reports associated with the safety topic; (●) licensee's safety evaluations associated with a standard set of 64 cleanup activities, systems, and tools; (●) the NRC's safety reviews as documented in its SERs; and (●) endnotes that include cited references. Some safety evaluations of cleanup activities and systems did not address a particular safety topic. In these cases, the section title indicates not applicable (NA).

---

[i] An example of a licensee's SER being submitted to and approved by the NRC, but the request never being implemented, is the hydraulically powered shredder. This 2.5-ton device could have been placed in the reactor vessel to reduce the size of fuel pins and other core debris and facilitate the loading of fuel canisters or debris buckets. The Electric Power Research Institute (EPRI) report, "The Cleanup of Three Mile Island Unit 2, A Technical History: 1979 to 1990," dated October 1, 1990 (EPRI-NP-6931), provides a concise overview of all proposed and implemented cleanup plans and activities. This report is currently (at the time of this publication) available from EPRI's Web site.

Some evaluations were repeated across multiple activities or simply referred to an earlier SER. As the cleanup proceeded, some discussions of the evaluations of a particular safety topic became shorter. In some later cases, the safety topic was no longer a concern. The level of detail presented in this supplement for each activity generally reflects the details that were documented in the cited SER.

- ***Companion Supplement***. Supplement 1 of this NUREG/KM [14] provides high-level overviews of various cleanup activities and descriptions of systems and equipment used during the cleanup. This summary information is not repeated in this supplement.

The Digital Versatile Discs (DVDs) accompanying Supplement 1 contain most of the references cited in Supplement 3. Contents of the DVDs are available at the Idaho National Laboratory Web site (https://tmi2kml.inl.gov).

- ***Report Structure***. To help the reader navigate this lengthy report, each chapter is tiered to no more than three numbered subsections and four tiers of bullets. Bullets are used extensively to help partition numbered sections in the following order of tiers: "•" first tier; "o" second tier; "–" third tier; and "■" fourth tier (infrequent). An additional bullet "(●)" is used in a paragraph at any tier for listing items or concepts.

- ***Completeness***. Chapters of this supplement include safety topics that had substantial evaluations. This supplement covers most cleanup activities, with the exceptions of those noted in the scope discussion above. Each activity is addressed in each safety topic chapter, whether an evaluation of that topic was discussed in the cited SER.

In most instances, the safety evaluation discussions reflect the content in the cited SER, almost in its entirety, unless otherwise stated. However, the discussion may not reflect all subsequent or supplementary information such as revisions and correspondences. The editors tried to select the documents that provided the most informative and complete evaluation. In some cases, informative supplementary materials are also discussed. This supplement generally does not include system and operational descriptions that were typically described in the SER.

- ***The NRC's Reviews***. The NRC's SER that was discussed for each activity may be associated with an earlier or later revision of the cited licensee's SER. However, the editors chose the revision to best represent a complete safety evaluation, given that the reviews of later SERs were most often brief or referred to previous SER revisions (or, in some cases, did not address a topic because its safety importance changed over time or the revision reflected a minor change).

- ***Editorial Changes***. Editorial changes to the original text were generally limited to standardize terminology and to use consistent grammatical tenses. The writing styles from numerous SERs that were written by different authors over the 13-year cleanup campaign were generally not edited. However, in many cases, statements were revised to provide clarity or improve readability. Text in many cited SERs contained lengthy paragraphs or nondescript paragraphs covering an assortment of safety topics. To provide some description of the contents, this supplement sectioned the text from the cited SER using bullets with keyword titles.

- ***Abbreviations***. To improve readability and results of language translator software, abbreviations were kept to a minimum. Only a few of the most common abbreviations are used throughout this supplement. Others that are frequently repeated in a numbered section (or subsection) are spelled out at the beginning of the section that contains them. A list of all abbreviations is provided at the beginning of this supplement.

- ***Units of Measure***. The unit of measure (in English units or the International System of Units) used in the original cited document also appears in this supplement. In some cases, the cited document uses both units. A conversion chart is provided on the back cover. Abbreviations of units were kept to a minimum to improve the results of language translations.

- ***Editor's Notes***: Editor's notes were added, as appropriate, to add context and to update information that was not included in the cited SER.

- ***Additional Information***. The reader should refer to the cited SER for complete details about a specific cleanup activity of interest. For the complete safety evaluation of a specific activity, refer to the SER, its revisions, and supporting correspondences. The importance of safety considerations often changed as the cleanup progressed over time and reactor core and plant conditions became better known.

1.4 Key NRC Regulations and Regulatory Guides

The key *Code of Federal Regulations* (CFR) sections and NRC regulatory guides that were often cited in the safety evaluation reports (SERs) and referenced in this supplement are listed below. Others that are referenced less frequently are noted in the text. To improve readability, titles are noted only once in each chapter that contains them. However, the full title is included in the endnotes of each chapter. The reader should consult the cited SER for the exact revisions of the regulations and guides, as they changed over time.

- ***Cited CFRs***. The following commonly cited regulations in this supplement are provided with full titles at the beginning of each chapter that cites them and throughout the endnotes:

  - 10 CFR Part 20, "Standards for Protection Against Radiation" [15]

  - 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities" [16]

  - 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants" [17]

  - 10 CFR Part 50, Appendix I, "Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion 'As Low as is Reasonably Achievable' for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents" [18]

  - 10 CFR Part 71, "Packaging and Transportation of Radioactive Material" [19]

- 10 CFR Part 100, “Reactor Site Criteria” [(20)]

- ***Cited NRC Regulatory Guides***. The full titles of the following commonly cited regulatory guides in this supplement are given at the beginning of each chapter that cites them and throughout the endnotes:

  - Regulatory Guide 1.109, “Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I” [(21)]

  - Regulatory Guide 1.111, “Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors” [(22)]

- ***CFR Past Editions***. Past editions of the CFRs from the 1980s are currently not available at the NRC or Government Publishing Office Web sites. However, early editions may be available at a local Federal Depository Library (refer to the Catalog of U.S. Government Publications Web site).

- ***NRC Regulatory Guides Past Editions***. Past and current revisions of NRC regulatory guides can be found at the NRC Web site (nrc.gov).

1.5 Endnotes

Reference citations are exact filenames of documents on the Digital Versatile Discs (DVDs) to NUREG/KM-0001, Supplement 1, [(23)] “Three Mile Island Accident of 1979 Knowledge Management Digest: Recovery and Cleanup,” issued June 2016, DVD document filenames start with a full date (YYYY-MM-DD) or end with a partial date (YYYY-DD). Citations for references that are not on a DVD begin with the author organization or name(s), with the document title in quotation marks. Contents of the DVDs are available at the Idaho National Laboratory Web site (https://tmi2kml.inl.gov).

---

[1] USNRC, “Three Mile Island Accident of 1979 Knowledge Management Digest: Recovery and Cleanup, Overview,” NUREG/KM-0001, Revision 1, June 2016 [Available at nrc.gov]
[2] USNRC, “Three Mile Island Accident of 1979 Knowledge Management Digest: Recovery and Cleanup,” NUREG/KM-0001, Supplement 1, June 2016 [Available at nrc.gov]
[3] USNRC, “Three Mile Island Accident of 1979 Knowledge Management Digest: Recovery and Cleanup, Overview,” NUREG/KM-0001, June 2016 [Available at nrc.gov]
[4] (1982-02-02) TMI-2 Organization Plan Change
[5] USNRC, “Three Mile Island Accident of 1979 Knowledge Management Digest: The Cleanup Experience—A Literature Review,” NUREG/KM-0001, Supplement 2, December 2020 [Available at nrc.gov]
[6] NUREG-0683, Vol. 1, PEIS-Decontamination and Disposal of Radioactive Wastes Resulting from TMI-2 (1981-03)
[7] NUREG-0683, Vol. 2, PEIS-Decontamination and Disposal of Radioactive Wastes Resulting from TMI-2 (1981-03)
[8] NUREG-0698, Rev. 0, NRC Plan for Cleanup Operations at TMI-2 (1980-07)

[9] NUREG-0698, Rev. 1, NRC Plan for Cleanup Operations at TMI-2 (1982-02)

[10] NUREG-0698, Rev. 2, NRC Plan for Cleanup Operations at TMI-2 (1984-03)

[11] (1981-04-27) NRC Policy, Programmatic Environmental Impact Statement of the Cleanup of TMI-2 (46 FR 24764)

[12] USNRC, "Three Mile Island Accident of 1979 Knowledge Management Digest: Recovery and Cleanup," NUREG/KM-0001, Supplement 1, June 2016 [Available at nrc.gov]

[13] USNRC, "Three Mile Island Accident of 1979 Knowledge Management Digest: Recovery and Cleanup," NUREG/KM-0001, Supplement 1, June 2016 [Available at nrc.gov]

[14] USNRC, "Three Mile Island Accident of 1979 Knowledge Management Digest: Recovery and Cleanup," NUREG/KM-0001, Supplement 1, June 2016 [Available at nrc.gov]

[15] *U.S. Code of Federal Regulations*, "Standards for Protection Against Radiation," Part 20, Chapter I, Title 10, "Energy"

[16] *U.S. Code of Federal Regulations*, "Domestic Licensing of Production and Utilization Facilities," Part 50, Chapter I, Title 10, "Energy"

[17] *U.S. Code of Federal Regulations*, "General Design Criteria for Nuclear Power Plants," Appendix A, Part 50, "Domestic Licensing of Production and Utilization Facilities," Chapter I, Title 10, "Energy"

[18] *U.S. Code of Federal Regulations*, "Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion "As Low as is Reasonably Achievable" for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents," Appendix I, Part 50, "Domestic Licensing of Production and Utilization Facilities," Chapter I, Title 10, "Energy"

[19] *U.S. Code of Federal Regulations*, "Packaging and Transportation of Radioactive Material," Chapter I, Title 10, "Energy"

[20] *U.S. Code of Federal Regulations*, "Reactor Site Criteria," Part 100, Chapter I, Title 10, "Energy"

[21] USNRC, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Regulatory Guide 1.109, October 1977 [Current and past regulatory guides are available at nrc.gov]

[22] USNRC, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors," Regulatory Guide 1.111, July 1977 [Current and past regulatory guides are available at nrc.gov]

[23] USNRC, "Three Mile Island Accident of 1979 Knowledge Management Digest: Recovery and Cleanup," NUREG/KM-0001, Supplement 1, June 2016 [Available at nrc.gov]

2 CLEANUP ACTIVITIES

TABLE OF CONTENTS

2.1 Introduction

This chapter describes the purpose of each cleanup activity discussed in the following chapters. In addition, the applicable safety topics that were evaluated in the licensee's and NRC's safety evaluation reports are listed for each activity. The purpose of each activity is repeated in the following chapters, as applicable. However, to simplify the chapters, the lists of applicable safety topics are not repeated. The reader should refer to this chapter for the listing of safety topics for each activity.

2.2 Key Studies

This section, along with the following chapters, summarizes any key studies and technical reports associated with a safety topic. The cited safety evaluation reports usually reference these documents.

2.3 Data Collection Activities

2.3.1 Axial Power Shaping Rod Insertion Test

- ***Purpose***. To individually move the eight axial power shaping rod assemblies in the core to provide insight into the extent of core and upper plenum damage. This early insight allowed time to factor this information into plans for subsequent inspections for the head, upper plenum, and core removal.

- ***Evaluation***. [1] Safety topics considered in the licensee's safety evaluation report included: (●) criticality; (●) fire protection; (●) hydrogen; (●) occupational exposure; and (●) radiological release.

- ***NRC Review***. [2] The NRC's safety evaluation report focused on criticality.

2.3.2 Camera Insertion through Reactor Vessel Leadscrew Opening

- ***Purpose***. To insert a camera into the reactor vessel through a control rod drive mechanism leadscrew opening to provide the first visual assessment of conditions inside the reactor vessel. This activity was known as "Quick Look."

- ***Evaluation***. [3] Safety topics considered in the licensee's safety evaluation report included: (●) boron dilution; (●) criticality; (●) decay heat removal; (●) fire protection; (●) hydrogen; (●) occupational exposure; and (●) radiological release.

- ***NRC Review***. [4] Safety topics considered in the NRC's safety evaluation report included: (●) boron dilution; (●) criticality; (●) decay heat removal; (●) hydrogen; and (●) occupational exposure.

2.3.3 Reactor Vessel Underhead Characterization (Radiation Characterization)

- ***Purpose***. To characterize radiation under the reactor vessel head to ensure that adequate radiological protection measures would be taken to keep radiation exposures ALARA during head removal. To achieve this objective, measurements were necessary with the reactor vessel head still in place.

- ***Evaluation***. [5] Safety topics considered in the licensee's safety evaluation report included: (●) boron dilution; (●) criticality; (●) decay heat removal; (●) hydrogen; (●) occupational exposure (internal and external); (●) pyrophoricity; (●) radiation protection/ALARA; and (●) radiological release.

- ***NRC Review***. [6] Safety topics considered in the NRC's safety evaluation report included: (●) criticality; (●) decay heat removal; (●) hydrogen; (●) occupational exposure; (●) pyrophoricity; and (●) radiological release.

2.3.4 Reactor Vessel Underhead Characterization (Core Sampling)

- ***Purpose***. To obtain up to six specimens of the core debris using specially designed tools. These tools would be lowered into the reactor to extract samples of the loose debris and to load the samples into small, shielded casks for offsite shipment and analysis. The analysis of the samples would: (●) identify the composition of the particulate core debris; (●) determine particle size; (●) determine fission product content; (●) determine fission product leachability from the debris; (●) analyze the drying properties of the debris; and (●) determine whether pyrophoric materials existed in the core debris. This examination was a follow-on activity to the underhead characterization study.

- ***Background***. Three shipping casks [a] were considered [7] for transporting the core debris samples to the laboratory. The selected cask was the modified and recertified Model CNS 1-13C Type B shipping cask, with the sample in a Specification 2R container. An earlier version of this cask was used to ship spent submerged demineralizer system liners. [8]

- ***Evaluation***. [9] Safety topics considered in the licensee's safety evaluation report included: (●) boron dilution; (●) criticality; (●) decay heat removal; (●) hydrogen; (●) load drop; (●) occupational exposure; (●) pyrophoricity; and (●) radiation protection/ALARA.

---

[a] Editor's Note: While large shipping containers of radioactive materials may often be referred to as "shipping casks," the proper term for such containers, when loaded with contents and in their transportation configuration, is "package." See 10 CFR Part 71, "Packaging and Transportation of Radioactive Material," Section 71.4, "Definitions."

- ***NRC Review***. [10] Safety topics considered in the NRC's safety evaluation report included: (●) occupational exposure and (●) pyrophoricity.

2.3.5 Core Stratification Sample Acquisition

- ***Purpose***. To perform the activities associated with: (●) installation, operation, and removal of the core boring machine; (●) acquisition of the core samples; (●) transfer of the samples from the machine to the defueling canisters; and (●) viewing of the lower vessel region through the bored holes.

- ***Evaluation***. [11] Safety topics considered in the licensee's safety evaluation report (SER) included: (●) boron dilution; (●) criticality; (●) load drop; (●) occupational exposure (internal and external); (●) pyrophoricity; (●) radiation protection/ALARA; (●) radiological release; (●) reactor vessel integrity; and (●) shielding.

The SER was revised four times; this NUREG/KM presents the latest revision.

- ***NRC Review***. [12] Safety topics considered in the NRC's SER included: (●) criticality; (●) pyrophoricity; and (●) reactor vessel integrity.

2.3.6 Use of Metal Disintegration Machining System to Cut Reactor Vessel Lower Head Wall Samples

- ***Purpose***. To remove metallurgical samples from the inner surface of the lower reactor vessel head using the metal disintegration machining system. An international research program used these samples to study core melt and reactor vessel interactions. This program was conducted after the reactor vessel was defueled.

- ***Evaluation***. [13, 14] Safety topics considered in the licensee's safety evaluation report included: (●) boron dilution; (●) criticality; (●) decay heat removal; (●) fire protection; (●) hydrogen; (●) instrument interference; (●) load drop; (●) pyrophoricity; (●) radiation protection/ALARA; (●) radiological release; and (●) reactor vessel integrity.

- ***NRC Review***. [15] Safety topics considered in the NRC's safety evaluation report included: (●) boron dilution; (●) criticality; (●) decay heat removal; (●) fire protection; (●) hydrogen; (●) load drop; (●) pyrophoricity; (●) radiological release; and (●) reactor vessel integrity.

2.4 Pre-Defueling Preparations

2.4.1 Containment Building Decontamination and Dose Reduction Activities

Purpose. To conduct decontamination and dose reduction activities in the containment building at elevation levels 305 feet (entry level) and above. Planned activities included: (●) flushing containment building surfaces with deborated water; (●) scrubbing selected components of the polar crane; (●) conducting hands-on decontamination of vertical surfaces; (●) decontaminating the air coolers; (●) flushing the elevator pit; (●) cleaning floor drains; (●) shielding the seismic gap and penetration; (●) decontaminating and shielding hot spots; (●) decontaminating missile shielding; (●) shielding reactor vessel head surface structure; (●) removing concrete and paint; (●) decontaminating cable trays; (●) decontaminating equipment; (●) remote flushing the containment building basement; and (●) testing remote decontamination technology.

- ***Evaluation***. [16] Safety topics considered in the licensee's safety evaluation report (SER) included: (●) boron dilution; (●) criticality; (●) fire protection; (●) industrial safety; (●) load drop; (●) occupational exposure (internal and external); (●) radiation protection/ALARA; (●) radiological release; (●) shielding; (●) structural integrity; and (●) vital equipment protection.

This SER was updated three times to reflect dose reduction plans for the upcoming year. This NUREG/KM presents the latest revision for containment building decontamination and dose reduction activities for 1986.

- ***NRC Review***. [17] The NRC's review of the licensee's SER [18] included the following safety topics: (●) industrial safety; (●) occupational exposure; (●) radiation protection/ALARA; (●) radiological release; and (●) vital equipment protection.

2.4.2 Reactor Coolant System Refill

- ***Purpose***. To refill the reactor coolant system (RCS) to the top of the hot legs in order to purge oxygen and to provide an RCS water level that would permit operation of the once-through steam generator (OTSG) recirculation/cleanup system. To operate the OTSG recirculation/cleanup system, the secondary-side water level in the OTSG must be raised to the vicinity of the upper tubesheet to minimize the chance of unborated water leakage from the OTSGs to the RCS.

As an added measure of protection against system overpressurization, the pressurizer would not be vented. This protective measure provided a surge volume for increases to the RCS or for inadvertent introduction of pressurization to the RCS, such as by activating pumps or changing valve lineups.

- ***Evaluation***. [19, 20] Safety topics considered in the licensee's safety evaluation included: (●) hydrogen and (●) RCS integrity.

- ***NRC Review***. The NRC's safety evaluation was not located.

2.4.3 Reactor Vessel Head Removal Operations

2.4.3.1 Polar Crane Load Test

- ***Purpose***. To conduct load testing of the polar crane in preparation for reactor vessel head lift and removal. The polar crane load test required four major sequential activities: (1) relocation of the internals indexing fixture; (2) assembly of test load; (3) load test; and (4) disassembly of test load.

- ***Evaluation***. [(21)] Safety topics considered in the licensee's safety evaluation report (SER) included: (●) criticality; (●) decay heat removal; (●) load drop; (●) occupational exposure; (●) radiological release; (●) reactor coolant system integrity; and (●) shielding.

- ***NRC Review***. [(22)] The NRC's review of the licensee's SER [(23)] included the following safety topics: (●) criticality; (●) decay heat removal; (●) load drop; (●) occupational exposure; (●) radiological release; (●) reactor vessel integrity; and (●) vital equipment protection.

- ***NRC Review (Background)***. [(24)] The NRC recognized that the polar crane was an absolute prerequisite for major activities (i.e., reactor pressure vessel head lift and plenum removal) leading to the defueling of the damaged core. The refurbishment and requalification of the damaged crane were essential to further progress of the cleanup. Accordingly, the NRC developed guidance for the refurbishment of the crane and forwarded the criteria to the licensee by letter dated April 1, 1982. [(b)] The licensee submitted the functional description for the polar crane refurbishment and revisions following NRC reviews of the functional description reports. Almost a year later, the licensee submitted its SER [(25)] for the polar crane test to the NRC. Soon after, a licensee contractor employee made allegations about the safety of the polar crane and other cleanup-related issues. The NRC's extensive investigation of the allegations found that there were administrative and procedural deficiencies in the crane refurbishment program. The NRC requested additional information from the licensee to provide assurance that the refurbishment and testing of the crane would have proper management controls to ensure quality workmanship. The NRC met numerous times with the licensee throughout the review process, and the licensee submitted many responses to requests for additional information.

The NRC concluded that, notwithstanding the identified procedural deficiencies in the refurbishment of the polar crane, the program used to refurbish, test, and operationally verify a

---

[b] Editor's Note: The cited letter from the NRC could not be found in either public or nonpublic ADAMS. The SER restated the guidance: (1) resistance measurements should be taken to verify that no unacceptably or high resistances existed between the various circuits and circuits to ground, (2) the quantity and quality of lubricants should be checked and found acceptable, or a suitable replacement of the lubricant should be made, (3) due to the past potentially corrosive environment, a thorough inspection should be performed of the wire rope system of the 500-ton main hoist using the "Wire Rope User's Manual," which was published by the American Iron and Steel Institute as a guide, and (4) the checklist in Table 3.1 of the NRC letter should be used as a guide for a recommended inspection plan.

working crane was made technically sufficient and provided reasonable assurance that the crane was safe for the conduct of the requalification test. Further, Section 9 of the NRC's SER included a detailed discussion of the agency's evaluation of the licensee's quality assurance and quality control requirements and practices.

2.4.3.2 First Pass Stud Detensioning for Head Removal

- ***Purpose***. To perform the first-pass detensioning of the 60 reactor vessel studs and the removal of up to 5 reactor vessel studs to check for stuck nuts and to examine the condition of the removed studs.

- ***Evaluation***. [26, 27] Safety topics considered in the licensee's safety evaluation report included: (●) criticality; (●) occupational exposure; (●) radiological release; and (●) reactor coolant system integrity.

- ***NRC Review***. [28] The NRC's review of the licensee's safety evaluation report [29] and supplemental information [30] included the following safety topics: (●) occupational exposure; (●) radiological release; and (●) reactor coolant system integrity.

2.4.3.3 Reactor Vessel Head Removal Operations

- ***Purpose***. To prepare for reactor vessel head removal to perform the lift and transfer of the head and conduct the activities necessary to place and maintain the reactor coolant system (RCS) in a stable configuration for the next phase of the recovery effort.

- ***Evaluation***. [31] The removal of the reactor vessel head was the most complex operation in the containment building up to that point in time. The licensee's safety evaluation report (SER) contained over 180 pages and covered a wide range of safety considerations: (●) boron dilution; (●) criticality; (●) decay heat removal; (●) fire protection; (●) hydrogen; (●) load drop; (●) occupational exposure (internal and external); (●) pyrophoricity; (●) radiation protection/ALARA; (●) radiological release; (●) RCS integrity; and (●) shielding.

The SER was revised five times; this NUREG/KM presents the latest SER.

- ***NRC Review***. [32] The NRC's review of the licensee's SER [33] included the following safety topics: (●) boron dilution; (●) criticality; (●) decay heat removal; (●) fire protection; (●) hydrogen; (●) load drop; (●) occupational exposure; (●) pyrophoricity; (●) radiation protection/ALARA; (●) radiological release; (●) RCS integrity; and (●) vital equipment protection.

In addition, the NRC's SER included an extensive evaluation that considered the long-term safety of removing the reactor vessel head and the potential for future delays in cleanup activities from funding constraints or technical problems, such as a stuck plenum. For details, refer to Section 5 of the NRC's SER.

2.4.4 Heavy Load Handling inside Containment

- ***Purpose***. To ensure that handling of heavy loads in the containment building and spent fuel pool "A" within the fuel handling building was in accordance with the safety requirements of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," (34) issued July 1980.

- ***Evaluation***. (35) Safety topics considered in the licensee's safety evaluation report (SER) included: (●) boron dilution; (●) criticality; (●) decay heat removal; (●) load drop; (●) radiological release; (●) reactor vessel integrity; and (●) vital equipment protection.

The licensee's SER addressed the handling of heavy loads within the containment building and spent fuel pool "A" during defueling and described load handling areas and any necessary restrictions to be applied while handling these loads. This SER did not address loads in areas above the in-core instrument seal plate, the reactor vessel, and the northwest corner of the "A" D-ring. [c]

- ***NRC Review***. (36, 37) The NRC's review of the licensee's SER (38) included the following safety topics: (●) boron dilution; (●) criticality; (●) decay heat removal; (●) hydrogen; (●) load drop; (●) radiological release; (●) reactor vessel integrity; and (●) vital equipment protection.

The scope of the NRC's review included transfer within the containment building, transfer to the fuel handling building, and canister handling inside the fuel handling building but did not include transfers to shipping casks. The NRC stated that an additional SER was required for that activity.

2.4.5 Heavy Load Handling over the Reactor Vessel

- ***Purpose***. To permit the handling of heavy loads (greater than 2400 pounds including rigging weight) in the vicinity of the reactor vessel. This included any load handling activity that could result in a heavy load drop onto or into the vessel either directly or indirectly (as the result of the collapse of structures or equipment installed over the vessel).

- ***Evaluation***. (39) Safety topics considered in the licensee's safety evaluation report (SER) included: (●) criticality; (●) load drop; (●) occupational exposure; (●) radiological release; and (●) reactor vessel integrity.

This SER addressed all such load handling activities through to the completion of reactor vessel fuel removal activities but excluded removal of the core support assembly. In addition, this SER

---

[c] D-rings were shield enclosures around the steam generator compartments; they were so named because of their shape.

addressed the potential impact of heavy load handling activities on the integrity of the reactor coolant system; the SER did not address the potential damage to the item dropped or the consequences of that damage (e.g., this SER did not address damage to a dropped defueling canister and the consequences of canister damage).

Heavy load handling activities outside the area over the reactor vessel were addressed in the SER (40) for heavy load handling inside containment and the SER (41) for plenum lift and transfer.

- ***NRC Review***. (42) The NRC's review of the licensee's SER (43) included the following safety topics: (●) load drop; (●) radiological release; and (●) reactor vessel integrity.

2.4.6 Plenum Assembly Removal Preparatory Activities

- ***Purpose***. To confirm the adequacy of plenum removal equipment and techniques, the preparatory activities included: (●) video inspection of potential interference areas; (●) video inspection of the core void space; (●) video inspection of the axial power shaping rod (APSR) assemblies; (●) measurement of the loss-of-coolant accident restraint boss gaps; (●) measurement of the elevations of the APSR assemblies; (●) cleaning of the plenum and potential interference areas; (●) separation of unsupported fuel assembly end fittings; and (●) movement of the APSR assemblies.

- ***Evaluation***. (44) Safety topics considered in the licensee's safety evaluation report (SER) included: (●) boron dilution; (●) criticality; (●) load drop; (●) occupational exposure (internal and external); (●) radiation protection/ALARA; (●) radiological release; and (●) reactor vessel integrity.

The SER was revised once; this NUREG/KM presents the first SER. A meeting was held between the licensee and the NRC's onsite office to exchange information and comments. The NRC documented discussions and additional requests for information in a letter (45) to the licensee. The licensee responded with a revised SER, which could not be located.

- ***NRC Review***. (46, 47) The NRC's review of the licensee's original SER (48) included the following safety considerations: (●) boron dilution; (●) criticality; (●) decay heat removal; (●) load drop; (●) occupational exposure; (●) radiation protection/ALARA; and (●) radiological release.

The NRC's preliminary review (49) of the licensee's original SER approved the first five activities, as listed in the "Purpose" of this section. The NRC's safety evaluation stated that the remaining three activities were still under review. The licensee revised its SER based on further discussions and comments (50) from the NRC. The revised licensee's SER could not be located. The NRC gave final approval in its subsequent safety evaluation. (51)

2.4.7 Plenum Assembly Removal

- ***Purpose***. To remove the plenum assembly from the reactor vessel to gain access to the core region for defueling. The plenum was moved to the deep end of the fuel transfer canal that contained reactor coolant for shielding.

- ***Evaluation***. [52] Safety topics considered in the licensee's safety evaluation report (SER) included: (●) boron dilution; (●) load drop; (●) radiation protection/ALARA; (●) occupational exposure (internal and external); and (●) radiological release.

The SER was revised three times; this NUREG/KM presents the latest SER.

- ***NRC Review***. [53] The NRC's review of the licensee's SER [54] included the following safety topics: (●) boron dilution; (●) criticality; (●) load drop; (●) occupational exposure; (●) radiation protection/ALARA; (●) radiological release; and (●) reactor vessel integrity.

2.4.8 Makeup and Purification Demineralizer Resin Sampling

- ***Purpose***. To obtain resin samples from the two makeup and purification demineralizers. Resin samples were required to characterize the present resin conditions for the development of a technically sound resin removal and disposal program.

- ***Evaluation***. [55] Safety topics considered in the licensee's safety evaluation report included: (●) criticality; (●) hydrogen; (●) occupational exposure; and (●) radiological release.

- ***NRC Review***. The NRC's safety evaluation report was not located.

2.4.9 Makeup and Purification Demineralizer Cesium Elution

- ***Purpose***. To remove most of the radioactivity from the resins while they were in the demineralizers to the extent that standard resin sluice procedures could complete the task. The scope of this evaluation included only the first phase of a three-phase process for disposition of the makeup and purification of resins. This first phase included the rinse and elution of the demineralizer resins. The latter two phases would include the sluicing, removal, solidification or other packaging, and disposal of these resins. Separate safety evaluations would address the latter phases.

- ***Background***. During 1984 and 1985, the resins were flushed with an elution solution to reduce the cesium-137 content in the resins. The elution process removed about 790 curies of cesium-137 (68 percent of original inventory) from the "A" demineralizer vessel and about 3455 curies of cesium-137 (89 percent of inventory) from the "B" demineralizer vessel. The sluicing proved more difficult. Between October 1987 and September 1988, most of the resins were sluiced to a spent resin storage tank that had been modified to operate as a radioactive sediment separator. A variety of resin transfer methods were employed in 51 separate transfer operations. The "A" vessel was left essentially empty with 0.2 cubic feet of the initial 25 cubic feet remaining, while the "B" vessel contained about 7.1 cubic feet of agglomerated resins of the initial 25 cubic feet. Over 1 kilogram of residual fuel debris and about 1300 curies of radioactivity were transferred, solidified, and shipped for offsite waste burial. Details of these activities can be found in DOE [(56)] and Electric Power Research Institute [(57)] reports.

The safety evaluations could not be found for the sluicing, removal, solidification, packaging, and disposal of the resins and fuel debris.

- ***Evaluation***. [(58)] Safety topics considered in the licensee's safety evaluation report included: (●) criticality; (●) hydrogen; (●) radiation protection/ALARA; (●) radiological release; and (●) shielding.

The licensee reported results of elutions by letter [(59)] to the NRC.

- ***NRC Review***. [(60)] The NRC's review of the licensee's safety evaluation report [(61)] and supplemental information included the following safety topics: (●) boron dilution; (●) criticality; (●) hydrogen; (●) occupational exposure; (●) radiation protection/ALARA; (●) radiological release; and (●) shielding.

2.5 Defueling Tools and Systems

2.5.1 Internals Indexing Fixture Water Processing System

- ***Purpose***. To provide reactor coolant water processing capability during head removal until plenum removal. The internals indexing fixture (IIF) processing system was selected based on the limited reactor coolant processing capability in the drained-down condition and the desire to provide adequate water cleanup capacity to minimize radiation dose rates around the IIF.

- ***Evaluation***. [(62)] Safety topics considered in the licensee's safety evaluation report (SER) included: (●) boron dilution; (●) decay heat removal; (●) occupational exposure (internal and external); (●) radiation protection/ALARA; and (●) radiological release.

- ***NRC Review***. [(63)] The NRC's review of the licensee's SER [(64)] and supplemental information included the following safety topics: (●) boron dilution; (●) radiation protection/ALARA; and (●) reactor vessel integrity. In addition, the NRC reviewed the functional testing of the IIF water processing system and the associated safety controls (refer to the NRC's SER for details).

2.5.2 Defueling Water Cleanup

2.5.2.1 Defueling Water Cleanup System

- ***Purpose***. To remove organic carbon, radioactive ions, and particulate matter from the reactor vessel and fuel transfer canal/spent fuel pool "A" (FTC/SFP-A). The defueling water cleanup system (DWCS) actually comprised two independent subsystems (sometimes called trains): the reactor vessel cleanup system and the FTC/SFP-A cleanup system. The DWCS was totally contained within areas that had controlled ventilation and could be isolated. This limited the environmental impact of the system during normal operations, shutdown, or postulated accident conditions.

- ***Evaluation***. [65] Safety topics considered in the licensee's safety evaluation report (SER) included: (●) boron dilution; (●) criticality; (●) decay heat removal; (●) hydrogen; (●) load drop; (●) occupational exposure; and (●) radiological release. The SER also contained evaluations of postulated system failures that included: loss of power; loss of instrumentation/instrument air; filter media rupture in the filter canister; and system line break (refer to Sections 3.2.1 and 3.2.2 of the SER for details).

The SER was revised 12 times over the 5.5-year operational period; this NUREG/KM addresses Revision 12. This SER discussed operational improvements that were implemented because of the unexpected concentrations of suspended solids during the initial operation. These solids blinded the filters and reduced their performance. To alleviate this problem, the following modifications were made: (●) use of filter aids; (●) use of coagulants; (●) use of cartridge filters; (●) use of series filters and/or demineralizers; (●) cross-connection of reactor vessel cleanup pumps and reactor vessel filter train inlets; (●) use of modified knockout canisters as deep-bed filters; and (●) the installation of suction tubes taking suction much deeper than originally designed. Section 3.8 and Attachment 1 of the SER discussed these modifications.

- ***NRC Review***. [66] The NRC's review of previous licensee SERs [67, 68, 69, 70] for the DWCS included the following safety topics: (●) criticality; (●) load drop; (●) occupational exposure; and (●) radiological release.

The NRC's review was based on Revision 6 of the SER. Reviews of subsequent revisions were minor. In addition, the NRC reviews consisted of evaluations of the design, installation, and testing of the DWCS (refer to the NRC SER for details).

2.5.2.2 Cross-Connect to Reactor Vessel Cleanup System

- ***Purpose***. To modify the fuel transfer canal/spent fuel pool "A" (FTC/SFP-A) cleanup system portion of the defueling water cleanup system (DWCS) to allow processing of the FTC/SFP-A water through the "B" train of the DWCS reactor vessel cleanup system. The purpose of this modification was to provide the capability to effectively process the FTC/SFP-A water in a manner similar to the reactor vessel cleanup process without the installation of additional body-feed and coagulant equipment in the fuel handling building. In addition, the proposed modification would authorize the use of FTC/SFP-A filtered effluent as a water source for the body-feed tank and as dilution water for the coagulant addition unit.

- ***Evaluation***. [71] Safety topics considered in the licensee's safety evaluation report included: (●) criticality and (●) reactor coolant system integrity (leaks).

- ***NRC Review***. [72] The NRC's review of the licensee's safety evaluation report [73] included the following safety topics: (●) boron dilution; (●) criticality; and (●) reactor coolant system integrity (leaks).

2.5.2.3 Temporary Reactor Vessel Filtration System

- ***Purpose***. To restore and maintain the visibility in the reactor vessel to acceptable levels to ensure the continuation of the early defueling operations. Operation of the defueling water cleanup system revealed that a differential pressure across its filter canisters would increase rapidly as the result of microorganism growth in the reactor coolant. Consequently, the defueling water cleanup system was able to process only a relatively small amount of reactor coolant before the maximum design pressure was reached and the filter canister had to be replaced. These developments created the need to design and operate a temporary filter system while a permanent program to control this phenomenon was being developed.

- ***Background***. Safety topics considered in the licensee's safety evaluation report and subsequent revisions included: (●) criticality; (●) occupational exposure; and (●) radiological release.

The safety evaluation report was revised three times to update calculations relating dose rates to fuel content of the temporary reactor vessel filtration system (TRVFS). This NUREG/KM presents the latest revision.

- *Revision 1:* [74] This revision reflected a change to the calculations relating dose rates to fuel content of the TRVFS based on a more realistic and conservative approach. The resulting dose rate at the filter housing, with the revised assumptions, was about 340 roentgens per hour. Assuming a radiation alarm setpoint of 3 roentgens per hour on the filter housing monitor, the result was a safety margin of about 110. A possible increase of the radiation alarm setpoint was proposed as an ALARA measure since, at that time, the filter was being replaced based on radiation levels as opposed to differential pressure. The frequent filter replacements were adding significantly to the exposure for defueling operations. The changes in the proposal would limit personnel doses based on operating experiences.

- *Revision 2:* [75] This revision reflected the use of a new larger filter vessel that operated at higher flow rates and had the potential for accumulation of a larger quantity of fuel debris in the filter media. The use of a defueling knockout canister as a receptacle of the discharged filter media was also included. The safety evaluation re-analyzed the operation of the existing TRVFS for use with a new filter vessel and residue canister.

- *Revision 3:* [76] This revision reflected the proposed use of filter canisters and knockout canisters as receptacles for discharged diatomaceous earth, fuel debris, and backwash water. The revision reflected the proposed allowance of deeper suction within the reactor vessel. The safety evaluation reanalyzed the operation of the existing TRVFS at lower depths within the reactor vessel and the use of filter canisters as receptacles for TRVFS filter backwash.

- ***NRC Review***. [77, 78, 79] The NRC's review of the licensee's SERs [80, 81, 82, 83, 84] included the following safety topics: (●) criticality and (●) occupational exposure.

2.5.2.4 Filter-Aid Feed System and Use of Diatomaceous Earth as Feed Material

- ***Purpose***. To add a feed material into the filter canisters to promote the buildup of cake on the filter media, thereby significantly improving the performance of the defueling water cleanup system filter canisters. A filter-aid feed system that used diatomaceous earth as the feed material was installed as an ancillary system to the defueling water cleanup system.

- ***Evaluation***. [85] Safety topics considered in the licensee's safety evaluation report included: (●) criticality and (●) hydrogen.

- ***NRC Review***. [86] The NRC's review of the licensee's safety evaluation report [87] and supplemental information [88, 89] included the following safety topics: (●) criticality; (●) hydrogen; and (●) radiological release.

2.5.2.5 Use of Coagulants

- ***Purpose***. To demonstrate the use of coagulants and body-feed material to improve the performance of the defueling water cleanup system (DWCS) filter canisters in maintaining water clarity. Operating experience with the DWCS had not achieved the desired clarity in the reactor coolant system (RCS) water to support defueling operations within the reactor vessel. The DWCS filters required changeout because of high differential pressure without the expected high filter throughput. The root cause of shortened filter canister life was expected to be the presence of hydrated metallic oxides in colloidal suspension within the RCS that were plugging the filter media. The addition of the coagulant with body-feed was expected to agglomerate the colloids to filterable sizes, thus forming a filter cake on the filter media.

- ***Background***. The first coagulant evaluated [90] was about 20 weight percent of $C_8H_{16}NCl$ and 80 weight percent unborated water when undiluted. The body-feed material was diatomaceous earth. The polymer melamine-formaldehyde was the second alternate coagulant that was evaluated [91] and showed a greater potential as a filter aid. The undiluted solution of this coagulant was about 8 percent of the polymer and 92 percent unborated water. The expected dosage of the undiluted solution to the DWCS processing stream was 10 to 20 parts per million (ppm) with a maximum dosage of 50 ppm. Both safety evaluations were similar; this NUREG/KM discusses the first evaluation.

- ***Evaluation***. [92,93] Safety topics considered in the licensee's safety evaluation report included (●) criticality; (●) hydrogen; (●) radiological release; and (●) RCS integrity (chemistry control).

- ***NRC Review***. [94] The NRC's review of the licensee's safety evaluation report [95] focused on criticality.

2.5.2.6 Filter Canister Media Modification

- ***Purpose***. To reduce the potential for clogging of the filter canister's filter elements by increasing the pore size of the filter bundle in some of the filter canisters from 0.5-micron nominal size (2 microns absolute) to 16-micron nominal size (25 microns absolute).

- ***Evaluation***. [96] The licensee's safety evaluation report focused on criticality.

- ***NRC Review***. The NRC's safety evaluation report was not located.

2.5.2.7 Addition of a Biocide to the Reactor Coolant System

- ***Purpose***. To evaluate the effects of adding hydrogen peroxide as a biocide to the reactor coolant system (RCS) to aid the removal of biological contamination. This biological growth reduced water clarity and fouled the water processing system filters.

- ***Evaluation***. [97] Safety topics considered in the licensee's safety evaluation report included: (●) boron dilution; (●) occupational exposure; (●) radiological release; and (●) RCS integrity (chemistry compatibility).

- ***NRC Review***. [98] The NRC's review of the licensee's safety evaluation report [99] included the following safety topics: (●) boron dilution; (●) criticality; (●) hydrogen; (●) occupational exposure; and (●) RCS integrity (chemistry compatibility).

2.5.3 Defueling Canisters and Operations

The defueling canisters were designed to accept and confine core debris ranging in size from particles (known as fines) of about 0.5 micron in diameter to partial-length fuel assemblies of full cross section. The canisters were intended to provide confinement for offsite transport, using a shipping cask, and long-term storage of core debris. Three types of defueling canisters were designed and fabricated: a fuel canister, knockout canister, and filter canister. Each canister required fixed neutron-absorber material for criticality control; catalytic recombiners to control the concentration of combustible gas mixtures generated from radiolytic decomposition of water; and appropriate process connections for filling, closing, dewatering, inerting, and monitoring. All three canisters were 150 inches long, 14 inches in diameter, and 0.25 inch thick.

2.5.3.1 Defueling Canisters: Filter, Knockout, and Fuel

- ***Purpose***. To provide loading, handling, and storage of the canisters (filter, knockout, and fuel) for the long-term storage of core debris, ranging from very small fines to partial length fuel assemblies.

- ***Evaluation***. [100] Safety topics considered in the licensee's safety evaluation report (SER) included: (●) criticality; (●) hydrogen; (●) load drop; and (●) occupational exposure. The SER also included an extensive evaluation of the canister structural integrity during normal operations (refer to Section 3.1.1 of the SER for details).

The SER was revised four times; this NUREG/KM presents Revision 4 of the SER.

- ***NRC Review***. [101, 102, 103] The NRC's reviews of the licensee's SER and subsequent revisions [104, 105, 106, 107, 108] included the following safety topics: (●) criticality; (●) hydrogen; and (●) load drop.

The NRC review provided reasonable assurance that the canisters, if fabricated in accordance with the design specifications, were capable of performing their intended function. [d]

2.5.3.2 Replacement of Loaded Fuel Canister Head Gaskets

- ***Purpose***. To replace head gaskets on the loaded fuel canisters in spent fuel pool "A" because of excessive leakage. The replacement of gaskets involved: (●) moving a loaded canister from the spent fuel pool "A" storage rack to the dewatering station rack with the canister handling bridge; (●) removing the canister head; and (●) transferring the head to a worktable. At the worktable: (●) the head was rotated for access to the gaskets; (●) the metallic gaskets were

[d] Editor's Note: NRC inspections of one of the licensee's canister fabricators, as well as the licensee's audits and surveillance of the vendor, identified significant deficiencies in the implementation of the vendor's quality assurance program. These noted deficiencies cast doubt on whether equipment provided by this vendor met required design specifications and, accordingly, whether the equipment was suitable for use during defueling. The licensee and others implemented a program involving an extraordinary level of quality assurance oversight in an attempt to correct the deficiencies and to verify the canister conformance to the design specifications. The NRC's approval of the use of the canisters would be contingent on the agency's determination that there was reasonable assurance that the canisters met all design specifications.

removed; (●) new synthetic gaskets were installed; and (●) the head was inspected before reinstallation.

- ***Evaluation***. (109) Safety topics considered in the licensee's safety evaluation report included: (●) criticality; (●) load drop; (●) hydrogen; (●) occupational exposure; (●) radiation protection/ALARA; and (●) radiological release.

- ***NRC Review***. (110) The NRC's review of the licensee's safety evaluation report (111) included the following safety topics: (●) criticality; (●) load drop; and (●) radiation protection/ALARA.

2.5.3.3 Use of Debris Containers for Removing End Fittings

- ***Purpose***. To use modified fuel canisters as "debris containers" for removing fuel assembly upper end fittings, control component spiders, or other structural material from the reactor vessel. This activity was performed to expedite access to the vacuumable fuel and debris in the core. The modified fuel canister did not have internal neutron-absorbing plates, concrete filler, recombiner catalyst, dewatering capability, or a relief valve. After the debris containers were loaded, they would be closed and stored in the spent fuel pool "A" racks until final dispositioning of the containers and their contents. There were no plans to use these debris containers for shipment. Since these canisters would not have relief valves installed (a prerequisite for shipping), they could be easily identified.

- ***Evaluation***. (112) Safety topics considered in the licensee's safety evaluation report (SER) included: (●) criticality; (●) hydrogen; (●) load drop; and (●) radiological release.

- ***NRC Review***. (113) The NRC's SER did not identify any specific safety considerations. The NRC's review concurred with the licensee's assessment that the safety consequences of the proposed activity were bounded by the previously approved technical evaluation report for the defueling canister and SER for early defueling.

2.5.3.4 Fuel Canister Storage Racks

- ***Purpose***. To provide storage for the three different types of canisters (fuel, filter, and knockout) filled with debris material from the reactor vessel. Storage for 263 canisters was available in the racks located in spent fuel pool "A" and in the deep end of the fuel transfer canal.

- ***Evaluation***. (114) Safety topics considered in the licensee's safety evaluation report (SER) included: (●) criticality; (●) decay heat removal; (●) load drop; and (●) occupational exposure.

The SER also included evaluations of structural and seismic analyses. Refer to Section 3.1 of the SER for details. Revision 1 of the SER reflected the final design of the fuel canister storage racks, which contained corrections in the equations for determining the allowable lift height of loads handled over the racks.

- ***NRC Review***. [115,116] The NRC's review of the licensee's SER [117] and subsequent revision [118] included the following safety topics: (●) criticality; (●) load drop; and (●) occupational exposure. The safety review also included evaluations of structural and seismic analyses (refer to the NRC's SER for details).

2.5.3.5 Canister Handling and Preparation for Shipment

- ***Purpose***. To transfer defueling canisters from spent fuel pool "A" to the shipping cask for offsite shipment. Defueling canisters were transferred to locations within the containment building and fuel handling building using a transfer shield. The transfer of canisters to the shipping cask used a different device called a "fuel transfer cask."

- ***Evaluation***. [119] Safety topics considered in the licensee's safety evaluation report (SER) included: (●) criticality; (●) fire protection; (●) load drop; (●) occupational exposure; (●) radiation protection/ALARA; (●) radiological release; (●) seismic hazard; and (●) Unit 1 impact.

The licensee's SER was revised six times; this NUREG/KM presents the licensee's evaluation in Revision 4.

- ***NRC Review***. [120, 121] The NRC's review of the licensee's SERs [122, 123, 124, 125] and supplemental information [126, 127, 128] for the canister handling and preparation for shipment included the following safety topics: (●) hydrogen; (●) load drop; (●) occupational exposure; and (●) radiological release.

The licensee's SER was revised six times; this NUREG/KM presents NRC reviews of the first five revisions. The NRC's safety evaluation for Revision 6 of the SER was not located. Revision 6 removed the restriction on the use of borated water for spraying canisters during transfers from spent fuel pool "A."

2.5.3.6 Canister Dewatering System

- ***Purpose***. To remove and filter the water from submerged defueling canisters and to provide a transfer path to the defueling water cleanup system for processing. The dewatering system also provided the cover gas for canister shipping.

- ***Evaluation***. The licensee's safety evaluation of the canister dewatering system was provided in the safety evaluation reports [129, 130] for canister handling and preparation for shipment.

- ***NRC Review***. The NRC's safety evaluation of the canister dewatering system was provided in the safety evaluation reports [131, 132] for canister handling and preparation for shipment.

2.5.3.7 Use of Nonborated Water in Canister Loading Decontamination System

- ***Purpose***. To use nonborated water for canister decontamination before shipment in order to stabilize the boron concentration in the fuel transfer canal/spent fuel pool "A". Boron concentration in spent fuel pool "A" was increased by adding borated water and by water evaporation.

- ***Evaluation***. [133] Safety topics considered in the licensee's safety evaluation report (SER) focused on (●) boron dilution and (●) criticality.

The SER was revised once; this NUREG/KM presents the latest revision.

- ***NRC Review***. [134] The NRC's review of the licensee's SER [135] and supplemental information [136] focused on (●) boron dilution and (●) criticality.

2.5.4 Testing of Core Region Defueling Techniques

- ***Purpose***. To use hydraulic heavy-duty defueling tools for limited bulk defueling operations on the hard crust layer of the damaged core.

- ***Evaluation***. [137] Safety topics considered in the licensee's safety evaluation report (SER) included: (●) radiological release and (●) reactor vessel integrity.

Certain activities associated with the proposed bulk defueling activities were not previously evaluated in approved SERs. These activities included the breaking up of the hard crust layer of the core, the removal of partial fuel assemblies, and the potential for inadvertent pulling of an in-core instrument string. Revision 4 of the SER [138] for early defueling previously addressed use of the light-duty tong tool and the light-duty spade bucket. The SER [139] for the use of the hydraulic impact chisel to separate fused material addressed limited use of the hydraulic impact chisel.

- ***NRC Review***. [140] The NRC's review of the licensee's SERs [141, 142, 143] included the following safety topics: (●) boron dilution; (●) criticality; (●) pyrophoricity; (●) radiological release; and (●) reactor vessel integrity.

2.5.5 Fines/Debris Vacuum System

- ***Purpose***. To modify the fines/debris vacuum system using a knockout canister and a filter canister in series. Modifications included: (●) use of a vacuum nozzle to allow larger debris particles to be vacuumed into the knockout canisters; (●) use of mechanical probes and water jets on the end of the vacuum nozzle to loosen the packed rubble; (●) use of a larger vacuum tool to allow debris removal from the lower head; and (●) temporary use of the vacuum system without a filter canister. The initial use of the fines/debris vacuum system was previously approved in the safety evaluation report (144, 145) for early defueling.

- ***Evaluation***. (146) Safety topics considered in the licensee's SER included: (●) boron dilution; (●) criticality; (●) decay heat removal; (●) occupational exposure; and (●) reactor vessel integrity.

- ***NRC Review***. (147) The NRC's review of the licensee's SER (148) and supplemental information (149, 150) included the following safety topics: (●) boron dilution; (●) criticality; (●) occupational exposure; and (●) reactor vessel integrity.

2.5.6 Hydraulic Shredder

- ***Purpose***. To use a hydraulically powered shredder to reduce the size of fuel pins and other core debris and to facilitate the loading of fuel canisters or debris buckets.

- ***Evaluation***. (151) Safety topics considered in the licensee's safety evaluation report included: (●) boron dilution; (●) load drop; (●) occupational exposure; (●) pyrophoricity; and (●) reactor vessel integrity.

- ***NRC Review***. (152) The NRC's safety evaluation report did not identify any specific safety considerations. In its approval letter, the NRC concurred with the assessment that safety issues associated with the use of the tool were bounded by the previously approved early defueling safety evaluation and concluded that the proposed activity did not constitute an unreviewed safety question.

2.5.7 Plasma Arc Torch

2.5.7.1 Use of Plasma Arc Torch to Cut Upper End Fittings

- ***Purpose***. Use of a plasma arc torch to cut the upper end fittings inside the reactor vessel to allow the fittings to be placed directly in fuel canisters for transfer from the reactor vessel to the fuel canal. This was the first use of the plasma arc torch inside the reactor vessel.

- ***Evaluation***. (153) Safety topics considered in the licensee's safety evaluation report included: (●) decay heat removal; (●) flammable gas; (●) industrial safety; (●) instrument interference; (●) pyrophoricity; (●) radiological release; and (●) reactor vessel integrity.

- ***NRC Review***. [154] The NRC's review of the licensee's safety evaluation report [155] and supplemental information [156, 157, 158, 159] included the following safety topics: (●) industrial safety and (●) instrument interference.

2.5.7.2 Use of Plasma Arc Torch to Cut the Lower Core Support Assembly

- ***Purpose***. To use the plasma arc torch to cut the lower core support assembly (LCSA), including the flow distributor head.

- ***Evaluation***. [160,161] The focus of the licensee's safety evaluation report (SER) included: (●) boron dilution and (●) criticality. Other safety topics were addressed as part of the SER [162] for defueling of the LCSA.

The licensee's SER was revised once; this NUREG/KM addresses both the original and the revision.

- ***NRC Review***. [163] NRC reviews of both of the licensee's safety evaluations were addressed under the NRC's SER [164] for the defueling of the LCSA.

2.5.7.3 Use of Plasma Arc Torch to Cut Baffle Plates and Core Support Shield

- ***Purpose***. To use the plasma arc torch to cut the upper core support assembly baffle plates and the core support shield and to increase the maximum allowable drainable volume for the plasma arc torch coolant system from 3.0 to 3.5 gallons. Cutting the baffle plates was required to gain access to the core debris located behind the baffle plates in the core formers.

- ***Evaluation.*** [165, 166] The focus of the licensee's safety evaluation report (SER) was criticality. Other safety topics were addressed as part the licensee's SER [167] for defueling of the upper core support assembly.

The SER was revised twice; this NUREG/KM presents both revisions.

- ***NRC Review***. [168] The NRC's review of the licensee's SERs [169, 170] focused on criticality.

2.5.7.4 Use of Air as Secondary Gas for the Plasma Arc Torch

- ***Purpose***. To replace nitrogen gas with air as the secondary gas to improve plasma arc torch performance by achieving longer and more efficient cuts.

- ***Evaluation***. [171] Safety topics considered in the licensee's safety evaluation report included: (●) hydrogen and (●) industrial safety.

- ***NRC Review***. The NRC's safety evaluation report could not be located.

2.5.8 Use of Core Bore Machine for Dismantling Lower Core Support Assembly

- ***Purpose***. To use the core bore machine, in conjunction with the automatic cutting equipment system, to dismantle the lower core support assembly and facilitate defueling by providing access to the reactor vessel lower head.

- ***Evaluation***. [172] Safety topics considered in the licensee's safety evaluation report (SER) included: (●) criticality; (●) pyrophoricity; (●) radiological release; and (●) reactor vessel integrity.

The SER was revised once; this NUREG/KM presents the revised SER.

- ***NRC Review***. [173] The NRC's review of the licensee's SER [174] included the following safety topics: (●) criticality; (●) pyrophoricity; and (●) reactor vessel integrity.

2.5.9 Sediment Transfer and Processing Operations

- ***Purpose***. To collect sediment from tanks and sumps in the auxiliary and fuel handling buildings, and also from the containment building basement and sump, in order to transfer the sediment to the spent resin storage tanks and treat or process the sediment (for disposal).

- ***Evaluation***. [175] Safety topics considered in the licensee's safety evaluation report (SER) included: (●) criticality; (●) occupational exposure; (●) radiation protection/ALARA; (●) radiological release; and (●) intersystem interactions.

The SER was revised four times; this NUREG/KM presents the latest SER.

- ***NRC Review***. [176] The NRC's review of the licensee's SER [177] and supplemental information [178] included the following safety topics: (●) criticality and (●) radiological release.

The NRC's reviews of subsequent revisions of the SER did not change the original conclusions.

2.5.10 Pressurizer Spray Line Defueling System

- ***Purpose***. To flush fuel fines and core debris from the pressurizer spray line to the pressurizer vessel and the reactor coolant system cold-leg loop 2A. The source of flush water for the pressurizer spray line defueling system was the defueling water cleanup system. Defueling consisted of flushing the pressurizer spray line in a series of steps to adequately remove fuel fines and debris in each different flowpath from the spray line tie-in.

- ***Evaluation***. [179] Safety topics considered in the licensee's safety evaluation report (SER) included: (●) boron dilution; (●) criticality; (●) occupational exposure; and (●) reactor coolant system integrity (inventory loss).

The SER was revised twice; this NUREG/KM presents the latest SER.

- ***NRC Review***. [180] The NRC's review of the licensee's SER [181] and supplemental information [182] included the following safety topics: (●) boron dilution; (●) criticality; (●) occupational exposure; and (●) reactor coolant system integrity (inventory loss).

2.5.11 Decontamination Using Ultrahigh Pressure Water Flush

- ***Purpose***. To use ultrahigh pressure water flush at 20,000 to 55,000 pounds per square inch to remove surface coatings and surface contamination inside the containment building.

- ***Evaluation***. [183] Safety topics considered in the licensee's safety evaluation report included: (●) boron dilution; (●) criticality; (●) industrial safety; (●) load drop; (●) occupational exposure (internal and external); (●) radiation protection/ALARA; (●) radiological release; (●) reactor coolant system integrity; and (●) vital equipment protection.

- ***NRC Review***. [184] The NRC's review of the licensee's safety evaluation report [185] included the following safety topics: (●) boron dilution; (●) criticality; (●) structure/system interaction; (●) load drop; (●) radiation protection/ALARA; (●) radiological release; and (●) reactor coolant system integrity.

2.6 Evaluations for Defueling Operations

This section discusses criticality safety reviews of the various stages of defueling. The phases of defueling included the following:

Preliminary defueling—rearrangement of debris inside the reactor vessel

Early defueling—removal of small amounts of loose debris, structural material, and intact fuel assembly segments from the reactor vessel

Bulk defueling—removal of the remaining fuel and structural debris located in the original core volume and in other regions of the reactor vessel; use of the core bore machine to break apart solidified melt material down to the lower grid support structure

Lower core support assembly and lower head defueling—dismantling (cutting, drilling) and defueling of the lower core support assembly and partial defueling of the reactor vessel lower head, followed by the removal of the elliptical flow distributor, gusseted in-core guide tubes, and subsequent completion of the defueling of the lower head

Upper core support assembly defueling—cutting and moving the baffle plates to defuel the upper core support assembly

2.6.1 Preliminary Defueling

- ***Purpose***. To allow movement of debris within the reactor vessel, to allow installation of defueling equipment, and to identify core debris samples for later removal and analysis by INEL. Some rearrangement of debris in the reactor vessel was required before the actual removal of fuel. These activities included the loading of small pieces of core debris into debris baskets but did not include actual loading of fuel debris into fuel canisters.

- ***Evaluation***. [186] Safety topics considered in the licensee's safety evaluation report (SER) included: (●) criticality and (●) load drop. The request for preliminary defueling was submitted while the NRC was reviewing the proposal for early defueling.

- ***NRC Review***. [187] The NRC's review of the licensee's SER [188] included the following safety topics: (●) boron dilution; (●) criticality; (●) decay heat removal; (●) load drop; (●) radiological release; and (●) reactor vessel integrity.

The safety issues related to the movement of material in the reactor vessel were identified and addressed in the NRC's previous SERs [189, 190, 191., 192] for head lift; plenum removal preparatory activities; plenum assembly lift and transfer; and heavy load handling over the reactor vessel. This safety evaluation, in part, summarized the conclusions of these earlier NRC safety evaluations as they applied to the proposed preliminary defueling activities.

2.6.2 Early Defueling

- ***Purpose***. To remove end fittings, structural materials, and related loose debris that would not involve removal of significant amounts of fuel material, to remove intact segments of fuel rods and other pieces of core debris, and to remove loose fuel fines (particles) by vacuuming operations.

- ***Evaluation***. [193] Safety topics considered in the licensee's safety evaluation report (SER) included: (●) boron dilution; (●) criticality; (●) decay heat removal; (●) fire protection; (●) hydrogen; (●) Unit 1 impact; (●) load drop; (●) occupational exposure; (●) pyrophoricity; (●) radiation protection/ALARA; (●) radiological release; and (●) shielding.

The evaluations of safety topics documented in the subsequent SER [194, 195] for bulk defueling were practically identical to this SER. [e] With one exception (occupational exposure), the reader will be referred to the SER for bulk defueling for the other safety topics.

- ***NRC Review***. [196] The NRC's review of the licensee's SER [197] included the following safety topics: (●) boron dilution; (●) criticality; (●) decay heat removal; (●) fire protection; (●) hydrogen; (●) load drop; (●) occupational exposure; (●) pyrophoricity; (●) radiation protection/ALARA; and (●) radiological release.

2.6.3 Storage of Upper End Fittings in an Array of 55 Gallon Drums

- ***Purpose***. To use shielded 55-gallon drums to store end fittings and other structural material removed from the reactor vessel.

- ***Evaluation***. [198, 199] Safety topics considered in the licensee's safety evaluation report (SER) included: (●) criticality; (●) hydrogen; (●) load drop; (●) occupational exposure; (●) pyrophoricity; and (●) radiological release.

- ***NRC Review***. [200, 201] The NRC's review of the licensee's SER [202] and supplemental information [203] focused on (●) criticality and (●) load drop.

The first NRC SER approved the loading and storage of a single drum in the 347-foot elevation storage area. A follow-on review approved loading and storage of an array of 55-gallon drums.

2.6.4 Defueling (Also Known as "Bulk" Defueling)

- ***Purpose***. To develop defueling tools and to conduct activities necessary to remove the remaining fuel and structural debris located in the original core volume. An additional activity was to address vacuuming of debris from the lower head of the vessel, a primary activity associated with lower head defueling.

- ***Evaluation***. [204, 205] Safety topics considered in the licensee's safety evaluation report (SER) included: (●) boron dilution; (●) criticality; (●) decay heat removal; (●) fire protection; (●) hydrogen; (●) Unit 1 impact; (●) load drop; (●) occupational exposure (internal and external);

[e] Editor's Note: Revision 4 of the licensee's SER for early defueling was subsequently revised to address the remaining defueling activities of the reactor core region. Revisions 5 through 9 were internal revisions, which were not issued for NRC review or use. Revision 10, which was submitted for NRC review and approval, included core region defueling tools and activities, lower head vacuuming, and the use of core bore equipment as a defueling tool. Revision 4 was called the "early defueling" SER, and Revision 10 was called the "defueling" or "bulk defueling" SER. Revision 10 became the baseline document, which was to be revised, as needed, to incorporate any additional defueling operations when those additional operations were identified. However, this SER was never revised. The safety evaluations of subsequent defueling activities following the completion of defueling the reactor core region were addressed in separate SERs.

(●) pyrophoricity; (●) radiation protection/ALARA; (●) radiological release; (●) reactor vessel integrity; and (●) shielding.

Revision 10 of the SER was the revision for early defueling (Revision 4). Revisions 5 through 9 were internal revisions that were not issued for NRC review or use. Revision 4 was called the “early defueling” SER; Revision 10 was called the “defueling” SER. The safety evaluations reported in Revision 10 were practically identical to those in Revision 4 with a few exceptions.

- ***NRC Review***. [206] The NRC’s review of the licensee’s SER [207] and supplemental information [208] included the following safety topics: (●) boron dilution; (●) criticality; (●) occupational exposure; (●) pyrophoricity; (●) radiation protection/ALARA; (●) radiological release; and (●) reactor vessel integrity. Other safety issues addressed in earlier NRC safety evaluations included: (●) decay heat removal; (●) fire protection; (●) hydrogen; and (●) load drop.

The NRC’s SER stated that the agency’s review and approval of the licensee’s proposal was limited to those activities to be conducted in the core region, specifically above the lower grid support structure. The remaining activities addressed in the licensee’s SER, including debris removal from the core support structure and lower vessel head regions, would be the subject of a separate safety evaluation.

2.6.5 Use of Core Bore Machine for Bulk Defueling

- ***Purpose***. To use the core stratification sample acquisition (core bore) tooling as a defueling tool so that other defueling tools could more effectively break up and remove the remaining core debris. The core bore tool used a solid-faced bit to perforate the hard crust region of the core, down to the lower grid support structure, at multiple locations. The defueling work platform orientation system was used to position the drill mechanism with restrictions.

- ***Evaluation***. [209] Safety topics considered in the licensee’s safety evaluation report included: (●) criticality; (●) pyrophoricity; and (●) reactor vessel integrity.

- ***NRC Review***. [210] The NRC review of the licensee’s proposal was documented in the agency’s safety evaluation report [211] for bulk defueling.

2.6.6 Lower Core Support Assembly Defueling

- ***Purpose***. To dismantle and defuel the lower core support assembly (LCSA) and to partially defuel the lower reactor vessel head. Structural material removed from the LCSA included the: (●) lower grid rib assembly; (●) lower grid forging; (●) distribution plate; and (●) in-core guide support plate. The lowest elliptical flow distributor and the gusseted in-core guide tubes would not be removed under this proposed activity. The use of the core bore machine, plasma arc cutting equipment, cavitating water jet, robot manipulators, automatic cutting equipment system, and other previously approved tools and equipment was included in this activity and associated safety evaluations.

- ***Evaluation***. [212] Safety topics considered in the licensee's safety evaluation report (SER) included: (●) boron dilution; (●) criticality; (●) decay heat removal; (●) fire protection; (●) hydrogen; (●) industrial safety (submerged combustion); (●) instrument interference; (●) Unit 1 impact; (●) load drop; (●) occupational exposure; (●) pyrophoricity; (●) radiological release; and (●) reactor vessel integrity.

The SER was revised twice; this NUREG/KM presents the latest SER. The use of the core bore machine to dismantle the LCSA was initially evaluated under separate SERs by both the licensee and the NRC. Revision 2 of the SER for LCSA defueling included the use of the core bore machine. In addition, a separate SER was later submitted for approval to complete LCSA defueling that encompassed the removal of the elliptical flow distributor, gusseted in-core guide tubes, and subsequent defueling of the reactor vessel lower head.

- ***NRC Review***. [213] The NRC's review of the licensee's SER [214] and supplemental information [215, 216, 217, 218] focused on criticality. The removal of any gusseted in-core guide tubes and the elliptical flow distributor was not included in the scope of this safety evaluation.

2.6.7 Completion of Lower Core Support Assembly and Lower Head Defueling

- ***Purpose***. To remove the elliptical flow distributor and gusseted in-core guide tubes of the lower core support assembly and to defuel the reactor vessel lower head.

- ***Evaluation***. [219, 220] Safety topics considered in the licensee's safety evaluation report (SER) included: (●) boron dilution; (●) criticality; (●) decay heat removal; (●) fire protection; (●) hydrogen; (●) industrial safety; (●) instrument interference; (●) Unit 1 impact; (●) load drop; (●) occupational exposure; (●) pyrophoricity; (●) radiological release; and (●) reactor vessel integrity.

This SER was revised once to update expected occupational exposure estimates based on the previous experience. Further, the licensee provided clarification [221] to a question from the NRC on the impact of accident melt jet impingement on the integrity of the reactor vessel lower head. This NUREG/KM presents both revisions and the clarification.

- ***NRC Review***. [222] The NRC's review of the licensee's SERs [223, 224] and supplemental information [225, 226, 227] included the following safety topics: (●) criticality; (●) load drop; and (●) reactor vessel integrity.

2.6.8 Upper Core Support Assembly Defueling

- ***Purpose***. To cut and move the baffle plates and to defuel the upper core support assembly. This evaluation addressed the following activities: (●) cutting the baffle plates for later removal; (●) removing bolts from the baffle plates; (●) removing the baffle plates; and (●) removing core debris from the baffle plates and core former plates.

- ***Evaluation***. [228] Safety topics considered in the licensee's safety evaluation report included: (●) boron dilution; (●) criticality (reactor vessel and containment building sump); (●) decay heat removal; (●) fire protection; (●) hydrogen; (●) industrial safety; (●) Unit 1 impact; (●) load drop; (●) occupational exposure; (●) pyrophoricity; (●) radiological release; and (●) reactor vessel integrity.

- ***NRC Review***. [229] The NRC's review of the licensee's safety evaluation report [230] for the upper core support assembly defueling included the following safety topics: (●) criticality; (●) load drop; and (●) pyrophoricity.

2.7 Evaluations for Waste Management

2.7.1 EPICOR II

- ***Purpose***. To decontaminate accident-generated, intermediate-level radioactive wastewater being held in tanks in the auxiliary building. Later, the system was used to polish effluents from the submerged demineralizer system during the cleanup of highly radioactive water from the containment building sump, reactor coolant system, and reactor coolant drain tanks. Following the decommissioning of the submerged demineralizer system, EPICOR II was used to clean residual wastewater from decontaminating the structures and systems.

- ***Background***. Planning for the EPICOR II system started 9 days following the accident and was completed about 7 weeks later. Before the system's use, the NRC Commissioners directed the staff in their policy statement [231] to prepare an environmental impact statement, which was completed [232] in August 1979. In October 1979, the NRC Commissioners directed [233] the NRC staff to order [234, 235] the licensee to operate the EPICOR II filtration and ion exchange decontamination system to decontaminate intermediate-level radioactive wastewater being held in tanks in the auxiliary building.

- ***Evaluation and NRC Review***. A traditional postaccident safety evaluation of the EPICOR II system was not submitted to the NRC. However, the NRC documented its formal review in NUREG-0591, "Environmental Assessment for Use of EPICOR II at Three Mile Island Unit 2," [236] issued October 1979. The PEIS [237], issued March 1981, documented an updated environmental assessment that applied to EPICOR II and all other cleanup activities.

Both environmental assessments included the following safety topics: (●) occupational exposure; (●) radiation protection/ALARA; (●) radiological release; and (●) shielding.

Given that these evaluations were conducted early in the cleanup when the safety evaluation and review processes were just being established, this NUREG/KM supplement does not summarize the safety evaluations. Refer to the EPICOR II environmental impact statement and PEIS for details of the NRC's safety reviews.

2.7.2 Submerged Demineralizer System

2.7.2.1 Submerged Demineralizer System Operations

- ***Purpose***. To decontaminate the containment building sump water and reactor coolant system (RCS) water using the submerged demineralizer system (SDS), followed by effluent polishing with the EPICOR II system.

- ***Evaluation***. [238] Safety topics considered in the licensee's safety evaluation report (SER) included: (●) industrial safety; (●) occupational exposure (normal operations); (●) radiation protection/ALARA; (●) radiological release (normal and accident conditions); and (●) nonradiological environmental effects.

- ***Evaluation: Background***. The safety evaluation for the SDS design and operations was documented in a technical evaluation report (TER). [f] Several revisions of this report were issued over the 6-year life of the SDS to account for changes in system configurations, water sources, and assumptions. The first revision of the TER, [239] submitted March 1981, documented the safety evaluation of the cleanup of containment building sump water and RCS water. The NRC evaluated the 1981 TER to support the NRC order [240] issued June 1981 to start the SDS operations in July 1981 [241] and to begin processing containment building sump water [242] in September. This TER and additional supplemental information, as requested by the NRC during its review, were the basis of the NRC's SER. The NRC issued the SER as NUREG-0796, "Operation of the Submerged Demineralizer System at Three Mile Island Unit No. 2," [243] in June 1981, and it remained applicable throughout the SDS lifetime. Although the 1981 TER and the NRC SER applied to both the processing of containment building sump water and RCS water, both evaluations were based on highly radioactive sump water as the source term, which was significantly lower than the RCS water. These safety evaluations, therefore, bounded the later processing of RCS water.

Subsequent revisions were submitted to the NRC included: (●) processing of RCS water without the use of EPICOR II (1982 [244]); (●) processing of RCS water after depressurization and draindown (1982 [245]); and (●) processing of RCS water with the use of EPICOR II to filter increased levels of antimony-125 (1984 [246]). Other annual revisions followed. [g, 247] Subsequent NRC evaluations were minor compared to the 1981 SER. The TER revisions

---

[f] The licensee's safety evaluation of a system design was typically documented in what is called a "technical evaluation report."

[g] The NRC issued a revision to section 10 CFR 50.71, "Maintenance of records, making of reports," in 1981 that required all licensees to review their final safety analysis report annually and update as needed. The NRC permitted TMI-2 to use TERs and system descriptions as an alternative method for documenting changes to the facility and providing associated safety evaluations.

provided simplified discussions of the evaluations based on actual SDS operating experience instead of preoperational assumptions. This experience revealed that original assumptions were conservative and radiological consequences were much lower. Also, RCS water was less contaminated than sump water.

The safety evaluations presented in this NUREG/KM were from the licensee's 1981 TER and the NRC's 1981 SER.

- ***NRC Review***. [248] Safety topics considered in the licensee's TER included: (●) occupational exposure (normal operations); (●) radiation protection/ALARA; and (●) radiological release (normal and accident conditions).

- ***NRC Review: Background***. The NRC documented its safety evaluation of the SDS and operations documented in NUREG-0796. [249] This report provided the NRC's evaluation of the licensee's request to decontaminate the containment building sump water and RCS water using the SDS, with effluent polishing by the EPICOR II system. The NRC's SER addressed only the processing of the containment building sump water and RCS water and did not consider the disposition of the process water.

This SER provided the safety basis for the NRC order [250] issued on June 18, 1981, to promptly commence and complete processing of the intermediate-level contaminated water in the auxiliary building tanks and the highly contaminated water in both the containment building sump and the RCS using the SDS with effluent polishing by the EPICOR II system, if necessary.

2.7.2.2 Submerged Demineralizer System Liner Recombiner and Vacuum Outgassing System

- ***Purpose***. To eliminate the potential of a combustible hydrogen and oxygen mixture existing in the submerged demineralizer system (SDS) liners and to facilitate the ultimate shipment and burial of the SDS liners. The liner recombiner and vacuum outgassing system was designed to remove moisture by evaporation from the zeolite beds of SDS spent liners. This operation dried the beds but did not remove the water in the zeolite.

- ***Evaluation***. [251, 252] Safety topics considered in the licensee's safety evaluation report included: (●) hydrogen; (●) radiation protection/ALARA; and (●) radiological release.

- ***NRC Review***. [253] The NRC's review of the licensee's safety evaluation report [254] and supplemental information [255] included the following safety topics: (●) hydrogen; (●) radiation protection/ALARA; and (●) radiological release.

2.8 Endnotes

Reference citations are exact filenames of documents on the Digital Versatile Discs (DVDs) to NUREG/KM-0001, Supplement 1, (256) "Three Mile Island Accident of 1979 Knowledge Management Digest: Recovery and Cleanup," issued June 2016. DVD document filenames start with a full date (YYYY-MM-DD) or end with a partial date (YYYY-DD). Citations for references that are not on a DVD begin with the author organization or name(s), with the document title in quotation marks. Contents of the DVDs are available at the Idaho National Laboratory Web site (https://tmi2kml.inl.gov).

---

[1] (1982-06-15) GPU Safety Evaluation, Axial Power Shaping Rod Insertion Test
[2] (1982-05-17) Order Amendment
[3] (1982-07-06) GPU Safety Evaluation, Insertion Camera Through Reactor Vessel Leadscrew Opening, Rev. 2
[4] (1982-07-13) NRC Review, Control Rod Drive Mechanism Quick Look Camera Inspection (re 07-06-1982) (2)
[5] (1983-05-19) GPU Safety Analysis, Radiation Characterization Under Reactor Vessel Head (No Polar Crane), Rev. 0
[6] (1983-07-13) NRC Review, Reactor Vessel Underhead Characterization (re Various Letters)
[7] (1983-07-20) GPU Safety Evaluation, Underhead Characterization, Core Sampling Addendum
[8] USNRC, "Three Mile Island Accident of 1979 Knowledge Management Digest: Recovery and Cleanup," NUREG/KM-0001, Supplement 1, June 2016 [Available at nrc.gov]
[9] (1983-07-20) GPU Safety Evaluation, Underhead Characterization, Core Sampling Addendum
[10] (1983-08-19) NRC Safety Evaluation, Addendum to the Underhead Characterization Study (re 07-30-1983)
[11] (1986-06-11) GPU Safety Evaluation, Core Stratification Sample Acquisition, Rev. 4
[12] (1986-05-05) NRC Safety Evaluation, Core Stratification Sample Acquisition SER, Rev. 1 (re 08-30-1985, 12-31-1985)
[13] (1989-08-18) GPU Safety Evaluation, Remove Metallurgical Samples from Reactor Vessel SER
[14] (1989-10-20) GPU Safety Evaluation, Remove Metallurgical Samples from Vessel, Rev. 1 (re 08-18-1989) (effective pages)
[15] (1989-11-28) NRC Safety Evaluation, Reactor Vessel Lower Head Metallurgical Sampling (re 08-18-1989)
[16] (1986-03-28) GPU Safety Evaluation, Reactor Building Decontamination and Dose Reduction Activities for 1986, Rev. 0
[17] (1984-02-03) NRC Safety Evaluation, Containment Decontamination and Dose Reduction (re 09-29, 01-23-1983)
[18] (1986-03-28) GPU Safety Evaluation, Reactor Building Decontamination and Dose Reduction Activities for 1986, Rev. 0
[19] (1982-11-22) GPU, Inform Plans to Refill RCS
[20] (1982-12-14) GPU, Changes in Plan to Vent Pressurizer During RCS Refill (re 11-22-1982)
[21] (1983-02-18) GPU Safety Evaluation, Polar Crane Load Test
[22] (1983-11-18) NRC Safety Evaluation, Reactor Building Polar Crane Load Test
[23] (1983-02-18) GPU Safety Evaluation, Polar Crane Load Test
[24] (1983-11-18) NRC Safety Evaluation, Reactor Building Polar Crane Load Test
[25] (1983-02-18) GPU Safety Evaluation, Polar Crane Load Test
[26] (1983-09-29) GPU, Plans for First Pass Stud Detensioning for Head Removal

[27] (1984-01-30) GPU Response to NRC, First Pass Stud Detensioning SER (re 09-29, 12-06-1983)
[28] (1984-02-17) NRC Safety Evaluation, First Pass Vessel Stud Detensioning (re 09-29, 12-06-1983, 01-30-1984)
[29] (1983-09-29) GPU, Plans for First Pass Stud Detensioning for Head Removal
[30] (1984-01-30) GPU Response to NRC, First Pass Stud Detensioning SER (re 09-29, 12-06-1983)
[31] (1984-03-09) GPU Safety Evaluation, Head Removal, Rev. 5
[32] (1984-07-17) NRC Safety Evaluation, Reactor Pressure Vessel Head Lift (re -3-09-1984)
[33] (1984-03-09) GPU Safety Evaluation, Head Removal, Rev. 5
[34] USNRC, "Control of Heavy Loads at Nuclear Power Plants," NUREG-0612, July 1980 [Available at nrc.gov]
[35] (1986-06-02) GPU Safety Evaluation, Heavy Load Handling Inside Containment, Rev. 3
[36] (1985-11-15) NRC Safety Evaluation, Heavy Load Handling Inside Containment (re 09-11-1985)
[37] (1987-07-23) NRC Review, Heavy Load Handling Inside Containment (re various letters)
[38] (1986-06-02) GPU Safety Evaluation, Heavy Load Handling Inside Containment, Rev. 3
[39] (1985-04-19) GPU Safety Evaluation, Heavy Load Handling Over Reactor Vessel SER
[40] (1984-11-01) GPU Safety Evaluation, Reactor Building Heavy Loads, Rev. 0
[41] (1985-01-25) GPU Safety Evaluation, Plenum Lift and Transfer SER, Rev. 1
[42] (1985-05-02) NRC Safety Evaluation, Heavy Load Handling Over the TMI-2 Reactor Vessel (re 04-19-1985)
[43] (1985-04-19) GPU Safety Evaluation, Heavy Load Handling Over Reactor Vessel SER
[44] (1984-06-18) GPU Safety Evaluation, Preparatory Activities for Plenum Assembly Removal
[45] (1984-08-15) NRC Review, Plenum Removal Preparatory Activities (re 06-18, 07-31, 01-05-1984)
[46] (1984-07-31) NRC Review, Preparatory Activities for Plenum Assembly Removal SER (re 06-18-1984)
[47] (1984-09-14) NRC Safety Evaluation, Plenum Assembly Removal (re various letters)
[48] (1984-06-18) GPU Safety Evaluation, Preparatory Activities for Plenum Assembly Removal
[49] (1984-07-31) NRC Review, Preparatory Activities for Plenum Assembly Removal SER (re 06-18-1984)
[50] (1984-08-15) NRC Review, Plenum Removal Preparatory Activities (re 06-18, 07-31, 01-05-1984)
[51] (1984-09-14) NRC Safety Evaluation, Plenum Assembly Removal (re various letters)
[52] (1985-05-01) GPU Safety Evaluation, Plenum Removal and Transfer, Rev. 3
[53] (1985-05-07) NRC Safety Evaluation, Lifting, Transfer, and Storage of Plenum Assembly (re various letters)
[54] (1985-05-01) GPU Safety Evaluation, Plenum Removal and Transfer, Rev. 3
[55] (1983-02-25) GPU Safety Evaluation, Makeup and Purification Demineralizer Resin Sampling
[56] GEND-039, TMI-2 Technical Information and Examination Program Annual Report 1983 (1984-04)
[57] Electric Power Research Institute, "TMI-2 Waste Management Experience," EPRI-TR-100640, April 1992, Palo Alto, California
[58] (1984-07-19) GPU Safety Evaluation, Makeup and Purification Cesium Elution SER
[59] (1986-01-28) GPU, Results of Makeup and Purification Demineralizers Elution Process and Status
[60] (1984-09-19) NRC Safety Evaluation, Makeup and Purification Demineralizer Cesium Elution (re 07-19, 08-29-1984)

[61] (1984-07-19) GPU Safety Evaluation, Makeup and Purification Cesium Elution SER
[62] (1984-05-31) GPU Safety Evaluation, Internals Indexing Fixture Processing System, Rev. 1
[63] (1984-07-24) NRC Safety Evaluation, Internals Index Fixture Water Processing System (re 05-31, 07-16-1984)
[64] (1984-05-31) GPU Safety Evaluation, Internals Indexing Fixture Processing System, Rev. 1
[65] (1989-05-12) GPU Technical Evaluation, Defueling Water Cleanup System, Rev. 12
[66] (1985-08-06) NRC Safety Evaluation, Defueling Water Cleanup System (re various letters)
[67] (1985-01-14) GPU Technical Evaluation, Defueling Water Cleanup System (with System Descriptions)
[68] (1985-04-26) GPU Technical Evaluation, Defueling Water Cleanup System, Rev. 4
[69] (1985-05-28) GPU Technical Evaluation, Defueling Water Cleanup System TER (effective pages)
[70] (1985-06-13) GPU Technical Evaluation, Defueling Water Cleanup System, Rev. 6 (effective pages only)
[71] (1987-05-21) GPU Safety Analysis, Defueling Water System Cross-Connect to Reactor Vessel Cleanup System, Rev. 0
[72] (1987-06-11) NRC Review, DWCS Cross-Connect to Reactor Vessel Cleanup System (re 05-21-1987)
[73] (1987-05-21) GPU Safety Analysis, Defueling Water System Cross-Connect to Reactor Vessel Cleanup System, Rev. 0
[74] (1986-03-06) GPU Safety Evaluation, Temporary Reactor Vessel Filtration System, Rev. 1
[75] (1986-04-14) GPU Safety Evaluation, Temporary Reactor Vessel Filtration System, Rev. 2
[76] (1986-11-17) GPU Safety Evaluation, TMI-2 Temporary Reactor Vessel Filtration System, Rev. 3
[77] (1986-02-07) NRC Safety Evaluation, Temporary Reactor Vessel Water Filtration System
[78] (1986-03-18) NRC Review, Restricted Operations of Temporary Vessel Filtration System (re various letters)
[79] (1986-04-18) NRC Review, Temporary Reactor Vessel Filtration System, Rev. 2 (re various letters)
[80] (1986-02-06) GPU Safety Evaluation, TMI-2 Temporary Reactor Vessel Water Filtration System
[81] (1986-03-06) GPU Safety Evaluation, Temporary Reactor Vessel Filtration System, Rev. 1
[82] (1986-04-14) GPU Safety Evaluation, Temporary Reactor Vessel Filtration System, Rev. 2
[83] (1986-11-17) GPU Safety Evaluation, TMI-2 Temporary Reactor Vessel Filtration System, Rev. 3
[84] (1986-12-01) GPU Safety Evaluation, TMI-2 Temporary Reactor Vessel Filtration System, Rev. 3 (Missing pages)
[85] (1986-06-30) GPU Safety Evaluation, Operation of Filter-Aid Feed System
[86] (1986-07-09) NRC Review, Operation of Filter-Aid Feed System, Use of Diatomaceous Earth (re various letter)
[87] (1986-06-30) GPU Safety Evaluation, Operation of Filter-Aid Feed System
[88] (1985-08-06) NRC Review, Defueling Water Cleanup System (re various)
[89] (1985-11-05) NRC Safety Evaluation, Defueling Canisters (re various letters)
[90] (1986-12-15) GPU Safety Evaluation, Addition of Coagulants to the Reactor Coolant System, Rev. 0
[91] (1986-12-31) GPU, Use of Different Coagulants (re 12-15-1986)
[92] (1986-12-15) GPU Safety Evaluation, Addition of Coagulants to the Reactor Coolant System, Rev. 0
[93] (1986-12-31) GPU, Use of Different Coagulants (re 12-15-1986)

[94] (1987-03-27) NRC Review, Criticality Safety Evaluation for Coagulants (re 12-11-1986, 02-20, 01-05-1987)
[95] (1987-02-20) GPU Safety Evaluation, Criticality Safety Evaluation for Coagulants (re 12-15, 12-31-1986)
[96] (1987-06-10) GPU, Filter Canister Media Modification
[97] (1986-05-05) GPU Safety Evaluation, Addition of a Biocide to the TMI-2 Reactor Coolant System, Rev. 0
[98] (1986-05-06) NRC Review, Addition of a Biocide to the TMI-2 Reactor Coolant System (re 05-05-1986)
[99] (1986-05-05) GPU Safety Evaluation, Addition of a Biocide to the TMI-2 Reactor Coolant System, Rev. 0
[100] (1987-09-22) GPU Technical Evaluation, Defueling Canister TER, Rev. 4
[101] (1985-11-05) NRC Safety Evaluation, Defueling Canisters (re various letters)
[102] (1986-08-12) NRC Review, Defueling Canister TER, Rev. 2 (re various letters)
[103] (1987-12-04) NRC Review, Defueling Canister TER (re various letters)
[104] (1985-04-09) GPU Technical Evaluation, Defueling Canisters, Rev. 0
[105] (1985-09-10) GPU Technical Evaluation, Defueling Canister TER (re 08-15-1985)
[106] (1986-03-06) GPU Technical Evaluation, Defueling Canister TER, Rev. 2
[107] (1987-05-07) GPU Technical Evaluation, Defueling Canisters, Rev. 3
[108] (1987-09-22) GPU Technical Evaluation, Defueling Canister TER, Rev. 4
[109] (1986-04-11) GPU Safety Evaluation, Replacement of Loaded Fuel Canister Head Gaskets
[110] (1986-04-29) NRC Review, Replacement of Loaded Fuel Canister Head Gaskets (re 04-11-1986)
[111] (1986-04-11) GPU Safety Evaluation, Replacement of Loaded Fuel Canister Head Gaskets
[112] (1986-01-17) GPU, Plan to Use Modified Debris Canisters for removing End Fittings, etc
[113] (1986-01-21) NRC Review, Plan to Use Modified Fuel Canisters to Remove End Fittings, etc (re 01-17-1986)
[114] (1986-05-29) GPU Technical Evaluation, Fuel Canister Storage Racks, Rev. 1
[115] (1985-05-03) NRC Review, Fuel Canister Storage Racks TER (re 02-27-1985)
[116] (1986-09-16) NRC Review, Fuel Canister Storage Racks TER (re various letters)
[117] (1985-02-27) GPU Technical Evaluation, Fuel Canister Storage Racks TER, Rev. 0
[118] (1986-05-29) GPU Technical Evaluation, Fuel Canister Storage Racks, Rev. 1
[119] (1987-04-02) GPU, Safety Evaluation, Canister Handling and Preparation for Shipment SER, Rev. 4 (Effective Pages Only)
[120] (1986-06-20) NRC Safety Evaluation, Canister Handling and Preparation for Shipment Program (re 02-17, 06-11-1986)
[121] (1988-03-03) NRC Review, Canister Handling and Preparation for Shipment Program TER (re various letters)
[122] (1986-02-17) GPU Safety Evaluation, Canister Handling and Preparation for Shipment
[123] (1986-06-11) GPU Safety Evaluation, Canister Handling and Preparation for Shipment SER
[124] (1987-04-02) GPU, Safety Evaluation, Canister Handling and Preparation for Shipment SER, Rev. 4 (Effective Pages Only)
[125] (1987-05-26) GPU Safety Evaluation, Canister Handling and Preparation for Shipment, Rev. 5 (Effective Pages Only)
[126] (1986-12-30) GPU, Additional Info on Alternative Dewatering Acceptance Criteria for Defueling Canisters
[127] (1987-01-07) NRC Review, Reduced Defueling Canister Void Volume (re 11-10, 12-30-1986)
[128] (1987-10-21) GPU, Plan to Discontinue Defueling Canister Gas Sampling at INL (re 01-07-1987)

[129] (1987-04-02) GPU, Safety Evaluation, Canister Handling and Preparation for Shipment SER, Rev. 4 (Effective Pages Only)
[130] (1987-05-26) GPU Safety Evaluation, Canister Handling and Preparation for Shipment, Rev. 5 (Effective Pages Only)
[131] (1986-06-20) NRC Safety Evaluation, Canister Handling and Preparation for Shipment Program (re 02-17, 06-11-1986)
[132] (1988-03-03) NRC Review, Canister Handling and Preparation for Shipment Program TER (re various letters)
[133] (1987-12-02) GPU Safety Evaluation, Use of Non-Borated Water in Canister Decontamination System, Rev. 1
[134] (1987-12-04) NRC Review, Use of Non-Borated Water in Canister Loading Decon System SER (re 12-02-1987, 08-12-1986)
[135] (1987-12-02) GPU Safety Evaluation, Use of Non-Borated Water in Canister Decontamination System, Rev. 1
[136] (1986-08-12) NRC Review, Defueling Canister TER, Rev. 2 (re various letters)
[137] (1986-06-02) GPU, Testing of Core Region Defueling Techniques
[138] (1985-10-10) GPU Safety Evaluation, Early Defueling of the TMI-2 Reactor Vessel, Rev. 4
[139] (1986-01-22) GPU, Plan to Use Hydraulic Impact Chisel to Separate Fused Material
[140] (1986-06-04) NRC Safety Evaluation, Heavy Duty Tong Tool, Spade Bucket, Hydraulic Impact Chisel (re various letters)
[141] (1986-06-02) GPU, Testing of Core Region Defueling Techniques
[142] (1986-05-15) GPU Safety Evaluation, Defueling SER, Rev. 10
[143] (1986-05-20) GPU Safety Evaluation, Defueling SER, Rev. 10 (Correction)
[144] (1985-10-10) GPU Safety Evaluation, Early Defueling of the TMI-2 Reactor Vessel, Rev. 4
[145] (1985-11-12) NRC Safety Evaluation, Early Defueling of the TMI-2 Reactor Vessel (10-10-1985)
[146] (1986-01-28) GPU, Fines-Debris Vacuum System Modifications (re various letters)
[147] (1986-02-21) NRC Review, Modification of Fines and Debris Vacuum System (re various letters)
[148] (1986-01-28) GPU, Fines-Debris Vacuum System Modifications (re various letters)
[149] (1985-11-05) NRC Safety Evaluation, Defueling Canisters (re various letters)
[150] (1985-11-12) NRC Safety Evaluation, Early Defueling of the TMI-2 Reactor Vessel (10-10-1985)
[151] (1986-03-26) GPU, Plan to Use Hydraulic Shredder to Reduce Core Debris
[152] (1986-04-16) NRC Review, Use of Hydraulic Shredder
[153] (1986-08-27) GPU, Plan to Use Plasma Arc Torch to Cut Upper End Fittings (re 08-18-1986)
[154] (1987-08-20) NRC Review, Use of Plasma Arc Torch to Cut Upper End Fittings (re various letters)
[155] (1986-08-27) GPU, Plan to Use Plasma Arc Torch to Cut Upper End Fittings (re 08-18-1986)
[156] (1987-01-20) GPU Response to NRC, Plasma Arc Cutting
[157] (1987-05-07) GPU Response to NRC, Plasma Arc Cutting (re 03-20-1987, 08-27-1986)
[158] (1987-06-25) GPU, Off-Gas Releases Generated During Plasma Arc Cutting (re 05-07-1987)
[159] (1987-06-25) GPU, Off-Gas Releases Generated During Plasma Arc Cutting (re 05-07-1987)
[160] (1987-11-30) GPU Criticality Assessment for Using Plasma Arc Torch to Cut Lower Core Support Assembly, Rev. 1
[161] (1988-03-16) GPU, Criticality Safety Assessment for Use of Plasma Arc Torch (re 11-30-1987)
[162] (1988-01-18) GPU Safety Evaluation, Lower Core Support Assembly Defueling, Rev. 2

[163] (1988-04-01) NRC Safety Evaluation, Lower Core Support Assembly Defueling (re 01-18-1988)
[164] (1988-04-01) NRC Safety Evaluation, Lower Core Support Assembly Defueling (re 01-18-1988)
[165] (1988-08-11) GPU, Criticality Assessment, Use of Plasma Arc Torch to Cut Baffle Plates and Core Support Shield
[166] (1988-12-27) GPU Criticality Assessment for Using Plasma Arc Torch to Cut Upper Core Support Assembly, Rev. 1
[167] (1988-09-15) GPU Safety Evaluation, Completion of Upper Support Core Assembly Defueling, Rev. 0
[168] (1989-01-31) NRC Safety Evaluation, Criticality Assess of Plasma Arc Torch to Cut Vessel Internals (08-11, 12-27-1988).pdf
[169] (1988-08-11) GPU, Criticality Assessment, Use of Plasma Arc Torch to Cut Baffle Plates and Core Support Shield
[170] (1988-12-27) GPU Criticality Assessment for Using Plasma Arc Torch to Cut Upper Core Support Assembly, Rev. 1
[171] (1989-02-08) GPU Safety Evaluation, Use of Air as Secondary Gas for Plasma Arc Torch
[172] (1987-12-28) GPU, Use of Core Bore Machine for Dismantling the Lower Core Support Assembly
[173] (1988-01-08) NRC Review, Use of Core Bore Machine for Dismantling Lower Core Support Assembly (re 12-03, 12-28-1987)
[174] (1987-12-28) GPU, Use of Core Bore Machine for Dismantling the Lower Core Support Assembly
[175] (1989-01-10) GPU Safety Evaluation, Sediment Transfer and Processing Operations, Rev. 4
[176] (1986-09-25) NRC Review, Sediment Transfer and Processing Operations (re 03-18-1985)
[177] (1989-01-10) GPU Safety Evaluation, Sediment Transfer and Processing Operations, Rev. 4
[178] (1986-09-11) NRC Review, Reactor Building Sump Criticality Safety Evaluation Report (re various letters)
[179] (1987-01-15) GPU Safety Analysis, Pressurizer Spray Line Defueling System, Rev. 2
[180] (1986-12-22) NRC Review, Proposal for Defueling the Pressurizer Spray Line (re 08-06-1985, 11-26-1986)
[181] (1986-12-22) NRC Review, Proposal for Defueling the Pressurizer Spray Line (re 08-06-1985, 11-26-1986)
[182] (1985-08-06) NRC Safety Evaluation, Defueling Water Cleanup System (re various letters)
[183] (1986-03-14) GPU Safety Evaluation, Decontamination Using Ultrahigh Pressure Water Flush, Rev. 0
[184] (1986-07-16) NRC Safety Evaluation, Decontamination using Ultrahigh Pressure Water Flush (re 03-14-1986)
[185] (1986-03-14) GPU Safety Evaluation, Decontamination Using Ultrahigh Pressure Water Flush, Rev. 0
[186] (1985-10-24) GPU, Commencement of Preliminary Defueling Operations
[187] (1985-10-29) NRC Safety Evaluation, Commencement of TMI-2 Preliminary Defueling Operations (re 10-24-1985)
[188] (1985-10-24) GPU, Commencement of Preliminary Defueling Operations
[189] (1984-07-17) NRC Safety Evaluation, Reactor Pressure Vessel Head Lift (re -3-09-1984)
[190] (1984-09-14) NRC Safety Evaluation, Plenum Assembly Removal (re various letters)
[191] (1985-05-07) NRC Safety Evaluation, Lifting, Transfer, and Storage of Plenum Assembly (re various letters)

[192] (1985-05-02) NRC Safety Evaluation, Heavy Load Handling Over the TMI-2 Reactor Vessel (re 04-19-1985)
[193] (1985-10-10) GPU Safety Evaluation, Early Defueling of the TMI-2 Reactor Vessel, Rev. 4
[194] (1986-05-15) GPU Safety Evaluation, Defueling SER, Rev. 10
[195] (1986-05-20) GPU Safety Evaluation, Defueling SER, Rev. 10 (Correction)
[196] (1985-11-12) NRC Safety Evaluation, Early Defueling of the TMI-2 Reactor Vessel (10-10-1985)
[197] (1985-10-10) GPU Safety Evaluation, Early Defueling of the TMI-2 Reactor Vessel, Rev. 4
[198] (1986-08-16) GPU, Storage of Upper End Fittings in Shielded 55 Gal Drums
[199] (1986-09-09) GPU Response to NRC (Discussion), Storage of Upper End Fittings (re 08-16-1986)
[200] (1986-09-10) NRC Review, Plans to Store Upper End Fittings (re various letters)
[201] (1986-09-25) NRC Safety Evaluation, Storage of Upper End Fittings in an Array of 55 Gallon Drums (re various letters)
[202] (1986-08-16) GPU, Storage of Upper End Fittings in Shielded 55 Gal Drums
[203] (1986-09-09) GPU Response to NRC (Discussion), Storage of Upper End Fittings (re 08-16-1986)
[204] (1986-05-15) GPU Safety Evaluation, Defueling SER, Rev. 10
[205] (1986-05-20) GPU Safety Evaluation, Defueling SER, Rev. 10 (Correction)
[206] (1986-07-24) NRC Review, Core Region Bulk Defueling Operations Limited to the Core Region (re 05-15, 07-23-1986)
[207] (1986-05-15) GPU Safety Evaluation, Defueling SER, Rev. 10
[208] (1986-05-05) NRC Safety Evaluation, Core Stratification Sample Acquisition SER, Rev. 1 (re 08-30-1985, 12-31-1985)
[209] (1986-07-23) GPU, Plan to Use Core Bore Tooling as Defueling Tool
[210] (1986-07-24) NRC Review, Core Region Bulk Defueling Operations Limited to the Core Region (re 05-15, 07-23-1986)
[211] (1986-07-24) NRC Review, Core Region Bulk Defueling Operations Limited to the Core Region (re 05-15, 07-23-1986)
[212] (1988-01-18) GPU Safety Evaluation, Lower Core Support Assembly Defueling, Rev. 2
[213] (1988-04-01) NRC Safety Evaluation, Lower Core Support Assembly Defueling (re 01-18-1988)
[214] (1988-01-18) GPU Safety Evaluation, Lower Core Support Assembly Defueling, Rev. 2
[215] (1987-11-30) GPU Criticality Assessment for Using Plasma Arc Torch to Cut Lower Core Support Assembly, Rev. 1
[216] (1987-11-30) GPU Criticality Assessment for Using Plasma Arc Torch to Cut Lower Core Support Assembly, Rev. 1
[217] GPU, "Criticality Safety Assessment for Use of the Plasma Arc Torch to Cut the Lower Core Support Assembly," GPU Letter 4410-88-L-0026/035PP, February 26, 1988 [ADAMS Accession No. ML20147D737]
[218] (1988-03-16) GPU, Criticality Safety Assessment for Use of Plasma Arc Torch (re 11-30-1987)
[219] (1988-06-06) GPU Safety Evaluation, Completion of Lower Core Support and Lower Head Defueling, Rev. 0
[220] (1988-06-27) GPU Safety Evaluation, Completion of Lower Core Support and Lower Head Defueling, Rev. 1
[221] (1988-09-09) GPU Response to NRC, Lower Head Defueling SER (re 08-11, 06-06-1988)
[222] (1988-12-01) NRC Safety Evaluation, Lower Core Support and Lower Head Defueling (re 06-06-1988)

[223] (1988-06-06) GPU Safety Evaluation, Completion of Lower Core Support and Lower Head Defueling, Rev. 0

[224] (1988-06-27) GPU Safety Evaluation, Completion of Lower Core Support and Lower Head Defueling, Rev. 1

[225] (1988-01-08) NRC Review, Use of Core Bore Machine for Dismantling Lower Core Support Assembly (re 12-03, 12-28-1987)

[226] (1988-04-01) NRC Safety Evaluation, Lower Core Support Assembly Defueling (re 01-18-1988)

[227] (1988-09-09) GPU Response to NRC, Lower Head Defueling SER (re 08-11, 06-06-1988)

[228] (1988-09-15) GPU Safety Evaluation, Completion of Upper Support Core Assembly Defueling, Rev. 0

[229] (1989-04-04) NRC Safety Evaluation, Completion of Upper Core Support Assembly Defueling (re 9-15-1988)

[230] (1988-09-15) GPU Safety Evaluation, Completion of Upper Support Core Assembly Defueling, Rev. 0

[231] (1979-05-25) NRC Policy, Direct Staff to Prepare Environmental Assessment for Decontaminate and Dispose Waste Water

[232] NUREG-0591, Environmental Assessment, Use of EPICOR-II at TMI-2 (1979-08)

[233] (1979-10-16) Memorandum and Order, Direct the Licensee to Operate EPICOR II

[234] (1979-10-22) NRC Order (10-18-1979), Operation of EPICOR II to Decontaminate Rad Waste Water in Auxiliary Building

[235] (1979-10-26) NRC Order Amend, Clarifies Order dated 10-26-1979 Concerning the Operation of EPICOR-II

[236] NUREG-0591, Environmental Assessment, Use of EPICOR-II at TMI-2 (1979-08)

[237] NUREG-0683, Vol. 1, PEIS-Decontamination and Disposal of Radioactive Wastes Resulting from TMI-2 (1981-03)

[238] (1981-03-11) GPU Technical Evaluation, SDS, Revised

[239] (1981-03-11) GPU Technical Evaluation, SDS, Revised

[240] (1981-06-18) NRC Order, Operation of SDS and EPICOR-II to Process Reactor Building Sump and RCS Water

[241] (1981-07-13) PNO-TMI-81-15, Initial Operation on SDS

[242] (1981-09-23) PNO-TMI-81-20, Reactor Building Water Processing

[243] NUREG-0796, Operation of the Submerged Demineralizer System at TMI-2 (1981-06)

[244] (1982-03-23) GPU Technical Evaluation, Addendum to SDS, RCS Processing Plan

[245] (1982-11-03) GPU Technical Evaluation, RCS Processing with RCS in a Partial Drained Condition

[246] (1984-07-19) GPU Technical Evaluation, SDS

[247] (1982-02-04) NRC Review, Update Recovery System SD and TER from 6 month to 1 year (re 01-20-1982)

[248] NUREG-0796, Operation of the Submerged Demineralizer System at TMI-2 (1981-06)

[249] NUREG-0796, Operation of the Submerged Demineralizer System at TMI-2 (1981-06)

[250] (1981-06-18) NRC Order, Operation of SDS and EPICOR-II to Process Reactor Building Sump and RCS Water

[251] (1982-10-13) GPU Safety Evaluation, SDS Liner Recombiner and Vacuum Outgassing System

[252] (1982-11-11) GPU Response to NRC (Discussions), SDS Liner Recombiner and Vacuum Outgassing System

[253] (1982-11-23) NRC Review, SDS Liner Recombiner and Vacuum Outgassing System

[254] (1982-10-13) GPU Safety Evaluation, SDS Liner Recombiner and Vacuum Outgassing System

[255] (1982-11-11) GPU Response to NRC (Discussions), SDS Liner Recombiner and Vacuum Outgassing System

[256] USNRC, “Three Mile Island Accident of 1979 Knowledge Management Digest: Recovery and Cleanup,” NUREG/KM-0001, Supplement 1, June 2016 [Available at nrc.gov]

3 CRITICALITY SAFETY EVALUATIONS

TABLE OF CONTENTS

Note: “NA” (not applicable) means the licensee and the NRC determined that the safety topic was not important for the activity

3.1 Introduction

3.1.1 Background

The area of nuclear criticality safety is broad and covers essentially the entire fuel cycle, ending with storage or disposal. This document focuses on the criticality evaluations and application of criticality safety measures for postaccident TMI-2 conditions along with the associated defueling operations. For the TMI-2 activities, the issue of criticality was of intense interest and considered to be a major safety concern by the NRC. Many interested parties provided analytical support, and the NRC sponsored independent analyses, to evaluate the potential for criticality for various scenarios. These analyses were based on various assumptions such as those concerning debris composition and configuration.

This chapter discusses the evaluations and actions performed to ensure subcriticality of TMI-2 postaccident and during defueling operations. It includes the analysis to address the effects of foreign materials inside the reactor vessel, measures to prevent the inadvertent dilution of borated coolant, and the increase of boron shim from 3500 to more than 5000 parts per million over time, as a result of changing knowledge about the condition of the damaged reactor core. Unlike the conditions at Tokyo Electric Power Company's Fukushima Daiichi Units 1, 2, and 3, the TMI-2 reactor coolant system was able to sustain a borated coolant mass. This condition allowed analysts to identify and perform bounding calculations. Also, the Fukushima units sustained greater reactor core damage and had fuel with a higher average enrichment than at TMI-2.

In this chapter, "criticality safety" refers to all activities that could affect the subcriticality of the fuel debris. Safety evaluations ensured that these activities were conducted in accordance with the plant's license, technical specifications, and applicable regulatory requirements. For TMI-2, some of these requirements were changed, as needed, to ensure that subcriticality was maintained during postaccident activities. For decommissioning of the reactor, the focus remained on protection against the consequences of an inadvertent nuclear chain reaction, preferably by preventing the reaction. To aid the decommissioning process, certain regulatory requirements were suspended. However, the NRC required that the fuel remain subcritical at all times.

3.1.2 Overview of Criticality Control at TMI-2

This section briefly overviews (●) early criticality studies that were conducted by the licensee, the NRC, and contractors; (●) special reliance on boron as a neutron-absorbing shim; (●) adoption of the safe mass fuel limit (the amount of fuel debris that could collect in any plant component without posing a criticality safety concern); and (●) subcritical criteria used to determine subcriticality in the criticality analyses to support recovery and defueling operations.

3.1.2.1 Early Criticality Studies

The licensee's criticality report [(1)] for the reactor coolant system (RCS) provided the following overview of early criticality studies. Soon after the TMI accident in March 1979, calculations of the shutdown [(a)] margin were performed on what was then known or believed to be the physical condition of the core. The consensus of the various groups of analysts [(b,, 2, 3)] was that the reactor fuel was subcritical and would remain subcritical if the boron concentration of the RCS water was maintained at 3000 to 3500 parts per million. These results were collected and reviewed by the Babcock & Wilcox Company (B&W), along with an additional series of bounding calculations. This work was described in B&W's reports, [(4, 5, 6)] "Methods and Procedures of Analysis for TMI-2 Criticality Calculations to Report Recovery Activities through Head Removal," and Addenda 1 and 2.

All of the analyses for shutdown margin reported in the above reports were based on the fuel remaining in a stable, although damaged, configuration. Reactor disassembly and defueling activities, which followed reactor vessel head removal, involved intentional disturbances of the fuel. The analyses of these reports were not directly applicable to the reconfigurations of the fuel that resulted from reactor disassembly and defueling activities. To ensure that the fuel would remain in a subcritical state during disassembly and defueling activities, an analytical criticality safety program was developed and described in the licensee's report TPO/TMI-071 "Technical Plan for Nuclear Reactivity," dated January 1984.

The technical plan initially proposed three approaches to demonstrate that the boron concentration in the RCS was high enough to ensure a subcritical reactor core for any reconfiguration of the damaged fuel. These approaches included: (•) systematic review of planned activities, definition of credible core configurations, and determination of poison requirements for maintaining these configurations subcritical; (•) use of an infinite poison that would maintain subcriticality for all core configurations (bounding poison concentration); and (•) use of design and procedural measures to preclude fuel configurations that were potentially more reactive than those previously analyzed.

To minimize schedule impact, the initial program strategy would include all three approaches until the best approach to serve the recovery effort was decided. However, at the end of 1983, the licensee realized that removal of the reactor vessel head would result in a substantially reduced capability for increasing the boron concentration throughout the RCS. Borating the RCS would require that the system be filled and pressurized to induce flow throughout the system. Once the head was removed, extraordinary measures would be required to ensure a uniform boron concentration in all parts of the RCS. Therefore, any decision that substantially

---

[a] Editor's Note: This summary document and the original documents often use terms such as "safe-shutdown condition," "shutdown margin," and "reactor shutdown." These terms are synonymous with terms denoting subcriticality such as "subcritical condition," "subcritical margin," and "subcritical."

[b] Editor's Note: Three other reports were cited in the licensee's report: GPU Nuclear Corporation, "TMI-2 Post Accident Criticality Analysis," TDR-049, August 31, 1979, Brookhaven National Laboratory, "Recriticality Calculations for TMI," May 18, 1979, Babcock & Wilcox, "TMI-2 Criticality Evaluation Notebook," NPGD-TM534, December 1979. These reports could not be found in the NRC record.

increased the boron concentration, either for the infinite poison approach or for the systematic review approach, had to be made well in advance of head lift.

The licensee's Criticality Task Force convened in February 1984 with the objective of determining the appropriate boron concentration in the RCS before head lift. The licensee decided to discontinue the systematic and procedural approaches in favor of an infinite poison approach. The procedural approach was dropped because of the potential to impose unacceptable restrictions on recovery methods for plenum removal and defueling sequences.

Similarly, the complex criticality analyses required by the systematic approach could not be performed on a schedule that would support a boron concentration increase before head lift. The systematic approach analyses might have shown that 3500 parts per million was an acceptable boron concentration for activities through defueling; however, this conclusion was not a certainty. Therefore, the task force decided that incurring the known costs of increasing the boron concentration in the vessel before head lift for use of the bounding approach would be the appropriate course of action. Subsequently, B&W and Oak Ridge National Laboratory performed several criticality analyses for defueling.

3.1.2.2 Reliance on Boron

Given the unknown severity of core damage at TMI-2 following the accident, the control rods were not relied on in any way for reactivity control and assurance of shutdown. The control room operators had no way of measuring the criticality margin; so, to ensure that the core would not become critical again, a high boron concentration was maintained in the reactor coolant. Later knowledge showed that the control rod material was essentially melted from the core region. The bounding analysis without credit for control rod material proved to be the appropriate assumption.

The physical and chemical properties of six neutron-absorbing elements were studied and combined with cost estimates to determine the feasibility of adding them to the TMI-2 reactor coolant to depress the effective neutron multiplication (k_{eff}) to less than or equal to 0.95. [(7)] Both soluble and insoluble forms of several elements were examined, such as boron (natural and fully enriched), cadmium, europium, gadolinium, lithium, and samarium. Oak Ridge National Laboratory performed criticality calculations to determine the absorber concentration required to meet the k_{eff} criterion of 0.95. The study concluded that all elements, with the exception of boron, had overriding disadvantages that precluded their use in the TMI-2 reactor. Solubility experiments in the reactor coolant showed that boron solubility was the same as boron in pure aqueous solutions of sodium hydroxide and boric acid; therefore, solubility was not a limiting factor in reaching the k_{eff} criterion with boron. An examination of the effect of pH on sodium requirements and costs for processing to remove radionuclides revealed a sharp dependence in which small decreases in pH resulted in a large decrease in both sodium requirements and processing costs. The study also concluded that, to meet any contemplated reactor safety requirements, boron could be added with existing equipment, but this addition had to be made with the reactor coolant system (RCS) filled and pressurized to ensure a uniform boron concentration.

The recovery project staff decided not to rely on analyses or models to show subcriticality when a method ensuring shutdown with boron was available. The normal boron concentration in the RCS was 1000 to 1500 parts per million (ppm). Just before the accident, a routine sample of the reactor coolant contained 1026 ppm boron. A sample taken shortly after the reactor trip indicated a boron concentration of only 700 ppm. This caused concern, especially when, 2 hours later, another sample showed a boron concentration of about 400 ppm. At the time, the operators believed that this was evidence of a boron dilution accident. In fact, it was due to reflux boiling in the core that was caused by low pressure and high temperatures. Much of the water in the coolant sample from the RCS piping was condensate that contained no boron. Because of the boiling, the actual boron concentration was higher than normal in the core.

Immediate steps were taken to raise the boron concentration in the RCS because of the low concentration samples and the higher-than-normal neutron flux readings from the source range monitors. Because of the uncertainty about the extent of damage to the core, the boron concentration was then raised to over 3000 ppm, which was established as an operating requirement. The limit was eventually raised to a minimum boron concentration of 4350 ppm to support defueling.

Boron concentration was controlled primarily by ensuring that the sources of makeup water contained the required concentrations. Boron dilution events due to possible operational error were avoided by ensuring that only approved sources of makeup water were available for injection. Reactor coolant samples were analyzed weekly to confirm this method of control. Because samples were analyzed off site at the Babcock & Wilcox laboratory in Lynchburg, Virginia, this method was not able to detect rapid changes in boron concentration resulting from possible equipment failures. A boron meter (an instrument that measures boron concentration in reactor coolant) was initially added to a temporary sample sink. Later, online boron concentration readings were established to quickly detect any boron dilution events.

3.1.2.3 Safe Fuel Mass Limit

In the early stages of the TMI-2 cleanup activities, the licensee performed analyses [c] to establish limits on the amount of fuel debris that could collect in any plant component without posing a criticality safety concern. The significant assumptions used in that analysis included a fuel enrichment of 3 weight percent uranium-235, unborated water reflection and moderation, and a maximum fuel rod diameter of 0.4 inch. The 3 weight percent enrichment approximately corresponded to the unburned condition of the highest enriched Batch 3 fuel (2.96 weight percent). The unburned enrichments for the other fuel batches at TMI-2 were 1.98 weight percent (Batch 1) and 2.64 weight percent (Batch 2). [8]

- ***Original Limit***. Based on data previously compiled [9] by the Savannah River Laboratory, the licensee's analysis (TPO/TMI-132) indicated that the minimum critical mass was 93 kilograms of

[c] The analysis was documented in the licensee's report TPO/TMI-132, "Technical Plan for Ex-RCS Criticality Safety," Rev. 1, November 1985.

uranium dioxide for the previously stated assumptions. The critically safe fuel mass limit [d] for the TMI-2 defueling operations was established at 70 kilograms, which was about 75 percent of the calculated minimum critical mass. This limit provided the criterion for the maximum amount of fuel that could collect in an isolated unit and be assured to remain subcritical regardless of other parameter values. This limit was applied to the various defueling activities at TMI-2 unless a specific evaluation demonstrated that a larger mass would be maintained subcritical.

Following the defueling of the reactor vessel, the licensee revised its criticality safety analysis to increase the TMI-2 safe fuel mass limit accordingly.

- ***Revised Limit***. The purpose of this revision was to develop a refined safe fuel mass limit for use at TMI-2 during the remaining defueling activities and to evaluate long-term storage conditions (i.e., post-defueling monitored storage). This limit was developed based on more realistic assumptions that were still conservative but less overly conservative than those used in the previous analyses. The significant data collected from debris samplings, video inspections, and other defueling data, which were unavailable at the time of the previous analyses, justified using realistic assumptions. These data provided a better understanding of the accident scenario and the actual debris configuration and composition. Therefore, the creation of a refined and more realistic model of the fuel debris was permitted.

The criterion for the new mass limit assumed that the calculated effective neutron multiplication factor$_f$ would not exceed 0.99, including a computer code uncertainty bias. This acceptance criterion was consistent with the previous licensing basis [10, 11, 12] for the reactor coolant system during defueling.

The revised analysis showed that, when more realistic assumptions were made about the composition of the fuel debris remaining at TMI-2, the critically safe fuel mass limit could be increased to 140 kilograms. This increase resulted even with the use of conservative modeling assumptions in the base case model, such as spherical geometry, unborated water in optimal mixture with the fuel, and no credit taken for impurities. Essentially, the fuel enrichment was adjusted to be made more realistic, and full-sized fuel pellets were used. No attempt was made to adjust the other three major assumptions (i.e., impurity concentration, moderation, and particle size).

The revised limit was considered applicable for isolated accumulations of fuel debris (i.e., those accumulations of fuel that would remain physically and neutronically decoupled from other fuel accumulations) at TMI-2. Fuel accumulations were considered neutronically decoupled if the equivalent of 12 inches of water separated the accumulations (based on TID-7016, [13] "Nuclear Safety Guide").

---

[d] Editor's Note: While the mass discussed here is labeled as the "critically safe fuel mass limit," the same mass is often referred to as the "minimum critical mass" throughout much of this summary document and the source documents. While use of both labels for the same mass may be common, the more correct label is the "critically safe fuel mass limit" since an accumulation of fuel up to that mass quantity would be subcritical.

Based on the available sample data, as well as the various fuel relocation pathways, the three cases presented in this evaluation were considered to bound any accumulations of fuel debris remaining at TMI-2 in excess of the original 70-kilogram limit. Also, the degree of conservatism for a particular assumption could have been modified and still demonstrate the appropriateness of the 140-kilogram limit. For example, additional cases were provided in which the highest Batch 3 fuel enrichments were used, with minimum credit for impurities and all other significant assumptions unchanged. These cases showed allowable masses in excess of 140 kilograms.

The limit of 140 kilograms was not considered applicable in cases where the fuel debris was surrounded by a thick lead reflector (e.g., the shipping casks) because, under certain conditions, lead could be a better neutron reflector than unborated water. In such cases, separate evaluations were performed.

3.1.2.4 Subcritical Criterion

Two criteria for the effective neutron multiplication (k_{eff}) were used to determine subcriticality during recovery and defueling operations: 0.99 and 0.95. For the most part, the criterion that was used in a particular criticality analysis was dependent on the results of previous analyses. However, for some evaluations, the criterion was different for the different conditions being analyzed (e.g., normal recovery operations or plant accident conditions).

The analyses for the reactor coolant system (RCS), including the reactor vessel, used the k_{eff} criterion of 0.99 (i.e., to be subcritical, the k_{eff} could not exceed 0.99). The analysis that supported selection of the additional neutron poison to use in the RCS [(14)] and the heavy load drop analysis for the polar crane load test [(15)] used the k_{eff} criterion of 0.95.

- ***Criterion (Reactor Cooldown)***. The 0.99 criterion was applied during the reactor cooldown period and was defined in the TMI-2 recovery technical specifications [(16)] as applicable to the shutdown mode of reactor operation. TMI-2 was officially placed in the shutdown cooling mode on February 13, 1980, in accordance with the TMI-2 recovery technical specifications. These recovery technical specifications defined an additional shutdown mode, called the "recovery mode," and specified maximum and minimum coolant temperatures and boron concentrations. This was the applicable mode during the long-term cooling of the core, including facility cleanup and recovery operations. [(17)] Therefore, the specific criticality safety requirements for the activities proposed through reactor vessel head removal pursuant to the recovery technical specifications included a subcritical multiplication, k_{eff}, less than 0.99. [(18)] This criterion was used for all RCS criticality evaluations, with the two noted exceptions.

- ***Criterion (Polar Crane Load Test)***. The use of the 0.95 criterion for the polar crane load test analysis was based on Criterion II of Section 5.1 of NUREG-0612, [(19)] "Control of Heavy Loads at Nuclear Power Plants." NRC generic letter [(20, 21)] on the control of heavy loads required licensees to address the guidelines in NUREG-0612. The analysis in the licensee's safety evaluation report [(22)] for the polar crane load test stated that damage to fuel and fuel storage racks based on calculations involving accidental dropping of a postulated heavy load did not result in a configuration of the fuel such that k_{eff} was larger than 0.95. Thus, the licensee followed Criterion II from NUREG-0612, consistent with the generic letter from the NRC.

- ***Criterion (Neutron Absorber Addition)***. For the evaluation of adding neutron absorber to the RCS, a shutdown margin that would render a criticality accident incredible had to be defined and the corresponding concentration of the absorber had to be quantified. When the evaluation began, the shutdown margin provided by the existing boron concentration was a subject of debate because of assumptions about the extent of core damage and the core configuration.

For the purposes of this evaluation only, the criterion was established that a criticality event would be incredible when k_{eff} was less than 0.95 with the fuel in a highly reactive configuration, not suitably poisoned by added soluble or insoluble neutron absorbers. A conservative model of the reactor was developed to serve as the foundation for calculations in determining the absorber concentrations required to meet the reactivity criterion. Both the model and the criterion were formulated based on the advice of reactor physics consultants. Oak Ridge National Laboratory performed the calculations to determine the maximum reactivity of the model at various poison concentrations. Given the highly conservative assumptions used in the model, a higher k_{eff} and a lower boron concentration were established than were quantified in the evaluation as conditions, which ensured that subcriticality under all defueling conditions would be justified. [23]

- ***Criterion (Handling and Storage Defueling Canisters)***. Criticality analyses for handling and storage of the defueling canisters used the 0.95 criterion. The conditions that were analyzed included: (●) a single defueling canister in normal and damaged conditions; (●) arrays of canisters in normal and damaged conditions; and (●) a single canister drop. [24] Analyses [25] of the canisters for plant accidents (e.g., draining of the spent fuel pool) used the 0.99 criterion. The use of the 0.95 criterion for the defueling canisters was consistent with the requirements that INEL had specified for storage of defueling canisters in the fuel storage pool at INEL's site. Since the canister storage rack size at TMI-2 was the same as the rack size to be used in INEL's pool (center-to-center spacing of 18 inches), the INEL subcritical criterion applied to the defueling canisters at TMI-2. [e] The plant accidents for the defueling canisters (e.g., draining of the spent fuel pool), however, involved different conditions than were part of the analysis requirements for storage at INEL. Thus, the analysis for these conditions was not tied to the same subcritical criterion. [26]

- ***Criterion (Handling and Storage Debris Containers)***. Criticality evaluation for canisters (referred to as debris containers) was modified for end-fitting removal and storage to use the 0.99 criterion. This evaluation did not include a separate, explicit analysis. Instead, the safety

---

[e] Editor's Note: The 0.95 criterion for subcriticality is the same criterion as in the guidance used by the NRC for analyses of transportation packages (e.g., NUREG-1617, "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel," issued March 2000) and for analyses of interim, independent storage of spent nuclear fuel (e.g., NUREG-1567, "Standard Review Plan for Spent Fuel Dry Storage Facilities," issued March 2000). It is also consistent with the guidance in the NRC's NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Section 9.1.1, "Criticality Safety of Fresh and Spent Fuel Storage and Handling," and Section 9.1.2, "New and Spent Fuel Storage." These guidance documents apply this criterion to all conditions (e.g., normal, off-normal, accident). These NUREGs can be accessed at nrc.gov.

evaluation used the results from the RCS criticality analysis [27] and the analysis for foreign material in the RCS during defueling activities. [28] Although there were differences between the assumptions used in the previous criticality analyses and those that would actually have been used for an explicit analysis of submerged containers, direct application of the criticality results to this evaluation was conservative. The evaluation of the containers demonstrated that the k_{eff} for these canisters would not exceed the 0.99 criterion during loading, handling in the canister transfer shield, and storage in the fuel transfer canal and the “A” spent fuel pool. [29]

3.1.3 Chapter Contents

This chapter presents criticality safety evaluations of the postaccident TMI-2 and defueling operations. It describes the many studies performed to give the reader an understanding of the thinking of the analysts at the time, the expectations and the reality, the uncertainties in the data, and the measurement and mitigation methods. It also presents a high-level timeline of cleanup activities.

The evaluations presented in this chapter ensured that all activities that could create an inadvertent criticality condition were addressed and consequences evaluated. Controls were maintained in accordance with the requirements of the plant’s license, technical specifications, procedures, and applicable regulatory requirements. Additionally, adequate contingencies were developed for normal operations and accident conditions.

Most cleanup activities were evaluated for criticality.

Section 2 summarizes the key studies that were used to support safety evaluations. The remaining sections present the safety evaluation for each applicable cleanup activity or system. Section 8 lists the endnotes for references cited throughout this chapter.

3.2 Key Studies

This section provides a high-level chronological progression of the criticality analyses that followed the TMI-2 event. Each subsection below summarizes the referenced document. The intent is to help the reader understand the thinking that occurred with the progression of time and growth of knowledge throughout the defueling procedures.

In particular, from the outset of the event, significant consideration was given to: (●) the potential for criticality; (●) the lack of information; (●) prevention of a criticality; and (●) issues, such as nonuniform boron concentrations, that resulted from a dilution event. Boron shim was an early choice, but other poisons were also considered.

3.2.1 Recriticality Potential of TMI-2 Core
(NRC, May 1979)

This early NRC evaluation report [30] provided the results of a number of KENO [f] Monte Carlo analyses to establish the potential of a criticality in the TMI-2 core. All models in this analysis assumed the complete, simultaneous loss of all movable control rods and all fixed burnable poison rods. Results indicated that 3500 parts per million natural boron uniformly distributed and maintained in the moderator/coolant would guarantee subcriticality for all credible possible abnormal states of the core. The study indicated that the highest enriched peripheral region of the core was controlling for criticality for an accidental boron dilution event. Regardless of the boron concentration throughout the core, a slug of completely unborated water passing through a minimum of four contiguous fuel assemblies (in a square) and through the full length of the core would cause a criticality. Four fuel assemblies correspond to about 2 percent of the core volume.

3.2.2 Criticality Analyses of Disrupted Core Models of TMI-2
(Oak Ridge National Laboratory, ORNL-CSD-TM-106, December 1979)

This report [31] formally documented a series of analyses performed to support the President's Commission on the Accident at Three Mile Island. This study determined the reactivity effects of various hypothetical scenarios in which the reactor core could have been disrupted and provided documented models and analytical methods used for those scenario analyses. The report included results of various parametric studies such as: (●) effects of fuel pin geometry changes determined through infinite lattice pin-cell calculations; (●) benchmark analysis of the preaccident, as-measured critical configuration at hot, zero-power reactor startup condition; and (●) analyses of the disrupted core models that included variations to determine the reactivity worth of soluble boron, control rods, and burnable poison rods.

3.2.3 A Further Evaluation of Risk of Recriticality at TMI-2
(NRC, April 1980)

This report [32] reviewed previous studies related to the probability and consequences of criticality from the damaged reactor. More detailed assessments were performed to confirm the adequacy of those studies and to provide additional insight into ways to minimize the risk of criticality. The most probable mechanism for criticality was boron dilution, but it was determined to be a slow enough process that, with appropriate instrumentation and procedures, the approach to criticality would be detected and corrected. To the extent that boron concentration in excess of 3500 parts per million could be ensured, the probability of criticality was further minimized. The most likely direct radiological consequence of criticality was from increased

---

[f] Editor's Note: The version of KENO available at the time of this evaluation is outdated and was replaced by later versions, notably KENO-V and KENO-VI in the currently available versions of SCALE that are supported by the code developer, Oak Ridge National Laboratory. More modern cross-section libraries, both multigroup and continuous energy, are also available.

dose rates inside containment. A study of more realistic and more probable criticality events also concluded that there would be no offsite consequences.

3.2.4 Programmatic Environmental Impact Statement
(NRC, March 1981)

The PEIS [33] for the cleanup of TMI-2 reported the options and associated environmental impacts of the activities necessary to the cleanup, as well as the potential radiological health and safety impacts.

- ***Summary***. At the time the report was issued, subcriticality of the reactor was ensured by the maintenance of sufficient boron concentration levels in the reactor coolant system (RCS). The one operable source range neutron detector was used to monitor subcriticality. A small amount of control rod material was believed to have melted during the accident. The available shutdown margin was estimated to be about 15 percent $\Delta k/k$ (i.e., there were about 15 percent too few neutrons to sustain nuclear chain reaction at a constant rate). Several groups independently examined the potential for criticality under various hypothetical circumstances. Based on these analyses, the PEIS concluded that the reactor would be maintained in a subcritical state with 3500 parts per million (ppm) of boron in the reactor coolant, even with the total absence of other control materials. The most probable (although very unlikely) cause of criticality was found to be boron dilution, which would be a slow enough process that any approach to criticality could be detected and remedied. To ensure that the reactor remained subcritical throughout the decontamination program, it would be necessary to maintain control of the boron concentration in the RCS until the defueling was completed.

- ***Evaluations***. The PEIS evaluations involving criticality concerns are sumerized below:

  - *Revised Boron Concentration*. Precise information on the extent of control rod damage was not available at the time the PEIS was published. Therefore, to ensure subcriticality, the amount of boron [g] in the reactor coolant was increased to about 3850 ppm, and a new lower limit of 3500 ppm was established. Numerous criticality analyses [h, 34, 35] agreed that the reactor core would remain subcritical with a boron concentration of 3500 ppm in any physically possible geometry even if all the fixed and movable absorber rods were removed.

  - *Temperature and pH*. The effects of temperature and pH on boron concentration in the RCS were ruled out as a major concern. The solubility of boron as boric acid was 4400 ppm at 35 degrees Fahrenheit (degrees F) and 7100 ppm at 59 degrees F. Based on criticality calculations from the Brookhaven National Laboratory report, “Recriticality Calculations for

---

[g] Editor’s Note: The original document(s) summarized here often referred to “boric acid” concentrations and used the term as a synonym for “boron.” Thus, the text in this summary has changed references to boric acid to refer to boron, or boron concentrations, where appropriate.

[h] Editor’s Note: The PEIS referenced other studies by Argonne National Laboratory, Babcock & Wilcox, Brookhaven National Laboratory, and the licensee. However, these documents could not be found.

TMI," dated May 18, 1979, large amounts of strong acid would have been added before significant decreases in soluble boron concentration were observed.

- *Underborated Water*. The only concern regarding criticality indicated by the studies in the TMI-2 situation was the introduction of water to the reactor core with a boron level of much less than 3500 ppm. Calculations from the licensee's technical evaluation report TDR-049, "TMI-2 Post Accident Criticality Analysis," dated August 31, 1979, which was supported by experimental data from the Westinghouse Reactor Evaluation Center, showed that the introduction of 1000 ppm borated water in a square array of the 2.96 weight percent enriched fuel assemblies from the core's outer region could result in criticality. Lack of information on the existing state of the core made it difficult to accurately calculate the critical boron concentration. However, calculations did show that the introduction of underborated water could result in the core becoming critical, assuming an undamaged core geometry.

- *Dilution Detection*. Several methods of detecting a reduction of boron in the RCS were available to alert the operators, so that the dilute water source could be terminated before core criticality. These methods included: (●) ex-core nuclear instrumentation; (●) periodic boron analysis of reactor coolant water; and (●) pressure and temperature readings of the RCS. Control of the reactor coolant chemistry and all sources of other water added to or mixed with the reactor coolant during the cleanup would prevent a dilution accident.

- *RCS Interfaces*. Systems connected to the RCS were borated to requirements from the recovery technical specifications. These systems included the standby pressure control system (installed for recovery), makeup and purification system (existing system), and decay heat removal system (existing system). A closed valve connection from the demineralized water system to the makeup pump suction did exist, but the valve was located in a high-radiation area where access would be administratively controlled. In addition, the power supply breakers to the makeup pumps would be danger-tagged open to prevent inadvertent pump operation.

- *Containment Barrier*. As a final barrier of defense, the containment building was specifically designed to contain fission product inventories. Most of the fission products produced in a criticality accident would have been extremely short lived. A review by the Argonne National Laboratory determined that the total amount of curies released to the atmosphere during the accident was greater than the total fission product inventory that would have been in the current core 10 hours after a hypothetical severe criticality transient.

3.2.5 Evaluation of Potential/Consequences of Recriticality During Cleanup and Defueling at TMI-2

(NRC, ANL-NRC-RAS-81-1, June 1981)

In support of the PEIS, this report [36] included: (●) an evaluation of the potential for achieving criticality in the disrupted TMI-2 core during the cleanup operations; (●) an assessment of the additional damage and fission product release from the damaged fuel during a criticality event;

and (●) an investigation of potential leak paths to the environment for any criticality-induced fission product releases.

3.2.6 Criticality Calculations To Support Recovery through Reactor Vessel Head Removal

(Babcock & Wilcox, BAW-1738, June 1982)

This report [37] documented the first criticality safety results specifically for reactor disassembly and defueling activities. Previous analyses were based on the fuel remaining in a stable, although damaged, configuration. The specific objectives of the report were to: (●) evaluate the reactivity of postulated TMI-2 core configurations; (●) evaluate the reactivity of potential fuel accumulations outside the core region; (●) evaluate the potential reactivity effects of various perturbations resulting from the proposed activities; and (●) verify that a boron concentration of 3500 parts per million would maintain an adequate margin of subcriticality under all postulated credible conditions.

Examples of some of the proposed activities that could rearrange fuel configurations or otherwise affect the subcriticality of the fuel system included: (●) insertion of the axial power shaping rods; (●) attempts to uncouple the control rod drive mechanism; (●) insertion of inspection and sampling equipment into the reactor vessel through penetrations in the head; and (●) removal of the reactor vessel head. The purpose of the analytical assessment was to demonstrate that the reactor would be subcritical at all times during all of these proposed activities.

Three approaches were proposed to demonstrate that the boron concentration in the reactor coolant system would be high enough to ensure the core would remain safely subcritical for any reconfiguration of the fuel. These approaches included: (●) systematic review of planned activities; (●) identifying credible core configurations and determination of poison requirements for maintaining these configurations subcritical; (●) use of an infinite poison that would maintain subcriticality for all core configurations (bounding poison concentration); and (●) use of design and procedural measures to preclude fuel configurations that could be potentially more reactive than those previously analyzed. To minimize schedule impact, the initial program strategy was to pursue all three approaches until the best approach for the recovery effort was decided.

This report provided the analyses of various geometrical configurations of moderator, reflector, and fuel. These configurations represented both credible and hypothetical fuel arrangements in the reactor coolant system as a result of activities relating to reactor vessel head inspections and reactor vessel head removal.

The analyses used conservative core configurations that were assumed to represent worst case conditions for recovery activities from the head removal, except for major core rearrangement associated with the head drop on the reactor vessel. These static configurations included a maximum credible core damage model (50 percent core damage in a debris bed over 50 percent intact fuel assemblies) and a model for 100 percent core damage. In addition, an analysis was performed with fuel in the vessel but outside of the core region. Models used in the analysis included a sphere of 50 percent of the damaged highest enrichment fuel (19 assemblies) in the bottom of the reactor vessel, a hemisphere of 50 percent of the core in the lower vessel, and a cylinder of fuel particles falling down from the core region.

3.2.7 Verification of Criticality Calculations for TMI-2 Recovery Operations through Head Removal

(Babcock & Wilcox, BAW-1738, Addendum 1, October 1982)

This report [38] verified that the models used in the original BAW-1738 report [39] were conservative based on the data obtained from axial power shaping rod insertion, control rod drive mechanism uncoupling inspections, and the through-head video inspections of the damaged fuel. Data from these recovery operations through the third video viewing inside the reactor vessel were used to create a revised model of fuel damage. This model was then compared to the fuel damage models used for the criticality calculations of the reactor shutdown margins and detailed in the revised BAW-1738. The results of the comparison verified that the criticality calculations were conservative since the calculations assumed more fuel damage than was evident from the data. Consequently, the revised BAW-1738 was determined to be valid for recovery operations through reactor vessel head removal.

3.2.8 Criticality Analysis for Heavy Load Drop Accident in Support of Recovery through Reactor Vessel Head Removal

(Babcock & Wilcox, December 1983)

This report [40] discussed the worst case model of additional fuel disruptions that were considered possible as a result of a heavy load drop accident, such as dropping the reactor head onto the vessel or plenum. The heavy load drop model was conservative for criticality analyses because it assumed the maximum credible amount of additional cladding failures with the fuel collapsed to the most reactive configuration. The analyses indicated that, with this conservative model, the core would remain subcritical with an effective neutron multiplication less than 0.99 at a boron concentration of 3500 parts per million.

3.2.9 TMI-2 Fuel Canister Interface Requirements for INEL

(INEL, EGG-TMI-6156-R1, June 1984)

This report [41] focused on the fuel canister interface requirements at INEL. Fuel canisters loaded with enriched uranium dioxide were required to remain subcritical under all conceivable loading, handling, and storage situations. If neutron absorbers were used inside fuel canisters, their continued effectiveness under transport and storage conditions was required to be demonstrated. An effective neutron multiplication less than 0.95 was required at INEL for loading parameters, which demonstrated subcriticality of a storage array of fuel canisters on 18-inch centers in water. Two independent criticality safety analyses were required by the INEL Safety Manual 9020, “Fissile Material Control Areas,” dated September 30, 1983 (copy provided in Appendix A to EGG-TMI-6156). The analyses considered the most reactive conditions that conceivably could have occurred. These conditions included: (●) misplacement of the fuel canisters in storage racks; (●) dropping of loaded fuel canisters into an already loaded rack; (●) reflection of neutrons in and by water; and (●) canister flooding with nonborated water.

Specific design requirements and considerations included the following: (●) verification and documentation of essential criticality safety design features such as canister size, canister material, and any fixed neutron poison; (●) consideration in the canister design for the "future use" of handling, transporting, and storage of canisters, including neutron interaction between canisters located in an array and the ability to periodically inspect any fixed neutron poison; (●) validation of computer codes and cross-section sets used for calculations against applicable critical experiments; (●) off-normal conditions; (●) reflectors more efficient than water (lead cask wall, concrete storage basin wall and floor, etc.); and (●) the highest enrichment instead of an average enrichment in the criticality calculations. Examples of off-normal conditions included: (●) batching of fissile material; (●) loss of solid neutron absorber; (●) redistribution of neutron absorber and fissile material; (●) change in canister dimensions or breach of containment; (●) lack of structural integrity (i.e., corrosion or gas pressurization); (●) loss of array neutron isolating material such as a storage basin drainage accident; and (●) credible moderation that could be mixed with core debris or an influx of storage basin water.

3.2.10 Addition of Soluble and Insoluble Neutron Absorbers to the Reactor Coolant System of TMI

(GPU Nuclear, GEND-026, July 1984)

This report [42] examined the feasibility of adding more neutron absorber, such as boron or suitable alternatives, to the reactor coolant system (RCS). This addition would increase the shutdown margin to the extent that a criticality accident would be incredible, regardless of the configuration of the fuel. Six elements (boron, cadmium, gadolinium, lithium, samarium, and europium) were studied for possible additions to the RCS to maintain the effective neutron multiplication below 0.95. Boron (as boric acid) was found to have a variety of advantages, such as lower cost, minimum impact on water cleanup systems, and no serious materials compatibility problems. In addition, boron could be added using existing chemical addition equipment. Dissolved boron concentration levels of 5500 parts per million were found to be adequate to maintain the core subcritical under all feasible configurations. Boron additions had to be made before lowering water in the vessel because once the water level was lowered, gas pockets at the tops of the steam generators would prevent the mixing of boron throughout the RCS.

3.2.11 Reactor Coolant System Criticality Report

(GPU Nuclear, November 1984)

This report [43] provided the criteria and rationale used for determining the proper boron concentration for the reactor coolant system (RCS). The chosen boron concentration ensured that the fuel in the RCS would remain subcritical throughout all reactor disassembly and defueling operations. This included the movement of any reactor component, including fuel within the vessel, whether planned or due to an accident, such as a heavy load drop. This report covered issues of criticality only within the RCS pressure boundary, predominantly the primary coolant loop. Included in this evaluation as part of the RCS were the: (●) reactor vessel; (●) steam generators; (●) pressurizer; (●) hot- and cold-leg piping; (●) reactor coolant pumps; (●) surge line; and (●) decay heat dropline.

- ***Results (Boron Concentration)***. This report provided the basis and criteria used for the selection of a boron concentration in the RCS that supported a shutdown margin of at least 1 percent or an effective neutron multiplication (k_{eff}) less than or equal to 0.99. The report defined a minimal acceptable boron concentration of 4350 parts per million (ppm) to ensure that the k_{eff} of the RCS would not exceed 0.99 for all credible configurations. For the RCS design-basis model, the calculated value of k_{eff} included a 2.5-percent delta-k computer code uncertainty bias. To provide an adequate operating margin, an administrative limit on the minimum operational RCS boron concentration would be established at 4950 ppm.

- ***Evaluation Approach***. The criticality study was based on four key considerations.

  - *Limited Knowledge*. First, there was limited knowledge of the spatial distribution of fuel within the reactor vessel. At the time that the report was issued, only visuals gathered from Quick Looks, a few “grab” samples from the core cavity region, and the topographic model of the upper core region (rubble bed) were available.

  - *Bounding Analysis*. Second, evaluations completed for this report were performed with the intent to bound all credible situations that could have been encountered during the entire defueling process. No attempt was made to define assumptions for a specific defueling activity or phase. The licensee chose this approach to reduce the potential of placing unacceptable restrictions on recovery methods. Although a more systematic approach for specific activities might have shown that 3500 ppm was an acceptable boron concentration, such an approach would be complex, take time, and produce uncertain conclusions.

  - *Most Reactive Fuel Configuration*. Third, the worst fuel configuration identified as part of the criticality safety evaluation for recovery activities from head removal was a pile of all Batch 3 fuel in the bottom of the reactor vessel. This was the most reactive location identified in the RCS. Therefore, calculations performed for fuel in the lower head would bound conditions in all other RCS locations.

  - *Limited Credit for Fuel Burnup*. Fourth, the study took credit for fuel burnup in the Batch 3 fuel only. Radial determination of the average assembly burnup showed that the least burned fuel was along the periphery of the core. Batch 3 fuel was originally located along the core periphery. The rationale for modeling burnup to only Batch 3 fuel was because the analysis model showed a small reactivity effect if fuel Batch 1 and 2 were added to Batch 3. So, any credit for burnup of Batch 1 and 2 fuel would have a negligible effect on k_{eff}. The reactivity effect of Batch 1 and 2 fuel was small, since the analyses placed the entire initial inventory of Batch 3 fuel, with the highest enrichment, in the center of the fuel arrangement.

- ***Assumptions***. Oak Ridge National Laboratory performed calculations that used three different concentrations to determine the k_{eff} for the hypothetical model at different poison levels. These calculations were performed with a number of conservatisms built into the model. The major points of conservatism included: (●) location of fuel mass (bottom reactor vessel head region); (●) use of entire core mass; (●) shape of the model (lenticular) that excluded the physical restrictions imposed by the lower core support structure; (●) placing highest enrichment fuel in the center of the model (Batch 3); (●) optimized rubble model shape and size; (●) no credit for cladding or structural materials; (●) no credit for solid poison (control rods); (●) optimized fuel to moderator ratio; (●) optimized fuel and moderator temperature; and (●) no burnup in Batch 1 and 2 fuel.

- ***Sensitivity Analyses***. Several sensitivity studies were performed to determine the reactive worth of several different parameters. The parameters studied included: (●) reflector; (●) fuel inventory; (●) geometry shape; (●) boron concentration; (●) fuel burnup; and (●) fuel temperature.

- ***NRC Review***. [44] The licensee's criticality report was revised by the NRC, and the agency's evaluation was attached to the transmittal letter. As discussed in the attachment, the NRC concluded that an RCS boron concentration of 4350 ppm would ensure at least 1-percent shutdown margin for the hypothetical conservative fuel model assumed in the licensee's analysis. The NRC noted that the maintenance of an operating RCS boron concentration of about 5000 ppm would provide a significant larger real shutdown margin, and a corresponding degree of enhanced safety, as the licensee conducted reactor disassembly and defueling operations. Refer to the NRC's safety evaluation report for further details of the agency's review.

3.2.12 TMI-2 Transfer System Criticality Technical Report

(Babcock & Wilcox, Document No. 77-1155739-02, June 19, 1985)

The defueling canisters were transferred to locations within the reactor and fuel handling buildings using a transfer shield containing lead. Transfer of canisters to the shipping cask used a different device called a "transfer cask." This report examined the effective neutron multiplication factor (k_{eff}) for both the transfer shield and cask. This report was included as Attachment 1 to the licensee's technical evaluation report [45, 46] on the defueling canister.

- ***Objectives***. Calculations in this report address the following objectives: (●) evaluate the optimal fuel composition with the transfer shield in place; (●) determine the effect of the gap region between the inserted canister and the cask or shield for centered and off-centered canisters; (●) determine the most reactive canister type in the transfer shield; (●) evaluate the most reactive insertion point for a canister in the transfer shield; and (●) evaluate the most reactive canister for the worst insertion point in the transfer cask.

- ***Reactivity Criterion***. The reactivity criterion for criticality safety used in this analysis was that the value of k_{eff} for the most reactive canister inside the transfer system could not exceed 0.95. [i]

- ***Conclusion***. The report results indicated that for ruptured and nonruptured canisters, no poison materials, other than those contained in the canisters, were required in the design of either the transfer shield or cask to maintain k_{eff} less than 0.95. Canisters with extensive internal or external damage from being dropped or deformed were not addressed, since these canisters would be handled on a case-by-case basis and therefore were not included in the current work scope.

- ***Reevaluation***. The preamble to Attachment 1 in Revision 3 of the technical evaluation report provided an update to the criticality evaluation. The results of the original analysis (Revision 1) assumed that the most reactive fuel particle capable of being in the knockout canister was an optimally moderated standard, whole fuel pellet. However, this assumption was no longer appropriate due to fuel particle sizes greater than whole pellets to be loaded into a knockout canister. To assess the impact of this assumption, an evaluation was performed to determine the infinity neutron multiplication factor ($k_{infinity}$) for the most reactive Batch 3 fuel particle, when optimally moderated with unborated water. The $k_{infinity}$ for the optimum size was found to be only 0.07 percent delta-k higher than the $k_{infinity}$ for the standard whole pellet. The results presented in this attachment were still considered appropriate, since this increase was small and the other assumptions included in the analysis were conservative. Additionally, even with this increase of 0.07 percent delta-k, which tends to increase the k_{eff}, the k_{eff} criterion for the canisters within the canister transfer shield was met.

3.2.13 Hazard Analysis for the Potential for Boron Dilution of the Reactor Coolant System

(GPU Nuclear, Rev. 2, September 1985)

This report [47] assessed the potential for boron dilution in the reactor coolant system (RCS) during a variety of pre-defueling recovery activities. Methods of isolating the RCS were identified to provide a high degree of assurance that a dilution event would not occur. The report provided probabilities of a dilution event for different operations. The probability of a dilution event was a function of the ability to isolate the RCS and not of the acceptable boron concentration.

The boron dilution analysis approach included the following tasks: (●) identify the potential points of water injection to the RCS, such as core flood tanks, pressurizer, reactor coolant pump seals, steam generator secondary side, reactor vessel nozzles, and top of the open reactor vessel; (●) track each potential RCS injection point to potential dilution sources, such as tanks,

---

i The safety evaluation report indicated that this criterion analysis was consistent with the legacy section 10 CFR 72.73, “Requirement for Advance Notice and Protection of Export Shipments of Special Nuclear Material of Low Strategic Significance,” and American National Standards Institute/American Nuclear Society (ANSI/ANS) Standard 8.1 (1983), “Nuclear Criticality Safety in Operations with Fissionable Material Outside Reactors”; ANSI/ANS Standard 8.17 (1984), “Criticality Safety Criteria for the Handling, Storage, and Transportation of LWR Fuel Outside Reactors”; and ANS 8.7/N16.5 (1982), “Guide for Criticality Safety in Storage of Fissionable Materials.” (Editor’s Note: Section 10 CFR 72.73 was removed from Part 72.)

coolers, demineralizers, evaporators, heaters, closed cooling water systems, and spent fuel pool; (●) identify isolation barriers for each dilution source, such as removed spool pieces, closed valves, and heat exchanger or pumps with head differences; (●) determine probability of failure of isolation barrier configuration due to hardware faults and human error; and (●) estimate total plant boron dilution potential by considering the number of injection paths, the reliability of each isolation barrier, and the possibility for operator error, or failure, in identifying and terminating a boron dilution event.

Credit for mitigation was heavily dependent on the detection capability. The means of detecting a boron dilution event at TMI-2 included: (●) monitoring reactor coolant level; (●) monitoring levels of dilution sources (e.g., tanks); (●) performing mass balance calculations of the RCS; (●) monitoring the status of the positions of valves, pumps, and breakers by using equipment checklists; (●) using operable source range neutron detectors; and (●) routine sampling of RCS boron concentration.

3.2.14 Limits of Foreign Materials Allowed in the TMI-2 Reactor Coolant System During Defueling Activities

(GPU Nuclear, Rev. 1, September 1985)

This revised report [(48)] assessed the effects of introducing foreign materials into the reactor coolant system (RCS) reactivity. The 4350 parts per million boron concentration established in the RCS criticality report [(49)] did not totally protect against the potential increase in the effective neutron multiplication (k_{eff}) caused by the introduction of foreign materials into the RCS. The foreign materials report concluded that the establishment of a 2-gallon limit on the amount of unborated moderating material (i.e., a material that could become interstitially dispersed within the fuel) in the RCS would ensure a k_{eff} no greater than 0.99 across all credible situations. This result was based on an RCS boron concentration of 4950 parts per million, which was the lower operational limit permitted by the administrative procedure at the time.

The NRC review [(j, 50)] concluded that the work described in the licensee's safety evaluation provided an excellent exploration and analysis of the problems of reactivity perturbations of foreign material insertions for TMI-2 defueling operations, including the areas of approach and criteria, calculation methodology, bounding geometry, and material selection. The resulting analyses, leading to a selection of limits for material addition, were reasonably conservative and determined to be acceptable.

3.2.15 Technical Plan: Ex-Reactor Coolant System Criticality Safety

(GPU Nuclear, GPU/TMI-132, Rev. 1, November 1985)

This internal licensee report [(51)] provided analyses to establish limits on the amount of fuel debris that could collect in any plant component without posing a criticality safety concern. The significant assumptions in the analysis included: (●) a fuel enrichment of 3 weight percent uranium-235; (●) unborated water reflection and moderation; and (●) a maximum fuel rod

[j] Editor's Note: Formal transmittals of the licensee's request for review and the NRC's review could not be located. However, the cited documents were found in internal NRC correspondence.

diameter of 0.4 inch. The 3 weight percent enrichment corresponded to the unburned condition of the highest enriched fuel, Batch 3 (2.96 weight percent). The unburned enrichments for the other fuel batches at TMI-2 were 1.98 weight percent (Batch 1) and 2.64 weight percent (Batch 2). The analysis concluded that the minimum critical mass was 93 kilograms of uranium dioxide. A safety margin was then applied, thus establishing the critically safe fuel mass for the TMI-2 defueling operations at 70 kilograms, which was about 75 percent of the calculated minimum critical mass. This limit provided the criterion for the maximum amount of fuel that could collect in an isolated unit and remain subcritical regardless of what other parameters changed. This limit was applicable to the various defueling activities, unless a specific evaluation demonstrated that a larger mass would be maintained subcritical.

3.2.16 TMI-2 Criticality Studies, Lower Vessel Rubble, and Analytical Benchmarking
(ORNL, ORNL-CSD-TM-222, December 1985)

This report [(52)] documented a bounding strategy that was adopted to ensure subcriticality during all TMI-2 defueling operations. This strategy was based on establishing a safe soluble boron level for the entire reactor core in an optimum reactivity configuration. This report described a two-step analysis process. First, an infinite lattice model was used to determine the fuel rubble lattice configuration that maximized reactivity. Second, the entire core was modeled using the fuel rubble lattice configuration and was placed into the lower vessel in maximum credible (albeit highly improbable) geometric configurations. [(k)] Included in the analyses were the effects of fuel burnup, which were determined from a simplified power history of the reactor. The report also discussed the analytical methods employed and the determination of an analytical bias with benchmark critical experiments.

3.2.17 Review of the State of Criticality of the TMI-2 Core and Reactor Vessel
(DOE, DOE-NCT-01, April 1987)

This report [(53)] reviewed the available information for the following topics: (●) physical and chemical state of the fuel in the TMI-2 reactor; (●) calculations of the reactivity of the core using the current configuration; (●) margin of safety estimates; and (●) level of dissolved boron to maintain subcriticality during defueling.

3.2.18 Criticality Safety Assessment for Increasing the TMI-2 Safety Fuel Mass Limit
(GPU Nuclear, Rev. 0, February 1989)

This report [(54)] presented a refined safe fuel mass limit for evaluating the remaining defueling activities at the time and post-defueling long-term storage conditions (i.e., post-defueling monitored storage). The development of this limit was based on more realistic assumptions, using significant data collected from debris samplings, video inspections, and other defueling

[k] Editor's Note: Various aspects of the evaluation technique described in the Oak Ridge National Laboratory report (e.g., creating an equivalent model and use of cell-averaged constants) are no longer necessary if more recent versions of the SCALE computer code are used. These more recent versions of SCALE have enhanced capabilities to model the rubble configurations described in the paper (e.g., a dodecahedral fuel lattice in KENO-VI geometry). Also note that more updates and changes have been made to the cross-section libraries in SCALE since the TMI-2 accident.

data that became available after the original analysis in 1985. The data provided a better understanding of the accident scenario and the actual debris configuration and composition. Results of this assessment increased the safe fuel mass limit to 140 kilograms for isolated fuel accumulations that would remain physically and neutronically decoupled from other fuel accumulation, by the equivalent of 12 inches of water separating the accumulations. This conclusion was not considered applicable in cases where the fuel debris was surrounded by a thick lead reflector (e.g., the lead shield of the shipping cask), since under certain conditions, lead could be a better neutron reflector than unborated water. The report stated that separate evaluations would be required for such cases. [l]

3.2.19 TMI Fuel Characteristics for Disposal Criticality Analysis
(DOE, DOE-SNF-REP-084, September 2003)

This report [55] provided details of the parameters needed to perform a criticality analysis of the various TMI canister types during intermediate storage at INEL. The document also stipulated the conditions anticipated for degradation failures within the repository that needed to be addressed in the criticality analysis before acceptance of this spent nuclear fuel in the repository. The knockout canister with a full assembly's worth of fuel pellets was selected as the bounding case for criticality analysis. The knockout canister design offered fewer constraints for fuel pellet distribution and for obtaining optimum moderation because of the greater free volume for the moderator. The report neglected the canister internals, which increased the amount of free volume available for moderator, fuel, and fuel movement. The report stated that analysis models should investigate optimized fuel distribution and moderator ratios for the purpose of determining maximum reactivity.

The report stated that no intact fuel assembly was loaded in a fuel canister. The highest reported physical mass of debris inside any canister (842.18 kilograms) represented 123 percent of the specified weight of an intact assembly. The highest reported fissile loading in any TMI-2 canister (10.06 kilograms) was only 73.3 percent of a beginning-of-life fissile load for an assembly with maximum enrichment (2.96 weight percent). The maximum beginning-of-life uranium-235 in any TMI-2 assembly of 13.72 kilograms provided the basis for any single-package criticality analysis conducted with TMI-2 canisters.

3.3 Data Collection Activities

3.3.1 Axial Power Shaping Rod Insertion Test

- ***Purpose***. To individually move the eight axial power shaping rod (APSR) assemblies in the core to provide insight into the extent of core and upper plenum damage. This early insight allowed time to factor this information into plans for subsequent inspections for the head, upper plenum, and core removal.

---

[l] Editor's Note: The licensee requested the NRC's review of its report; however, the NRC's safety evaluation was not located.

- ***Evaluation: Criticality***. [56] The purpose of this evaluation was to ensure that an adequate reactor shutdown margin existed for the APSR movement test with a reactor coolant boron concentration greater than 3500 parts per million (ppm). The licensee's safety analysis considered the effects of this test on the shutdown of the damaged reactor and the effects on the fuel that could have been transported out of the core into other regions of the reactor coolant system (RCS). The licensee's safety evaluation concluded that the existing boron concentrations of greater than 3500 ppm in the RCS, combined with existing operating procedures and systems, ensured that the reactor would remain shut down during the APSR testing.

- ***Evaluation: Criticality (Reactor Core Shutdown)***. [57] The evaluation considered the effects of positive reactivity insertions that could result from APSR motion and changes from fuel displacements.

  - *Previous Evaluations*. Following the accident, several organizations performed calculations to assess the shutdown margin of the reactor. These independent studies all supported the contention that the reactor would remain shut down at ambient temperature and boron concentration of 3500 ppm. These studies assumed various core damage models and neglected the APSR assemblies. In most cases, the other control rod assemblies and fixed burnable poisons were also neglected, and extensive fuel rearrangement was assumed. Parametric studies were performed to determine worst case conditions that were then used for the calculations.

    The APSR testing would not invalidate the results of the above studies, given that the shutdown reactivity provided by the APSR rods was not included in these studies. Further, the maximum reactivity addition that would result from the APSR motion was small compared to the shutdown calculated in these studies.

    Additional criticality studies performed by Babcock & Wilcox (B&W) and Oak Ridge National Laboratory (ORNL) investigated in further detail the shutdown of the reactor. The results of these studies had not been published at the time of this safety evaluation for APSR tests; however, these results supported the conclusions of the previously noted references indicating that the reactor was shut down at boron concentrations of 3500 ppm. The B&W studies investigated the reactivity effects, which included: (●) fuel enrichment and loading; (●) fuel and fuel fine (small particles) distribution; (●) fission product decay; (●) reduced temperatures; (●) core structural materials; (●) control rod worth (50-percent fuel damage model); (●) changed volume fractions, and (●) fuel burnup. The ORNL study investigated the reactivity effects of fuel fine distribution within fuel rod lattices.

  - *APSR Reactivity Worth*. The preaccident TMI-2 physics test manual provided rod worth curves for the percent of total worth of the APSRs as a function of vertical position. The curves showed that fully inserting these assemblies from their current position would reduce their worth by 50 percent in an undamaged core. Fuel redistribution in the middle region of the core and a less damaged lower core region would selectively reduce the worth of any partially withdrawn assemblies. Therefore, this decreased worth would reduce the absolute reactivity change from their insertion or further withdrawal.

The physics test manual provided a maximum worth of negative 0.24-percent change in reactivity ($\Delta\rho$) for the APSRs at zero effective full power days, 300 degrees Fahrenheit (degrees F), and 1506 ppm boron. The manual indicated that the worth would increase slightly because of depletion by a factor of 1.1. Using a total maximum worth of 0.264-percent $\Delta\rho$ (0.24 percent x 1.1) and a 50-percent change due to insertion of the APSRs, $\Delta\rho$ was calculated to be 0.132 percent for the change in reactivity. Higher boron concentrations would reduce the worth of the APSRs. Figures in the manual showed that lower temperatures would also reduce the APSR worth.

These calculations were for an undamaged core. If the upper half of the core was damaged, then the worth of the APSRs could be less in the withdrawn position. Therefore, the reactivity increase on insertion would be less than calculated. In any case, even if an uncertainty as large as 300 percent was assumed for APSR worth due to core damage, the total reactivity insertion remained less than 0.5-percent $\Delta\rho$.

- *In-core Fuel Displacement*. In addition to positive reactivity insertions, APSR motion could result in further fuel rearrangement. However, as the above-referenced calculations did not address specific core configurations, but rather worst case studies, these studies would cover changes to in-core fuel distribution resulting from APSR motion, which confirmed reactor shutdown at boron concentrations of 3500 ppm.

- ***Evaluation: Criticality (Out-of-Core Shutdown)***. [58] The evaluation of the possible reactivity consequences of APSR motion considered the shutdown of out-of-core fuel debris. Out-of-core fuel transport mechanisms during APSR motion would be limited to short periods of natural circulation flow, reactor coolant makeup flow, and gravity. Of these, only gravity is capable of moving any significant quantities of fuel. Therefore, the only out-of-core region that could credibly experience a change of fuel concentration as a result of APSR motion, was the reactor vessel lower plenum.

Given that fuel material was probably swept out of the core during the first day of the accident as a result of the initial coolant pump switching and reflood transients, an APSR motion that would have any effect on out-of-core reactivity shutdown margins was unlikely. If any fuel was dislodged during the testing, this fuel most likely would remain in the core region. The bottom undamaged portion of the core, the core support structure, and the damaged fuel would act as a screen, minimizing the possibility of fuel dropping into the pressure vessel's lower plenum.

- *Other Evaluations of Out-of-Core Shutdown*. The referenced B&W report "TMI-2 Criticality Evaluation Notebook," dated December 1979, evaluated the possibility of critical fuel configurations in the lower plenum volume. These calculations were supplemented by additional B&W studies that were not yet published at the time of the APSR testing safety evaluation. The new calculations included higher enrichments, more reactive fuel configurations, and other particle geometries. The results of these new studies showed that for the 50-percent damaged core model, subcriticality was achieved at 3500 ppm boron for the maximum 2.98 weight percent enrichment fuel loading. In this case, the B&W maximum damage model (top half of core damaged) would release the fuel from 30 assemblies with 2.98 weight percent. The calculation used a worst case volume fraction and hemisphere geometry as described in the referenced B&W report "Criticality Evaluation for Pre-Head Lift Technical Evaluation Report," dated March 4, 1982, which found that criticality could not be achieved with average enrichment fuel at 3500 ppm boron. Therefore, for criticality to occur in this region, the maximum enrichment fuel would have to somehow separate from the other enrichments.

- *Lower Plenum Shutdown Evaluation*. The method used in the evaluation to show an adequate out-of-core shutdown margin for APSR testing differed from past studies of TMI-2. This APSR method considered: (●) fuel distribution; (●) boron concentration; (●) pretest shutdown margin; (●) boron reactivity worth; and (●) other reactivity considerations.

  - *Fuel Distribution*. The APSR evaluation method did not rely on calculations of a specific assumed geometry. The APSR method used operational data to determine a shutdown margin and compared this margin to reactivity changes that could result from fuel redistribution. This method also evaluated the increased shutdown margin that resulted from the addition of 500 ppm boron. The shutdown margin calculated using the APSR method was compared to the amount of fuel that would be added to result in criticality. In turn, the amount of fuel was dependent on the amount of fuel initially in the lower plenum. Various initial concentrations of fuel were assumed, and the fuel increments required for criticality were calculated.

    The APSR evaluation noted the results from this evaluation method and the past studies concluded the existence of a shutdown margin. The evaluation showed that fuel transfers of the same order of magnitude (a minimum of 66 percent) or larger than those that were assumed to have already occurred were required to cause criticality. APSR testing results demonstrated that such transfers were not credible.

  - *Boron Concentration*. On April 27, 1979, forced circulation of the reactor coolant was terminated. At that time, the boron concentration was about 2900 ppm (the chemistry logs showed boron concentrations of 2869 ppm on April 25, 1979, and 2960 ppm on May 2, 1979). Because this concentration was established before forced circulation was terminated, this value could be considered representative of the entire RCS. The boron concentration in the RCS was greater than 3500 ppm on April 12, 1982 (the chemistry log showed a value of 3753 ppm). The chemistry log showed that the boron concentration was greater than 3500 ppm until October 1979. The plant makeup rate replaced many system volumes of coolant in the RCS during the last 2 1/2 years with natural circulation. Therefore, the current in-core boron concentration corresponded to the current chemistry sample results. Allowing for measurement accuracy (100 ppm), the boron concentration in the lower plenum was 500 ppm greater than when the forced circulation stopped.

– *Pretest Shutdown Margin*. The APSR evaluation assumed that the effective neutron multiplication (k_{eff}) of any fuel in the lower plenum on April 27, 1979, was less than 1 with a reactor coolant temperature of less than 180 degrees F. The conclusion that the reactor was shut down was supported by the following:

  - Previous calculations concluded that critical fuel configurations in out-of-core regions were unlikely. In addition to the analytical supposition that criticality in out-of-core regions was not credible a mechanistic evaluation also showed that fuel transport sufficient to support criticality in out-of-core regions was improbable. Both the upper and lower core end fittings of the fuel assemblies provided a barrier that would preclude fuel transport of large fuel particles out of the core. In addition, surveillance activities to date [m] had not located out-of-core regions containing the tons of fuel fines required for criticality.

  - Two previous criticality evaluation reports, the NRC's criticality evaluation report [59] on the risk of criticality at TMI-2 and the referenced B&W report "Criticality Evaluation for Pre-Head Lift Technical Evaluation Report," dated March 4, 1982, both concluded that sustained criticality in the core region was not credible These reports predicted that in the event of local criticality, fuel dispersal would cause a shutdown. In addition, they concluded that reactor coolant temperatures and reactor coolant activities would increase because of the energy required to establish core shutdown conditions. These increases would be detectable but were not observed. These studies, although performed for in-core regions, would also apply to sustained criticality in the lower plenum.

  - If the fuel in the lower plenum was not subcritical at 3000 ppm boron, but instead was in a sustained critical configuration, then criticality was not evident and did not present a safety problem. Reactor coolant activity did not increase during the period of time when the plant was at or less than 3000 ppm boron and under natural circulation flow conditions. No increase of neutron counts was observed on the plant nuclear instruments. If sustained criticality was not evident and did not represent a problem at 3000 ppm boron, then criticality would not present a problem at 3500 ppm boron. The increase of 500 ppm boron would not result in significant changes in the fuel reactivity coefficients; therefore, essentially the same shutdown mechanisms would be available. As a result, the consequence of criticality would be expected to be the same and not represent a safety problem. The licensee expected that coolant chemistry measurements would be the most sensitive means to detect sustained criticality in the lower plenum and an increase in boron concentration could be used to terminate such an event.

– *Boron Reactivity Worth*. An increased reactivity worth of 500 ppm in the reactor coolant boron concentration was dependent on the core configuration. The worth would be

[m] Editor's Note: Video examinations of the lower reactor vessel head region during the removal of the upper plenum showed extensive collection of fuel debris. Refer to later criticality evaluations for details on defueling the lower head.

lowest for a highly damaged fuel configuration. Therefore, a highly damaged core geometry was used to assess its worth. The boron worth was also sensitive to the volume fraction used in the calculations, as well as the assumed fuel enrichment. The boron worth in the reactor vessel lower plenum was based on the results of the NRC's criticality evaluation report [60] on the criticality potential of TMI-2. The NRC report provided a curve for the core of infinite neutron multiplication ($k_{infinity}$) versus the water-fuel ratio as a function of boron concertation for two fuel enrichments of 2.96 and 2.31 weight percent. The water-fuel ratio that gave the highest $k_{infinity}$ at the highest boron concentration (i.e., 3000 ppm) was used for this evaluation (i.e., 0.8 volume fraction). In addition, an enrichment of 2.96 percent was used since this enrichment resulted in the most reactive configuration. A value of 2.75-percent change in reactivity for the worth of 500 ppm boron was obtained from another curve in the NRC report for $k_{infinity}$ versus boron concentration.

– *Other Reactivity Changes*. On April 21, 1979, after the flow was terminated and an increase in the boron concentration was observed, the evaluation of the net reactivity change of fuel in the lower plenum considered the following:

  - *Xenon Decay*: The TMI-2 core physics manual showed that xenon decay was 80 hours after reactor shutdown. Therefore, no reactivity change would result from xenon decay.

  - *Samarium Decay*: The TMI-2 core physics manual showed the buildup of samarium was 20 days after reactor shutdown. Therefore, no reactivity change would result from samarium buildup.

  - *Other Fission Product Decay*: Two reports, ORNL's report [61] on the criticality analyses of disrupted core models of TMI-2 and the referenced licensee's technical data report TDR-049, "TMI-2 Postaccident Criticality Analysis," dated August 31, 1979, reported the effect of fission product decay on the reactivity of spent fuel. Both studies concluded that the consequence of fission product decay would increase shutdown margins. However, the APSR evaluation did not include this increase in shutdown.

  - *Temperature Change*: The temperature defect for fuel in the lower plenum, from 180 to 70 degrees, was calculated to be minus 0.64-percent $\Delta\rho$. This value was derived from data provided in the referenced B&W report "TMI-2 Criticality Evaluation Notebook," dated December 1979. The B&W report stated that, for a volume fraction of 0.63 and 2.6 weight percent uranium-235, the temperature coefficient varied from minus 0.8×10^{-4} $\Delta\rho$ per degree F at 2100 ppm to minus 0.5×10^{-4} $\Delta\rho$ per degree F at 4000 ppm. Interpolation yielded a value of minus 0.58×10^{-4} $\Delta\rho$ per degree F at 3500 ppm. Data from the referenced licensee's technical data report TDR-049, "TMI-2 Postaccident Criticality Analysis," dated August 31, 1979, showed that the magnitude of the negative temperature coefficient increased with higher volume fractions. Therefore, B&W's use of a large volume fraction was considered to be conservative. Temperature defect was primarily dependent on the moderator and boron condition. Fuel effects would be less important so this coefficient could be used for 2.96 weight percent fuel. An additional conservatism in this calculation was the assumption that 180 degrees F was the initial temperature. At the time, boron concentrations were less than 3000 ppm, there was natural circulation flow, and inlet temperatures were as low as, or lower than, 155 degrees F.

- *Net Shutdown Margin*. The APSR evaluation showed that 500 ppm boron would provide an additional shutdown margin in out-of-core regions of 2.75-percent $\Delta\rho$. Further, temperature change was the only significant source of reactivity addition since flow was terminated. This change was conservatively identified as a decrease to 70 degrees F with a corresponding reactivity increase of 0.64-percent $\Delta\rho$. Therefore, as a result of changes to plant conditions since flow was terminated, the net increase in shutdown was 2.11-percent $\Delta\rho$ (2.75 percent minus 0.64 percent) for any fuel that was located in the reactor vessel lower plenum.

- *Fuel Transfer Required To Offset Shutdown Margin*. The APSR evaluation reported that a minimum shutdown margin of 2.11 $\Delta\rho$ was adequate for any fuel located in the lower plenum. This evaluation determined the amount of additional fuel required to offset this margin and was based on calculations from the referenced B&W report "Criticality Evaluation for Pre-Head Lift Technical Evaluation Report," dated March 4, 1982. The results from the B&W study were similar to those reported in an earlier referenced proprietary B&W report, "TMI-2 Criticality Evaluation Notebook," dated December 1979.

  The results from the B&W calculations were determined from differences in reactivity; therefore, the results were not as sensitive to errors in absolute reactivity. Many of the variables that affected absolute reactivity, such as pellet geometry and temperature, were not significant in these calculations.

  The B&W calculations were performed for fuel volume fraction of 0.55 (maximum reactivity) and 3500 ppm boron. The APSR evaluation noted that the total calculated k_{eff} for the maximum enrichment fuel was greater than 1. Shutdown in these cases would result from geometry or poison effects not included in the B&W calculations. These additional shutdown mechanisms would cancel out in the differential shutdown calculations so criticality could not be achieved using average enriched fuel.

  For criticality to occur as a result of APSR motion, large quantities of fuel must transfer from the core to the lower plenum. These quantities were on the same order, or larger than, the quantities that were transferred during the accident. Such fuel transfer quantities as a result of APSR motion were not credible. As stated earlier, the hydraulic forces, temperatures, and other data available during the accident to support fuel transport would not be available during APSR testing.

- ***NRC Review: Criticality***. [62] The NRC provided its safety evaluation in an amendment to the recovery technical specifications. The licensee requested changes to the technical specifications to allow the movement of an individual control rod or an APSR, for the purpose of gathering additional information on the condition of the core and to prepare the reactor vessel for head removal. The NRC agreed with the licensee that, by moving an individual control rod or APSR, more information on the core condition could be obtained. Also, before head removal, it would be necessary to decouple the APSR lead screws. This could most easily be done when the rods were first fully inserted. Therefore, inserting the APSRs at this time on an individual basis was a step that would be performed sometime during the cleanup. At the time, the technical specifications did not allow any movement and required that all control rod drive breakers would remain open. The NRC's analysis of the shutdown margin assumed no control material or poison in the core; the shutdown margin was then calculated with a 3000-ppm boron concentration.

  - *Evaluation*. A previous NRC analysis showed that a boron concentration of at least 3000 ppm would provide an adequate shutdown margin with a maximum k_{eff} of 0.944. Even a more conservative analysis [63] showed that, with a boron concentration of 3500 ppm, the core would remain subcritical with a k_{eff} less than 0.90 in any physically reasonable rearrangement of the fuel. Both analyses had assumed the absence of all control rods and burnable poisons when calculating the k_{eff} values. Even though technical specifications required a minimum boron concentration of 3000 ppm, the licensee showed by sampling that the RCS was consistently maintained at about 3800 ppm. Therefore, the NRC concluded that moving a control rod or APSR would not significantly affect the shutdown margin. As verified by RCS sampling, the licensee had consistently been maintaining a boron concentration of about 3800 ppm and thereby had ensured even more shutdown margin than the recovery technical specifications required.

  - *Conclusion*. The NRC concluded that any single rod movement would have a minimal effect on the required boron concentration and shutdown margin.

  - *Operational Limitation*. All control rod mechanism energization and subsequent movement would be controlled NRC-approved procedures and accompanying safety reviews for specific tasks being performed. The procedures would limit the energization of one mechanism at a time and require upgraded surveillance to ensure that no unexpected change in core conditions occurred.

3.3.2 Camera Insertion through Reactor Vessel Leadscrew Opening

- ***Purpose***. To insert a camera into the reactor vessel through a control rod drive mechanism leadscrew opening to provide the first visual assessment of conditions inside the reactor vessel. This activity was known as "Quick Look."

- ***Evaluation: Criticality***. [64] The licensee's safety evaluation stated that this criticality analysis supplemented the criticality analysis provided previously for the axial power shaping rod insertion test. Reactivity changes as a result of postulated fuel disturbances or changes, were described in a previous Babcock & Wilcox (B&W) criticality evaluation report [65] on methods and procedures of analysis for TMI-2 criticality calculation to support recovery activities through reactor vessel head removal. This report showed that the core would remain subcritical during the activities associated with Quick Look. The evaluation in the B&W report considered credible fuel configuration inside and outside of the core region and the effects of postulated fuel disturbances or changes in physical conditions. The report concluded that this assurance of subcriticality would be provided at all times when the reactor coolant was borated to a concentration of 3500 parts per million (ppm) or greater.

The water level in the secondary side of the steam generators would be maintained at an elevation below that of the reactor coolant system (RCS) water level. This would ensure that, if any fuel had accumulated in the steam generator tubes, the accumulation of fuel would remain subcritical at all times during the Quick Look activities.

- ***Evaluation: Boron Dilution***. [66] The licensee's safety evaluation considered the potential for a boron dilution event and identified potential dilution paths. Appropriate administrative procedures and precautions were developed to preclude a boron dilution event during the Quick Look when the pressure and water level were lowered. Appendix C to the safety evaluation report described the actions taken to prevent boron dilution and discussed actions that would be taken in the unlikely event that dilution had occurred.

  - *Prevent Boron Dilution*. The actions to prevent boron dilution included: (●) removing spool pieces to ensure system flowpath isolation; (●) draining systems to reduce pressure differentials with the RCS; and (●) tagging valve lineups and tagging out unnecessary pumps that provided a greater margin of protection against leakage into the RCS. A boron dilution event caused by a secondary to primary system leak through steam generator tubes would be prevented by keeping the secondary water level below the RCS water level in the steam generator and maintaining equal nitrogen pressures on the primary and secondary sides of the steam generators. Other systems that could present a dilution pathway were identified and their operations were strictly controlled while the control rod drive mechanisms were open during Quick Look (refer to Appendix C to the safety evaluation report).

  - *Monitor Boron Content*. Even though a dilution event was unlikely, procedures were established to ensure its early detection. In such a case, action could be taken to find the source of the dilution and as needed, to stop the leak or to inject additional boron in the RCS. The means to identify a dilution event included the following:

    - *Boron Concentration*. The boron concentration would be monitored by the RCS coolant level.

    - *Leak Rate*. After the reduction in RCS level for Quick Look, a base system leakage rate would be established. Based on this leakage rate, the operations staff would receive a plot of predicted level versus time. Superimposed on this predicted level would be alarm and action level limits. The alarm levels were 24 inches above and below the base level. The action level was 12 inches above the base level. If unborated water was added to the RCS, the RCS boron concentration would still be above 3500 ppm at the higher alarm level. The lower alarm level was used as a precaution to indicate a possible increase in plant leakage and to take some action to change the level.

– *Nuclear Instrumentation*. In addition to monitoring the RCS coolant levels and the alarm and action levels, the source range neutron instrumentation would also be monitored. A base count rate would be established after the coolant level was lowered for the Quick Look. An increase in count rate of 2 times the base rate for more than 1 minute would be considered an alarm limit. An increase of 5 times the count rate for less than 1 minute would also be considered an alarm limit. On reaching these limits, the response would be the same as for the RCS coolant high-level alarm.

- *Detect and Terminate Boron Dilution*. The following procedures were required if the high alarm level was exceeded: (●) The control room operation log would be reviewed to determine the source of dilution. (●) The position of all isolation valves and status of all pumps would be checked. (●) Storage tank levels would be checked to determine the source of coolant dilution. (●) If the high alarm level was reached, Quick Look operations would be temporarily terminated, and the mechanism seal would be replaced.

  If the above actions did not stop the increase of RCS coolant level and the high alarm level was reached, the TMI-2 Emergency Procedure 2202-1.2, [n] "Unanticipated Boron Dilution," would be used to increase the RCS boron concentration.

- ***NRC Review: Criticality***. [67] The NRC's safety evaluation found that the results of the criticality analysis for Quick Look were consistent with the findings of previous studies of the potential for TMI-2 criticality. These studies all concluded that there was a substantial existing shutdown reactivity margin in the RCS. This margin was more than adequate to mitigate the consequences of any credible core fuel configuration, including any configuration that could result from disconnecting the control rod drive mechanism leadscrew.

- ***NRC Review: Boron Dilution***. [68] The NRC's safety evaluation considered measures for prevention and monitoring of boron dilution during the Quick Look and for corrective actions in the event of a dilution incident. For example, one dilution event involved leakage from the secondary system through the steam generator tubes to the primary system. The corrective action required the secondary-side water level to be lower than the primary-side level. This measure would ensure that any leakage would be from the primary system to the secondary system. Other measures included actions taken to isolate systems that interfaced with the RCS, such as the demineralized water system and the makeup and purification system. Such actions

[n] Editor's Note: The revision was not cited in the source document.

included tagging isolation valves shut, tagging pumps off, and removing piping spool pieces, where possible. The positions of all valves that provided isolation from the RCS would be confirmed every 24 hours. Storage tanks that provided a potential source of dilution water would be monitored every 24 hours to ensure a constant water level in the tanks.

The water level in the RCS would be monitored with an installed level monitoring system during the Quick Look activities. This system would provide early detection of any unplanned additions to (or losses from) the system and permit corrective action, such as the termination of Quick Look activities and the addition of borated water.

3.3.3 Reactor Vessel Underhead Characterization (Radiation Characterization)

- ***Purpose***. To characterize radiation under the reactor vessel head to ensure that adequate radiological protection measures would be taken to keep radiation exposures ALARA during head removal. To achieve this objective, measurements were necessary with the reactor vessel head still in place.

- ***Evaluation: Criticality***. (69) The licensee's safety evaluation considered the potential of core disturbances during the underhead radiation characterization program activities. This program involved inserting an instrument through the manipulator support tube and control rod drive mechanism housing, through the plenum, and into the rubble bed. This activity raised the potential for disturbing the core. The licensee's safety evaluation report (70) for Quick Look had previously evaluated this disturbance. The consequences of the instrument contacting the rubble bed, either intentionally during the insertion or inadvertently for any reason, were considered to be no more severe than probing the rubble bed, performed as part of Quick Look.

The probing of the rubble bed during Quick Look was within the bounds of the criticality analysis described in the Babcock & Wilcox (B&W) criticality evaluation report (71) for recovery activities through head removal, which was submitted to the NRC as part of the safety evaluation for Quick Look. Since the potential core disturbances associated with these data acquisition tasks were considered to be no more severe than those from the Quick Look core probe, the evaluation concluded that the consequences of potential core disturbances were bounded by the analysis described in the B&W report.

During the underhead characterization activities, the reactor coolant system (RCS) boron concentration would be maintained greater than or equal to 3500 parts per million (ppm), which, based on the B&W report, would ensure subcriticality.

- ***Evaluation: Boron Dilution***. (72) The licensee's safety evaluation considered (●) results of previous criticality analysis related to lowering reactor coolant level in the reactor vessel; (●) actions that would be taken to prevent boron dilution events; (●) actions that would be taken to monitor boron content in the RCS; and (●) actions that would be taken to detect and terminate inadvertent boron dilution.

- *Introduction*. During the underhead characterization program, reactor shutdown (subcriticality) would be ensured by the presence of boron in the reactor coolant. The licensee's safety evaluation report [(73)] for the Quick Look activity stated that maintaining RCS boron concentrations of 3500 ppm or greater ensured subcriticality under all credible conditions. A review of the information obtained during axial power shaping rod insertion and the Quick Look activity supported this finding.

  Operating experience since the accident demonstrated that controlling boron concentration in the RCS was possible. The RCS conditions differed from those that existed during the Quick Looks of previous years. The primary coolant level was lowered, and the primary coolant pressure reduced. During the underhead characterization program, the reactor coolant water level would be lowered below the Quick Look level (elevation range of 331 to 335 feet), by about 1 foot below the reactor vessel flange (i.e., elevation 321.5 feet). In view of these differences, the evaluation considered the ability to reliably maintain a controlled boron concentration in the RCS. The purpose of this evaluation was to review the precautions that would be taken to ensure the maintenance of the required RCS boron concentration.

  The RCS temperature and chemistry would not be significantly affected during the underhead characterization; therefore, boron solubility would remain essentially unchanged. In fact, the slight increase in RCS temperature due to lower water level would improve boron solubility conditions. The only way RCS boron concentration could be changed in an uncontrolled manner during underhead characterization was by dilution of the RCS coolant with water that was either unborated or borated below 3500 ppm.

  The evaluation showed that the existing procedures while the RCS was depressurized would prevent the uncontrolled addition of coolant to the RCS, thus preventing the uncontrolled reduction of the boron concentration. In addition, if for some unforeseen reason boron dilution should occur, the monitoring and corrective action procedures would preclude significant reductions in boron concentration and ensure that the reactor remained subcritical.

- *Prevent Boron Dilution*. Boron dilution would result when water containing boron concentrations less than 3500 ppm was added to the RCS. The potential sources of this water were the various systems connected to the RCS, which included the secondary system. Systems that potentially contained coolant with boron concentrations less than 3500 ppm were reviewed and isolated to ensure that they would not be credible sources of boron dilution. Two isolation boundaries were provided for each potential in-leakage path. An isolation boundary was defined as: (●) a closed tagged-out valve; (●) an electrically locked-out pump; (●) a removed spool piece; (●) a heat exchanger tube boundary; or (●) a reduced pressure differential.

  The following actions would be taken to prevent the unintentional dilution of the boron in the RCS. The evaluation concluded that these actions would prevent the dilution of the RCS boron concentration during the time when the pressure and water level were lowered.

– *Steam Generator*. One potential source of RCS boron dilution was a secondary coolant leakage through the once-through steam generator (OTSG) tubes. In the past, the potential for this leakage was precluded by maintaining the RCS pressure higher than the secondary cooling pressure, so any leakage would flow from the secondary system to the primary system.

- *RCS Pressure*. During the underhead characterization, the reactor vessel water level would be subjected to building pressure, while the hot legs and pressurizer would be under a nitrogen blanket of about 1 pound per square inch gauge (psig) pressure. To preclude RCS dilution, procedures required that water levels and cover pressures in the secondary side of the steam generators be maintained lower than those in the primary side.

- *OTSG Water Level*. The secondary side of the OTSG would be drained to below the 313-foot level, and the upper voided portion would be filled with nitrogen to a pressure of 1 psig. The reactor vessel water level would be lowered to a minimum level of 321 feet 3 inches. (This accounted for a 3-inch tolerance in the RCS level indicator.) This resulted in a minimum level in the hot leg or primary side of the steam generators of 318 feet 7 inches, and the nitrogen pressure above the primary side of the OTSGs would be maintained at 1 psig. Therefore, at a minimum, a small pressure difference would exist across the OTSG tubes that would cause flow to be from the primary to the secondary system should a leak occur. The pressure differential would increase at all reactor vessel water levels above 321 feet 3 inches.

- *Level Measurements*. To monitor possible water leakage into the OTSG, each generator was equipped with a level measuring device. The “A” OTSG incorporated a pressure gauge at the 281-foot elevation in the auxiliary building and a standpipe in the service building. The “B” OTSG was fitted with a water-filled flexible polymer tube located in the containment building to visually measure water level.

- *Isolation*. In addition, possible in-leakage paths would be isolated. Surveillance of the levels and valve positions was performed periodically as required by technical specifications. Under these conditions, the secondary volume of steam generators would not be a credible source of RCS boron dilution.

– *Makeup and Purification/Standby Pressure Control Systems*. These systems were borated to greater than 3500 ppm and would be operated by approved procedures to let down reactor coolant through the submerged demineralizer system (SDS) and makeup back to the RCS. The makeup pumps would be tagged off, and portions of connections to these systems that were not used for makeup would be isolated. Should the RCS level decrease below the controlled range, letdown from the RCS would be secured until the level increased back to the controlled range. Should the level continue to decrease, makeup would be initiated from the standby pressure control system or a bleed tank borated to greater than 3500 ppm using approved procedures.

Analysis of a sample taken from the appropriate reactor coolant bleed holdup tank of each makeup water batch would ensure that the makeup water was borated greater than 3500 ppm.

- *Demineralized Water System (DWS)*. The DWS was reviewed and, where possible, spool pieces in the flowpath to the RCS were removed. If spool pieces could not be removed, then isolation valves in the flowpaths would be tagged shut.

- *Submerged Demineralizer System*. The SDS could be operated to process the water letdown from the RCS. This would not create a dilution problem because the SDS would be isolated from the RCS except via the appropriate bleed holdup tank, which would be monitored for boron content.

- *Other Systems*. Other systems that could present dilution pathways to the RCS included: (●) decay heat removal; (●) mini-decay heat system; (●) core flood system; (●) intermediate closed cooling water; (●) decay heat closed cooling water; (●) chemical addition; (●) steam generator feedwater; and (●) spent fuel cooling.

  The following actions would be taken to prevent the dilution of RCS boron by unintentionally transferring reactor coolant from other systems that contained boron concentrations less than 3500 ppm to the RCS: (●) Systems were reviewed and isolation valves in the flowpaths were tagged shut. The isolation criteria provided two isolation boundaries for each potential in-leakage path. (●) A checklist was prepared listing all valves that would be used for isolation while the RCS was at reduced pressure (including those valves associated with the standby pressure control system, DWS, and SDS). The position of these valves would be confirmed every 24 hours during this period. (●) All pumps in these systems, except those required to be operable according to the recovery operations plan (technical specifications surveillance schedule) or technical specifications, would be tagged out to further preclude the inadvertent transfer of coolant to the RCS. (●) Levels of all storage tanks that could be sources of water into the RCS would be monitored and logged once every 24 hours.

o *Actions Taken To Monitor Boron Content*. The evaluation noted three means to monitor boron dilution events.

  - *Reactor Coolant Level Indication*. At the time, the RCS water level indication was available from four different instruments connected to the decay heat line, located outside of the containment building. This redundancy of level indication would ordinarily ensure sufficient information to properly ascertain the correct level. Since RCS level indication was a prime source of information to verify that a boron dilution circumstance had not occurred, another level indication system was made available. A flexible polymer tube connected to an RCS cold-leg pipe was added and was normally used to determine RCS level in the drained-down condition when the reactor vessel head was to be removed.

- *Boron Measurements*. An RCS sample would be obtained once a week as required by the technical specification and analyzed for boron content in accordance with approved procedures. The sampling frequency of once per week was deemed adequate, considering the RCS leak rate, instrument error, and sampling losses.

- *Neutron Monitoring*. The source range neutron instrumentation would be monitored from the control room.

- *Detect and Terminate Boron Dilution*. The actions that would be taken to prevent boron dilution in the RCS would make dilution unlikely; nevertheless, procedures were established to ensure the early detection of a dilution event. In such a case, action would be taken to find the source of the dilution and stop the leakage or inject additional boron as needed.

- *Conclusion*. The evaluation concluded that actions discussed above were sufficient to preclude inadvertent boron dilution. In the unlikely event such dilution occurred, procedures would provide for actions that would permit dilution detection and provide the information needed to terminate the coolant transfer. Based on the use of these plant limits and procedures, reactor shutdown was assured, and criticality was not considered credible.

- ***NRC Review***. [74] The NRC's safety evaluation stated that safety evaluations for inadvertent criticality, decay heat removal, and combustible gas accumulations were performed in previous evaluations. The results of these evaluations, combined with actual measurements and observations, provided high confidence that these issues would be adequately dealt with during the proposed underhead characterization study. The concern over inadvertent criticality was based on the potential for boron dilution events. Procedural controls, periodic chemical analysis, piping system isolations, and primary system level monitoring would be systematically performed to mitigate boron dilution.

3.3.4 Reactor Vessel Underhead Characterization (Core Sampling)

- ***Purpose***. To obtain up to six specimens of the core debris using specially designed tools. These tools would be lowered into the reactor to extract samples of the loose debris and to load the samples into small, shielded casks for offsite shipment and analysis. The analysis of the samples would: (●) identify the composition of the particulate core debris; (●) determine particle size; (●) determine fission product content; (●) determine fission product leachability from the debris; (●) analyze the drying properties of the debris; and (●) determine whether pyrophoric materials existed in the core debris. This examination was a follow-on activity to the underhead characterization study.

- ***Background***. Three shipping casks [o] were considered [75] for transporting the core debris samples to the laboratory. The selected cask was the modified and recertified Model CNS 1-13C Type B shipping cask, with the sample in a Specification 2R container. An earlier version of this cask was used to ship spent submerged demineralizer system liners. [76]

- ***Evaluation: Criticality***. [77] The licensee's safety evaluation stated that the core debris sample program involved inserting a sampling tool through the manipulator support tube and control rod drive mechanism housing, through the plenum, and into the rubble bed. The licensee considered two criticality scenarios. One instance involved the criticality of the reactor core due to disturbance and the second, the criticality of the sample in the transfer cask.

    - *Core Criticality*. The core debris sample program involved inserting a sampling tool through the manipulator support tube and control rod drive mechanism housing, through the plenum, and into the rubble bed. This raised the potential for disturbing the core. This disturbance was previously evaluated in the licensee's safety evaluation report [78] for Quick Look, and the consequences were considered no more severe than those of probing the rubble bed as part of Quick Look.

      The probing of the core rubble bed during Quick Look was within the bounds of the criticality analysis described in Babcock & Wilcox's (B&W's) criticality evaluation report [79] for recovery activities through head removal, which was submitted to the NRC as part of the safety evaluation for Quick Look. Since the potential core disturbances that were associated with debris sampling were considered to be no more severe than the Quick Look probe, which penetrated the debris bed about 14 inches, the evaluation concluded that the consequences of potential core disturbances were bounded by the B&W analysis and by the safety evaluation of the axial power shaping rod insertion tests. During core sample retrievals, the boron concentration in the reactor coolant system would be maintained greater than or equal to 3500 parts per million, which, based on the B&W report, would ensure subcriticality.

    - *Sample Criticality*. Six samples were obtained from the debris bed. Each sample would be retrieved in a container whose volume was about 2.4 cubic inches (39.3 cubic centimeters). The sample size would be limited to about 1 cubic inch, as demonstrated by tests. If the container were filled with fuel and unborated water with the optimum fuel-to-water ratio, the size of the container was too small for its contents to constitute a critical mass. This evaluation was supported by information contained in the Savannah River Laboratory's report [80] "Critical and Safe Masses and Dimensions of Lattices of U and UO_2 Rods in Water," and the Lynchburg Commercial Fuel Plant SNM-1168. [p] The first document indicated that 28,000 cubic centimeters of 3-percent enriched uranium dioxide in the

---

[o] Editor's Note: While large shipping containers of radioactive materials may often be referred to as "shipping casks," the proper term for such containers, when loaded with contents and in their transportation configuration, is "package." See 10 CFR Part 71, "Packaging and Transportation of Radioactive Material," Section 71.4, "Definitions."

[p] Editor's Note: The safety evaluation report did not specify the complete citation, and only the facility's license number was referenced.

optimum configuration in water was required to achieve criticality. The second document indicated that 14,000 cubic centimeters of 4-percent enriched uranium dioxide in the optimum configuration in water was required to achieve criticality. Consequently, one sample could not represent a critical mass, nor could all six samples when combined.

- ***Evaluation: Boron Dilution***. (81) The licensee's safety evaluation stated that it considered boron dilution but did not specifically address this concern. The evaluation concluded that the planned activities would not increase the probability of occurrence, the consequences of an accident, or malfunction of equipment that was important to safety previously evaluated. This conclusion was based on the work being performed in accordance with approved procedures, measures being taken to prevent a reactor coolant system boron dilution event, and potential disturbances of the core staying within previously evaluated bounds.

- ***NRC Review***. (82) Editor's Note: The NRC's safety evaluation report did not specifically address these evaluation topics.

3.3.5 Core Stratification Sample Acquisition

- ***Purpose***. To perform the activities associated with: (●) installation, operation, and removal of the core boring machine; (●) acquisition of the core samples; (●) transfer of the samples from the machine to the defueling canisters; and (●) viewing of the lower vessel region through the bored holes.

- ***Evaluation: Criticality***. (83) The licensee's safety evaluation stated that the only credible means of attaining criticality of the fuel within the vessel was through deboration of the reactor coolant system (RCS) water or introduction of foreign materials to the reactor vessel. The potential for boron dilution during defueling was addressed in the licensee's safety evaluation report (84) for the hazard analysis of the potential for boron dilution of the RCS. The analysis for potential boron dilution during defueling was determined to envelope the core sample acquisition activities. Components of the core boring system with hydraulic fluid that could potentially cause local deboration in the core were classified as important to safety in order to minimize the potential for failure of these components. The licensee's safety evaluation considered: (●) various hydraulic fluid leaks; (●) drill bit flushing and cooling; and (●) foreign materials in the reactor core.

  - *Hydraulic Fluid Leak (Hose or Tank Break)*. The main concern was leakage of hydraulic fluid from a hose break or from the reservoir attached to the core drilling machine. The two closed hydraulic systems on the drill unit contained a total of about 27 gallons of Houghto-Safe-620 hydraulic fluid, which was a mixture of 40 percent water, 40 percent glycol, 15 percent polyglycol, and 5 percent additives. Although slightly heavier than water, this hydraulic fluid was completely miscible in water. Therefore, any of this hydraulic fluid that entered the top of the reactor vessel was unlikely to flow down to the core region without being significantly diluted in the water above the core. The only pathway for the hydraulic fluid leakage was from the top of the vessel onto the surface of the vessel water.

There was a low-level sensor on the hydraulic fluid reservoir that would have initiated shutdown of the drill unit within 1 second of detecting a loss of 0.5 gallon of hydraulic fluid. Therefore, leakage from a hose break would be limited to 1.5 gallons. A drip pan with a capacity of 8 gallons would be provided to collect minor leakage. Because of a limit in reservoir tank size, no more than 27 gallons of Houghto-Safe-620 was available to leak into the top of the reactor vessel, even if the controls or drip protection would have failed to limit the leakage. Based on information and analyses presented in the licensee's hazard evaluation report (85) for the potential for boron dilution of the RCS, the evaluation concluded that the reactor coolant would dilute any leakage into the reactor vessel.

- *Drill Bit Flushing and Cooling*. The water used for drill bit flushing and cooling would be supplied from the reactor vessel. The flush water supply tank would be used as a secondary source. The water in the flush water supply tank would be borated to a concentration within the limits that was required by the recovery technical specifications. To ensure that the flush water was adequately borated at the start of the core boring operation, a sample would be taken and analyzed for boron concentration. During the core boring operation, the flush water supply tank would be refilled from the borated water storage tank.

- *Hydraulic Fluid Leak (Drill Unit)*. The evaluation considered a leak of hydraulic fluid from the drill unit that could enter the RCS and possibly cause deborated moderator to be injected as flush water into the drill bit. To prevent this, the relative location of flush water suction would be from an area remote from potential sources of unborated moderators. The flush water suction line arrangement also minimized possible deboration of the flush water in the unlikely event of deboration that resulted from the wrong resins being added to the defueling water cleanup system ion exchangers.

- *Foreign Materials*. The addition of the steel drill casing and core barrel into the core region did not represent a more reactive configuration than the conditions that were analyzed in the licensee's criticality report (86) for the RCS. Additionally, the evaluation concluded that the use of other foreign materials (e.g., cable, camera) during the sample acquisition activities would not increase the effective neutron multiplication of the RCS above 0.99. The licensee's safety evaluation report (87) for early defueling (Section 4.2.1) discussed the process used to review and control these materials.

- *Hydraulic Fluid Leak (Casing Clamp)*. The manually operated underwater casing clamp hydraulic system contained 1.4 gallons of demineralized water. Leakage of this small amount of water was previously shown in the licensee's foreign material report (88) to not be a criticality hazard, even when introduced directly into the rubble bed.

- ***NRC Review: Criticality***. (89) The NRC's safety evaluation agreed that insertion of the drill bit, casing, and core barrel into the reactor core would act as a diluent to the fuel. The NRC also agreed that this would cause the core to have less reactivity and increase the margin to criticality. The water supply used for bit flushing and cooling would be borated in excess of 4350 parts per million. Because of the high boron concentrations, the NRC determined that this water could not dilute the soluble boron concentration in the core region.

There were two potential sources of nonborated fluids: the underwater clamp hydraulic system, which contained 1.4 gallons of demineralized water, and the drill unit hydraulic system, which contained 27 gallons of Houghto-Safe-620 hydraulic fluid. The Houghto-Safe-620 was a mixture of glycols and water (95 percent) and additives (5 percent). This mixture was slightly heavier than demineralized water but essentially neutrally buoyant with respect to the highly borated water in the RCS.

- *Underwater Casing Clamp*. The NRC determined that the demineralized water in the underwater clamp would not present a viable criticality potential for several reasons. First, the quantity would be insufficient to cause a criticality. A criticality would require greater than 2 gallons of demineralized water, even if it was injected directly into the center of the reactor core under worst case conditions. (90) Second, although the demineralized water would be located in the reactor vessel, this water would be several feet from the nearest part of the core and would mix with several thousand gallons of highly borated RCS water before the demineralized water could reach the core. Given that the demineralized water was lighter than borated RCS water, the lighter water would rise to the reactor vessel water surface in the internal indexing fixture as it mixed with the RCS water, rather than sink toward the core.

- *Hydraulic Fluid Leak (Hose or Tank Break)*. The hydraulic fluid was contained in a reservoir and system hoses on the drill rig. The licensee installed a drip pan to collect leakage and a leak detection system with automatic unit shutdown. However, the evaluation considered that, in the worst case, the entire 27 gallons could leak. The hydraulic fluid would leak onto the top of the RCS water at the 327.5-foot elevation. Such a leak would be separated from the reactor vessel nozzles by 12 feet of vertical elevation and over 20,000 gallons of borated RCS water. Being neutrally buoyant and miscible, the hydraulic fluid would mix with the over 20,000 gallons of RCS water located above the reactor vessel nozzles. The core region was located several feet below the nozzles. Mixing with only 225 gallons of 4950-parts-per-million borated water would produce a critically safe concentration. The NRC concluded that the specified hydraulic fluid did not pose a credible source of inadvertent criticality.

- *Conclusion*. The NRC concluded that none of the potential criticality concerns represented a credible or significant safety risk and that the proposed activity was acceptable with respect to criticality issues. (91) The change in work scope to allow the option of boring through the flow distributor plate, if necessary, did not change the criticality considerations previously evaluated by the NRC. (92)

3.3.6 Use of Metal Disintegration Machining System to Cut Reactor Vessel Lower Head Wall Samples

- ***Purpose***. To remove metallurgical samples from the inner surface of the lower reactor vessel head using the metal disintegration machining (MDM) system. An international research program used these samples to study core melt and reactor vessel interactions. This program was conducted after the reactor vessel was defueled.

- ***Evaluation: Criticality***. [93, 94] The licensee's safety evaluation considered potential criticality in the reactor vessel and in the fuel debris located in the containment building basement.

  - *Reactor Vessel*. The potential for a criticality event during vessel wall cutting activities was greatly reduced compared to bulk defueling activities. The reduction was based on the expectation that the maximum amount of fuel that remained at the end of defueling would be less than or equal to 1 percent of the original fuel load. Regardless, a boron concentration equal to or greater than 4350 parts per million (ppm) would be maintained in the reactor coolant system (RCS) during vessel cutting. Additionally, the remaining residual fuel was determined to be in a subcritical configuration. The commitments made in the licensee's safety evaluation report [95] for defueling the reactor vessel regarding criticality safety and deboration control would be met. In addition, all existing recovery technical specifications and required surveillances would be in effect.

  - *Containment Building Basement*. The licensee's safety evaluation report [96] for the lower core support assembly defueling had previously addressed the potential for a criticality event in the containment building basement. The controls that were discussed in the safety evaluation to ensure subcriticality in the basement would be continued during sample cutting.

- ***Evaluation: Boron Dilution***. [97, 98] The licensee's safety evaluation considered the potential for boron dilution from leaks of water-based hydraulic fluids. Various tools that were planned for use during the sample-cutting activities were operated with water-based hydraulic fluids. As with past hydraulic tool operations, all hydraulic fluid used with sampling tools would be borated to at least 4350 ppm natural boron, with two exceptions. This precluded the possibility of a hydraulic fluid leak leading to boron dilution and a possible criticality concern. The two exceptions included the MDM system and expanding seal tool.

  - *MDM Cutting System*. The MDM system was used to cut "boat" samples from the inside surface of the lower reactor vessel head. The cutting tool used a separate hydraulic system with unborated hydraulic fluid. The hydraulic cylinders in the MDM head were driven in a small oscillatory motion (about 0.005 to 0.010-inch amplitude and about 30 to 40-Hertz frequency) during the 10 hours of cutting required for each sample. Testing indicated that when borated hydraulic fluid was used, cylinder seals became badly damaged and prevented cylinder operation within one or two sample cuts. Use of unborated hydraulic fluid provided better lubrication and significantly reduced cylinder seal failures.

Hydraulic cylinder failures during sampling cuts were expected to result in increased radiation doses to personnel involved in repair and refurbishment of the MDM head. Therefore, the use of unborated hydraulic fluid was justified in this application. The hydraulic system was set with a fixed 2-gallon volume for draindown. The physical design of the holding tank limited the maximum uncontrolled loss of hydraulic fluid into the reactor vessel to 2 gallons. This was consistent with the 2-gallon limit for nonborated fluid that was established in the licensee's safety evaluation report [99] on limits of foreign materials allowed in the RCS. As a result, use of this hydraulic fluid was determined to pose no risk of a criticality event due to boron dilution.

- *Expanding Seal Tool*. The other exception was the expanding seal tool, which was used to plug in-core instrumentation penetration tubes after being cut. The tool used less than 1 gallon of unborated hydraulic fluid and was, therefore, also consistent with the 2-gallon limit.

- ***NRC Review: Criticality***. [100] The NRC's safety evaluation considered the potential for inadvertent criticality in the reactor vessel and the containment building basement. At the time of the sampling, all reactor vessel defueling activities were completed, and the remaining residual fuel was determined to not pose a criticality concern. However, given that the verification of this safe configuration was not completed at the time of NRC review, the agency proceeded on the assumption that the amount of fuel necessary for a critical mass could still exist in the reactor vessel.

- *Reactor Vessel*. The licensee committed to remaining within the safety constraints imposed by the safety evaluation report for defueling and Mode 1 technical specifications throughout the sampling program. These constraints included maintaining boron concentration levels to greater than 4350 ppm. This concentration would keep the entire core subcritical in the most reactive geometry possible during defueling with an adequate safety margin. Except as noted in the licensee safety evaluation, all hydraulic fluid used with the sampling equipment would be borated to at least 4350 ppm boron.

- *Containment Building Basement*. The licensee and the NRC had evaluated a wide range of activities that could cause leakage of in-core instrument penetrations. These evaluations [101, 102] were associated with reactor vessel lower head defueling and previous defueling activities. Leakage, through the annular gap between an instrument tube and the reactor vessel wall, could produce a leak of 0.4 gallon per minute for each penetration. This leakage could result from a case in which the in-core instrument penetration and its weld were sheared off while the instrument tube remained in the hole of the reactor vessel wall. Another set of analyses evaluated a case in which an additional unspecified mechanism would force the instrument tube out of the vessel wall. This would result in a 1-inch-diameter hole and 120-gallon-per-minute leak. The licensee had safety systems to make up this potential leakage. These systems included gravity feed from the borated water storage tank and forced circulation via the containment building recirculation pumps. The cavity under the reactor vessel contained borated water to preclude criticality in the event that any fuel was flushed down with the leaking water.

- ***NRC Review: Boron Dilution***. [103] The NRC's safety evaluation considered the potential for boron dilution from leaks of water-based hydraulic fluids. The MDM system equipment would use nonborated water in the hydraulic system. The maximum potential loss from this system was 2 gallons. The licensee and the NRC had previously analyzed localized deboration due to leakage of 2 gallons of nonborated water, as reported in their safety evaluation reports [104] on the introduction of foreign materials in the RCS. These analyses involved the most reactive geometry of a full core. Both the licensee and the NRC had concluded that localized deboration from up to 2 gallons of nonborated water would not pose a criticality hazard. Additional analyses [105] were performed during the evaluation of the plasma arc cutting system. These analyses considered less than a full core but much more fuel and a more reactive geometry than was expected to exist during lower head sampling. This conservative analysis assumed 3 gallons of nonborated water. The results indicated that there was an adequate margin of safety (i.e., greater than 1-percent delta-k with an additional 2.5-percent delta-k uncertainty factor).

3.4 Pre-Defueling Preparations

3.4.1 Containment Building Decontamination and Dose Reduction Activities

- ***Purpose***. To conduct decontamination and dose reduction activities in the containment building at elevation levels 305-feet (entry level) and above. Planned activities included: (●) flushing containment building surfaces with deborated water; (●) scrubbing selected components of the polar crane; (●) conducting hands-on decontamination of vertical surfaces; (●) decontaminating the air coolers; (●) flushing the elevator pit; (●) cleaning floor drains; (●) shielding the seismic gap and penetration; (●) decontaminating and shielding hot spots; (●) decontaminating missile shielding; (●) shielding reactor vessel head surface structure; (●) removing concrete and paint; (●) decontaminating cable trays; (●) decontaminating equipment; (●) remote flushing the containment building basement; and (●) testing remote decontamination technology.

- ***Evaluation: Boron Dilution***. [106] The licensee's safety evaluation concluded that criticality of fuel, either in the core region or outside the core, was precluded by means of poison, geometry, or quantity of fuel. The activities that could increase the potential for a criticality event during decontamination were the dilution of the reactor coolant during water flushing activities and the dilution of borated sump water that could be used as an emergency source of injection into the reactor vessel.

  - *Reactor Coolant System (RCS)*. During water flushing activities in the containment building, potential boron dilution of the RCS through two pathways was considered. These pathways were through the open reactor vessel and through openings in other components of the RCS.

– *Open Reactor Vessel*. Administrative controls would prevent boron dilution through the open reactor vessel. Flushing directly above or around the top of the reactor vessel would not be permitted during defueling activities with shield panels removed from the defueling work platform. When decontamination operations presented a possible dilution pathway through the open vessel, all defueling work platform shield panels or a temporary support structure would be in place and an impervious covering would be installed, to prevent unborated water from entering the reactor vessel. Water borated to RCS levels could be used for decontamination when there was a potential for its introduction into the RCS.

– *RCS Openings*. During decontamination activities in the D-rings, [q] RCS components would be flushed with water. As part of the ex-vessel fuel location program, openings could be created in the RCS by removal of instrument ports and manways in the steam generators or pressurizer. Boron dilution of the RCS through these pathways would be prevented by one of the following methods:

  ▪ *Uncovered Openings*. When uncovered openings in RCS components existed, no water flushing would take place in the affected D-ring. Other decontamination methods that would not present a credible boron dilution hazard, such as hand scrubbing or vacuum techniques, could be used.

  ▪ *D-Rings*. Water flushing in the affected D-ring could take place only when the following conditions were met: (●) an impervious cover was used to seal the opening; (●) water flushing would not be performed in the area of the opening; and (●) water was borated to RCS levels. Procedures that verified the covered installation and its condition would be used in these circumstances.

  ▪ *Covered Openings*. Any replacement hatches or covers for RCS components would be capable of withstanding the decontamination water spray directed on the covers without leakage into the RCS equipment.

o *Containment Building Sump*. In the past, decontamination water that was used in the containment building for surface decontamination contained boron for the purpose of controlling criticality control. The two issues that involved unborated water in the containment building sump were subsequently addressed.

– *Sump Debris Criticality*. First, past evaluations conservatively assumed that the basement could contain sufficient fuel to present a criticality hazard if unborated water was used in quantity. The licensee's subsequent criticality evaluation report [107] for the containment building sump, which was based on measured data and analysis, demonstrated that the maximum quantity of fuel that could be located in the basement was insufficient to create a criticality concern under credible conditions. Before the NRC's approval of the licensee's criticality evaluation of the sump, water borated to

[q] These are the shield enclosures around the steam generator compartments; they were so named because of their shape.

1700 parts per million (ppm) would be used for general decontamination of the containment building. However, the use of unborated water for decontamination was preferred for the following reasons: (●) Accumulation of boron salts on decontaminated surfaces increased recontamination by increasing the mobility of contaminants. (●) Boron salts increased airborne radioactivity potentials and increased slip and fall hazards. (●) Borated water increased wear on decontamination equipment and limited the use of decontamination techniques, such as ultrahigh pressure water flush. Therefore, the use of unborated water for surface decontamination was preferred for normal surface decontamination throughout the containment building.

– *Sump Water Dilution*. If RCS integrity were lost, water from the basement could be used as a source of borated water injection into the reactor vessel. The recirculation water from the sump recirculation system would contain a maximum of 4350 ppm boron. This requirement would exist as long as a quantity of fuel equivalent to a critical mass was located in the RCS.

  - *Administrative Controls*. Requirements were established to ensure that sump water used for the emergency injection into the reactor vessel contained no less than 4350 ppm boron. To ensure this minimum concentration, no more than 70,000 gallons of unborated water from cleanup activities could be allowed to accumulate in the basement and sump. A previous safety evaluation report [108] for the technical specification requirement for the containment building sump recirculation system provided the basis for this inventory of unborated water in the sump.

    Administrative controls would be used to ensure that the maximum allowable quantity of unborated water in the basement was not exceeded. From a decontamination standpoint, the maximum allowable accumulation of 70,000 gallons in the basement was not a very restrictive limit and should not significantly impact water flushing operations. The licensee would maintain records of uses of all low borated and unborated water for decontamination purposes, as well as water inventory in the containment building. Activities would be coordinated to match water processing capabilities with decontamination use.

  - *Recirculation Pump Protection*. The intake for the portable pumps, which recirculated water from the containment building basement to the reactor vessel, was equipped with a screen with slots of 0.375 inch by 1.5 inches, which prevented large debris from entering the pump. The pump was designed to pump water with any entrained debris that could pass through the slots without damage to the pump. If the screen became clogged during operation, the screen could be cleared by momentarily shutting the pump off, or relocating the pump, or both. Therefore, the evaluation concluded that debris in the basement would not preclude recirculation.

  - *Reduced Moderation from Solid Materials*. Solid materials (e.g., concrete and paint chips) that could be removed from contaminated surfaces during decontamination activities could eventually accumulate in the containment building basement. These materials could be introduced into the reactor vessel in the unlikely event that the recirculation mode was used to maintain the reactor vessel water level. These solid materials were not expected to replace borated water as the primary moderator; therefore, based on the licensee's safety evaluation report [109] on limits of foreign materials allowed in the RCS, there was essentially no limit on the amount of solid material that could be introduced into the reactor vessel. Consequently, the evaluation concluded that the solid foreign materials, which were removed from contaminated surfaces during decontamination activities and transported to the reactor vessel during recirculation operations, would not cause a criticality safety concern.

- *Fuel Transfer Canal (FTC) Water*. The boron concentration in the FTC would be maintained at a level of 4350 to 6000 ppm according to the recovery technical specification. Any decontamination activity that could introduce water borated to levels less than 4350 ppm into the FTC would be evaluated to ensure that the operation would not dilute the FTC boron to a level below the technical specification limit. Adequate means such as FTC water level monitoring or boron sampling would be available during decontamination activities to ensure that the technical specification boron level was maintained.

- ***NRC Review: Criticality/Boron Dilution***. [110] Editor's Note: The NRC's safety evaluation report did not specifically address this evaluation topic. However, other NRC safety evaluations [111, 112, 113] addressed containment building sump criticality, limits of foreign materials allowed in the RCS, and the technical specification requirement for the containment building sump recirculation system.

3.4.2 Reactor Coolant System Refill

- ***Purpose***. To refill the reactor coolant system (RCS) to the top of the hot legs in order to purge oxygen and to provide an RCS water level that would permit operation of the once-through steam generator (OTSG) recirculation/cleanup system. To operate the OTSG recirculation/cleanup system, the secondary-side water level in the OTSG must be raised to the vicinity of the upper tubesheet to minimize the chance of unborated water leakage from the OTSGs to the RCS.

As an added protection against system overpressurization, the pressurizer would not be vented. This protective measure provided a surge volume for increases to the RCS or for inadvertent introduction of pressurization to the RCS, such as by activating pumps or changing valve lineups.

- ***Evaluation***. [114, 115] Editor's Note: The licensee's safety evaluation report did not specifically address these topics.

- ***NRC Review***. Editor's Note: The NRC's safety evaluation was not located.

3.4.3 Reactor Vessel Head Removal Operations

3.4.3.1 Polar Crane Load Test

- ***Purpose***. To conduct load testing of the polar crane in preparation for reactor vessel head lift and removal. The polar crane load test required four major sequential activities: (1) relocation of the internals indexing fixture; (2) assembly of test load; (3) load test; and (4) disassembly of test load.

- ***Evaluation: Criticality***. [116] The licensee's safety evaluation considered the consequences of a heavy load drop in the vicinity of the reactor core. The evaluation showed that such an event would not exceed the limits set by the evaluation criteria of NUREG-0612. Criterion 2 required that accidental dropping of a postulated heavy load would not result in a configuration of the fuel such that the effective neutron multiplication (k_{eff}) was larger than 0.95. Criterion 4 required that damage to redundant or dual safe-shutdown paths would be limited so that required safe-shutdown functions would not be lost. These functions included systems that were required to reach and maintain subcriticality.

  - *Reactor Vessel*. The precise configuration of the fuel before head removal was unknown at the time of this analysis. Therefore, the exact k_{eff} that would result from the potential redistribution of the fuel due to the impact of a missile shield block on the reactor vessel head could not be calculated. Despite the inability to calculate the exact k_{eff}, bounding analyses [117, 118] were performed for the 100-percent fuel damage case. These analyses concluded that the fuel debris would not be critical when the debris was in its most reactive condition. The analyses accounted for the effects of structural material and a reactor coolant boron concentration of 3000 parts per million (ppm). In view of these results and because the concentration of boron in the reactor coolant was over 3500 ppm, the evaluation concluded that a criticality was precluded.

  - *Steam Generator*. The following sequence of low-probability events would have to occur in sequence, each conditioned on the occurrence of all the prior low-probability events, before a criticality in the steam generator could occur: (1) A missile shield must be dropped above one of the D-rings. (2) The missile shield must travel far enough into the D-rings to impact a reactor coolant pump or cold-leg piping. Note that there were massive structural beams crossing the D-ring above the reactor coolant pump elevation where the reactor coolant pumps were vertically supported. (3) A missile shield must impact the reactor coolant pump or other structure in such a way as to rupture the pump suction line at a point well below the secondary-side water level. (4) An amount of fuel sufficient to raise criticality concerns must have been transferred to the steam generator during the accident, migrated into the steam generator tubes, and lodged in the tubes. (5) This fuel must be in a high-density, close-pack critical configuration inside the tubes, with borated water drained from the tubes and the tubes surrounded by unborated secondary water.

Even with the application of conservative probabilities to each event of the required sequence, the evaluation concluded that the probability of criticality was below any reasonable threshold to be a safety concern.

- ***Evaluation: Boron Dilution***. [119] The licensee's safety evaluation addressed the criteria in NUREG-0612 for criticality issues resulting from boron dilution events that could be caused by a heavy load drop. The evaluation considered boron dilution of the reactor vessel coolant and the borated containment building sump water.

  - *Reactor Vessel*. Because of the configuration of TMI-2, deboration of the reactor coolant system (RCS) water was determined to be the only credible mechanism that could compromise criticality control. Systems within load impact areas that contained unborated water were found to fail in such a way that these systems could drain their contents into the containment building basement and not into the RCS. For example, gross leakage from the secondary side of the steam generator into the primary side, as a result of a postulated load drop, was a possible scenario for boron dilution in the RCS. However, the evaluation concluded that damage to the steam generators that resulted in leakage would undoubtedly cause damage to the outer surface of the steam generators. Such damage would allow unborated water to drain from the secondary side to the containment sump. Additionally, systems capable of injecting highly borated water into the RCS were readily available. A single load drop could not feasibly disrupt boron injection.

  - *Containment Building Sump*. For the containment sump, the only potential concern was the drop of a heavy load onto the systems that could provide a source of unborated water to the sump. The evaluation identified two ways to address this concern. First, to limit the amount of unborated water available for leakage to the sump from these systems, the water supply to these systems would be isolated during load testing. (Note: The operable fire protection system was normally isolated.) The evaluation showed that a significant quantity of unborated water was required to lower the sump water concentration from existing values to a level below the 1700-ppm value. This concentration was specified as the reasonable concentration for avoiding reactivity issues in the sump.

    Second, to induce criticality, it would be necessary for a critical mass of fuel to have collected in the sump with an optimal configuration, regardless of the amount of unborated water delivered to the sump. The evaluation stated that the events resulting in this condition had low probabilities. A qualitative assessment indicated that the administrative controls effectively eliminated criticality as a safety concern. These controls limited the amount of unborated water that could be delivered to the sump. Given these controls and the low probability of simultaneous occurrence of the initial conditions for fuel deposition in the sump that could lead to a criticality problem, criticality concern was considered a low-probability event.

- ***NRC Review***. [120] The NRC's safety evaluation considered the potential for criticality in the core from an impact-induced rearrangement of fuel and debris. A number of criticality analyses were previously performed as part of the NRC's safety evaluation report [121] for the axial power shaping rod insertion tests that postulated fuel redistribution. The NRC concluded that these previous analyses bounded the crane load test.

3.4.3.2 First-Pass Stud Detensioning for Head Removal

- ***Purpose***. To perform the first-pass detensioning of the 60 reactor vessel studs and the removal of up to 5 reactor vessel studs to check for stuck nuts and to examine the condition of the removed studs.

- ***Evaluation: Criticality***. [122, 123] The licensee's safety evaluation postulated that detensioning could cause a very slight movement of the plenum assembly, but this movement would be less than 0.007 inch. The evaluation did not consider a plenum movement of this magnitude causing a gross rearrangement of the existing core configuration to be credible.

- ***NRC Review***. [124, 125] Editor's Note: The NRC's safety evaluation report did not specifically address this evaluation topic.

3.4.3.3 Reactor Vessel Head Removal Operations

- ***Purpose***. To prepare for reactor vessel head removal to perform the lift and transfer of the head and conduct the activities necessary to place and maintain the reactor coolant system (RCS) in a stable configuration for the next phase of the recovery effort.

- ***Evaluation: Criticality***. [126][r] The licensee's safety evaluation considered criticality in the reactor vessel and containment building sump.

  - *Subcriticality (Reactor Vessel)*. The licensee's safety evaluation stated that criticality calculations of shutdown margins for postulated core configurations was previously documented in the Babcock & Wilcox (B&W) criticality evaluation report [127] for recovery activities through head removal, which was issued before the camera insertion (Quick Look) through the leadscrew opening. After the camera inspection, the observed core features were compared to the postulated damage models that were used in those calculations to assess the validity of the estimated shutdown margins. Addendum 1 to the B&W report [128] concluded that the previous analysis was bounding and remained valid for recovery

[r] Editor's Note: An additional criticality evaluation was performed for the load drop evaluation as required by NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," issued July 1980 ". Refer to NUREG/KM Chapter 7 on load drop evaluations for details of NUREG-0612.

operations through vessel head removal, since the previous analysis assumed a greater amount of fuel damage than was observed during the camera inspection.

The analysis in the original B&W report used conservative core configurations that were assumed to represent worst case conditions for recovery operations through head removal activities, except for major core rearrangement associated with the head drop on the reactor vessel. These static configurations included a maximum credible core damage model (50-percent core damage in a debris bed over 50-percent intact fuel assemblies) and a model for 100-percent core damage. In addition, fuel in the vessel but outside of the core region was analyzed. Models used included a sphere of 50 percent of the damaged highest enrichment fuel (19 assemblies) in the bottom of the reactor vessel, a hemisphere of 50 percent of the core in the lower vessel, and a cylinder of fuel particles falling down from the core region.

The evaluation concluded that for all postulated conditions, subcriticality was maintained with a reactor coolant boron concentration of 3500 parts per million (ppm).

- *Subcriticality (Containment Sump).* With regard to subcriticality in the containment building sump, an evaluation indicated that the only point of potential concern would be the drop of a heavy load onto the systems that could provide a source of unborated water to the sump. Potential sources of unborated water in the containment building included: (●) fire protection system; (●) demineralized water system; (●) nuclear services closed cooling water system; (●) intermediate closed cooling water system; (●) normal cooling water system; (●) nuclear services river water system; (●) main steam and feedwater systems; and (●) decontamination water.

The safety evaluation report stated that this problem could be addressed in two ways. First, to limit the amount of unborated water available for leakage to the sump from these systems, the water supply to these systems would be isolated for the period of time that the head lift was being performed. (Note: The operable fire protection system was normally isolated.) An evaluation showed that a significant quantity of unborated water would be required to lower the sump boron concentration from current values to a level below the value of 1700 ppm, which was specified as adequate for avoiding sump criticality.

Second, several low-probability events would need to occur before valid concerns about sump criticality could arise, regardless of the amount of unborated water delivered to the sump. First, an amount of fuel sufficient to create a critical mass would have needed to wash into the sump during the TMI-2 accident. Then, this fuel would have to be in a configuration that could induce criticality if a global boron dilution of the sump were to occur.

A qualitative assessment indicated that the administrative controls described above effectively eliminated criticality as a safety concern. These controls limited the amount of unborated water that could be delivered to the sump. Combined with the low probability of simultaneous occurrence of the initial conditions for fuel deposition in the sump that could lead to a criticality problem, criticality was considered a low-probability event.

- ***Evaluation: Boron Dilution***. [129] Editor's Note: The licensee's safety evaluation for reactor vessel heard removal was almost identical to the detailed evaluation provided in its safety evaluation report [130] for reactor vessel underhead radiation characterization program. The following presents only the differences.

  - *Actions To Prevent Boron Dilution*. Additional boron dilution sources that would apply specifically to head removal operations included the following:

    – *Internals Indexing Fixture (IIF) Processing System*. After installation of the IIF and the IIF processing system, a potential boron dilution pathway through the transfer canal drain manifold was identified. This pathway was from the fuel transfer canal drain system and the sump sucker system that connected to the IIF processing system at the manifold. Upon installation of the IIF, the hose from the IIF pump to the manifold would not be connected, and the IIF processing system isolation valve (FCC-V002) would be closed. This provided double isolation of any new potential flowpath through the IIF processing pump. In addition, the hose to the fuel canal drain system would not be connected to the manifold, and branch valve FCC-V002 would be closed. Boron dilution potential during operation of the IIF processing system was not within the scope of this safety evaluation.

    – *Other Systems*. After the head was removed, the open reactor vessel provided an additional path for water to enter the vessel. Additional sources of unborated water or water borated to concentrations less than 3500 ppm were fire service water, decontamination water, and containment building spray system water. These systems were normally isolated by at least two isolation boundaries. When the systems were not isolated, administrative procedures would control use of fire service water and decontamination water in the containment building. Any water spray system that was used to control airborne radioactivity from the exposed plenum immediately after head removal would be borated to at least 3500 ppm. The IIF cover would limit the entry of water into the open vessel once the cover was installed, and an additional waterproof covering would be placed over the work platform when decontamination water was used directly over the vessel.

  - *Actions To Monitor Boron Content*. Additional means for monitoring a boron dilution event that would apply specifically to head removal operations included the following:

    – *RCS Level Indication*. The IIF was equipped with a level monitoring system in addition to the other systems that would indicate a drop in the RCS.

    – *Boron Measurements*. After head removal, the weekly RCS sample would be taken from the IIF cover by inserting a sample tube into the reactor vessel in the core void region. The time period between head removal and the installation of the IIF and the cover was expected to be less than 1 week to avoid any interruption in the RCS sampling schedule. The means to obtain an RCS sample at the required frequency would be available, even if the IIF or the cover was not successfully installed within the 1-week period.

– *Neutron Monitoring*. Editor's Note: The source range neutron instrumentation was not mentioned for reactor vessel head removal.

• ***NRC Review: Criticality***. [131] The NRC's safety evaluation concluded that there was little potential for core criticality, either by core reconfiguration or boron dilution. Further, the NRC found that the conservative assumptions used in the licensee's analyses ensured that an adequate degree of subcriticality would be maintained for any credible core configuration. The analyses also indicated that the damaged fuel outside the core would not achieve criticality under postulated credible worst case conditions.

The postulated damage models effectively bounded all credible fuel configurations, including those resulting from a reactor vessel head drop accident for the criticality analyses. For these analyses, the maximum boron concentration in the moderator was assumed to be 3500 ppm. All of the analyzed scenarios yielded effective neutron multiplication (k_{eff}) values below 0.99. The realistic k_{eff} values were calculated to be less than 0.90, indicating that the models incorporated substantial conservatism. Each analyzed case included several conservative assumptions: (●) hypothetical 100-percent fuel failure; (●) no neutron leakage; (●) no neutron absorption by structural or poison material; (●) no fuel burnup; (●) maximum fuel enrichment; and (●) optimum fuel-moderator ratio. All parameters affecting reactivity were optimized for models that assumed a more credible fraction of failed fuel. For the out-of-core criticality analysis, one model assumed that a sphere of 19 assemblies of the highest fuel enrichment (3 percent) collected in the lower vessel. Another model assumed that 50 percent of the core formed a hemisphere in the bottom of the vessel. These analyses demonstrated that 3500-ppm boron was adequate to maintain the fuel debris in a subcritical condition for all credible core damage models.

Even though 3500-ppm boron was adequate to maintain subcriticality for all credible core configurations, the licensee raised the boron concentration in the reactor coolant to 5000 ppm as an added margin of safety that bounded all potential core configurations. The associated analyses indicated that this concentration would maintain subcriticality for any postulated core configuration. Therefore, the NRC concluded that there was virtually no potential for criticality during reactor vessel head lift for any postulated core reconfiguration.

• ***NRC Review: Boron Dilution***. [132] The NRC's safety evaluation considered the measures taken by the licensee to prevent a boron dilution incident and their proposed corrective actions to ensure subcriticality of the core in the unlikely event of such an incident.

○ *Dilution Sources*. Potential boron dilution sources included those systems that connected with the RCS and contained unborated water or borated water of a lower concentration than in the RCS. Some of these systems included: (●) demineralized water system; (●) decay heat removal system; (●) mini-decay heat removal system; and (●) secondary cooling water systems. There was a minimum of two isolation barriers for each potential in-leakage pathway to prevent dilution of the RCS. Any combination of these barriers connected to the RCS included: (●) closed, tagged-out valves; (●) electrically locked-out pumps; (●) removed spool pieces; (●) heat exchanger tube boundaries; and (●) water pressure differentials. Periodic surveillance of valve positions and storage tank water levels in the systems would identify any potential dilution pathways and reduce the potential for boron dilution from these sources.

- *RCS Makeup and Processing*. Systems necessary for makeup and processing of RCS water during head lift activities would be borated to the same concentration as the RCS. All makeup and processing activities would be performed in accordance with approved procedures along with the samples taken to verify the boron concentration in makeup sources. Samples of RCS water would be taken and analyzed weekly, at a minimum, to verify boron concentrations. Two independent water level monitors (a third water level monitor would become available after the installation of the IIF) would provide continuous indication of RCS water level and an early indication of a dilution event, should one occur. Water level indication would ensure that corrective actions could be employed to isolate the dilution source.

- *Additional Dilution Pathways*. Head removal created an additional potential dilution path, because of the possibility that water from sources within the containment could enter the RCS through the open reactor vessel. These sources, which included fire service water, decontamination water, and the containment building spray system water, were normally isolated by at least two isolation boundaries. Administrative procedures would control these systems when not isolated. If spray systems had to be used to control airborne activity, water sources would be borated to the RCS concentration. After the installation of the IIF, a cover would be provided to limit entry of water through the open vessel, and when decontamination water was used near the vessel, an additional cover would be placed over the work platform.

- *Contingency*. If, despite these preventive measures, a boron dilution event did occur, contingency procedures existed to rapidly isolate the pathway or to inject boron into the RCS, as necessary to maintain a subcritical condition at a sufficient concentration. The NRC determined that the surveillances in place (e.g., RCS water level) would rapidly identify a dilution event in sufficient time to correct the problem.

- *Conclusion*. The NRC concluded that the preventive measures and corrective actions described by the licensee provided adequate assurance to preclude the occurrence of a boron dilution event during head removal activities.

3.4.4 Heavy Load Handling inside Containment

- ***Purpose***. To ensure that handling of heavy loads in the containment building and spent fuel pool “A” (SFP-A) within the fuel handling building was in accordance with the safety requirements of NUREG-0612, “Control of Heavy Loads at Nuclear Power Plants,” [133] issued July 1980.

- ***Evaluation: Criticality/Boron Dilution***. [s, 134] The licensee's safety evaluation considered the consequences of a heavy load drop in the vicinity of the reactor core. The evaluation showed that such an event would not exceed the limits set by the evaluation criteria of NUREG-0612. Criterion 2 required that accidental dropping of a postulated heavy load would not result in a configuration of the fuel such that the effective neutron multiplication (k_{eff}) was larger than 0.95. Criterion 4 required that damage to redundant or dual safe-shutdown paths would be limited, so that required safe-shutdown functions would not be lost. These functions included systems that were required to reach and maintain subcriticality.

This evaluation considered criticality inside the containment building, spent fuel pool, fuel transfer canal, and reactor vessel.

- *Inside Containment*. Areas inside the containment building were classified as an unrestricted lift area if all unborated water sources were isolated. This classification was based on the licensee's safety evaluation reports [135, 136] for the reactor vessel head removal and the polar crane load test. These evaluations demonstrated that load drops in these areas would not: (●) result in draining of the reactor vessel; (●) impact the availability of makeup to the reactor vessel; or (●) result in an inadvertent criticality. Criticality was prevented by the isolation of unborated water sources. As an alternative to isolating all unborated water sources for each heavy load handled inside containment, adherence to the load weight and height guidelines provided by the safety evaluation ensured that a dropped load would not cause the floor slab to fail. Consequently, the unborated water systems located beneath the floor slab where a load was being carried were not required to be isolated. However, any unborated water systems that could be directly impacted by a load drop within the area of a particular load handling activity would be isolated until completion of the activity.

- *Spent Fuel Pool and Fuel Transfer Canal*. The handling of heavy loads over the fuel canister storage racks in SFP-A and in the deep end of the fuel transfer canal (FTC), with filled or partially filled canisters in the racks, would be restricted so that the potential energy of the load would not be greater than the potential energy of a suspended fuel canister. This ensured that the storage racks would not be damaged to such an extent as to cause a criticality. Also, the licensee's technical evaluation report [137] for defueling canisters described an analysis that determined the potential for criticality to occur in SFP-A/FTC because of a catastrophic failure of the liner that caused the SFP-A/FTC to be drained of water. This analysis determined reactivity would continue to be controlled, if the level of borated water in the SFP-A/FTC was maintained and that a criticality event would not occur.

- *Reactor Vessel*. The licensee's safety evaluation report [138] for load handling over the reactor vessel discussed this topic and the associated safety issues. Reactivity would continue to be controlled if the level of borated water in the RCS was maintained. Therefore, dropping of a heavy load was determined to affect reactivity control only if the load drop resulted in breakage of in-core instrument tubes and subsequent draining of the reactor

[s] Editor's Note: An additional criticality evaluation was performed for the load drop evaluation, as required by NUREG-0612. Refer to NUREG/KM Chapter 7 on load drop evaluations for details of NUREG-0612.

vessel below the 314-foot elevation. However, the evaluation concluded that postulated load drops would not break in-core instrument tubes because there were no in-core instrument tubes outside of the load handling exclusion areas.

- ***NRC Review: Criticality***. [139, 140] The licensee's and NRC's safety evaluations considered cases in which a canister was dropped on another canister, either in the FTC or the spent fuel pool. The entire contents of the upper canister were assumed to spill and to form the worst-case geometry, including highest enrichment fuels surrounding the lower canister. No credit was taken for zirconium cladding material, poison (criticality control) materials incorporated in the canister, and the structural materials of the canister. The NRC's criticality evaluation report [141] for a loaded canister dropping its load onto another canister concluded that a considerable shutdown margin (i.e., k_{eff} less than 0.95) would exist. The licensee also examined the case of an infinite array in a drained fuel pool condition and concluded that subcriticality would be maintained with a k_{eff} less than 0.964. The infinite array scenario required dozens of consecutively dropped canisters. The NRC considered this scenario not to be credible.

3.4.5 Heavy Load Handling over the Reactor Vessel

- ***Purpose***. To permit the handling of heavy loads (loads greater than 2400 pounds including rigging weight) in the vicinity of the reactor vessel. This included any load handling activity that could result in a heavy load drop onto or into the vessel either directly or indirectly (as the result of the collapse of structures or equipment installed over the vessel).

- ***Evaluation: Criticality.*** [142] The licensee's safety evaluation concluded that the reactor vessel core debris bed would remain subcritical if the reactor coolant system (RCS) was maintained at a boron concentration of 4350 parts per million (ppm) or greater. This conclusion was based on the licensee's criticality evaluation report [143] for the RCS and its hazard analysis report [144] for the potential for boron dilution of the RCS. Following the loss of RCS water as a result of a heavy load drop accident, the subsequent operation of the makeup and recirculation systems would keep the core debris covered and would not decrease the boron concentration in the RCS below 4350 ppm. Therefore, the core would remain subcritical following postulated load drop accidents.

A postulated load drop accident into the core debris bed materials could act as a neutron moderator or could cause localized deboration. This accident could present the potential for localized criticality events. Before the removal of the plenum assembly, loads that could be dropped into the reactor vessel would impact the top of the plenum; thus, the loads would not contact the core debris bed. In addition, after the installation of the defueling work platform, loads handled in accordance with the height and weight guidelines (refer to Section 3.1.2.1 of the safety evaluation report) would not, if dropped, contact the debris bed if the tool slots in the platform were closed or if the load could not fit through the tool slots in any orientation. Consequently, for these cases, no restrictions on materials would be required. For all other heavy load handling activities over the vessel, to ensure that localized criticality would not occur, the materials of each load would be evaluated on a case-by-case basis before performing the load handling activities.

- ***NRC Review: Criticality***. [145] The portable sump recirculation system had the capability to inject sump water into the reactor vessel if needed. The system capacity (20 gallons per minute) was sized for a leak of an in-core instrument nozzle penetration at the lower reactor vessel head. The NRC's safety evaluation stated that ample time would be available to sample the containment building sump before initiating the recirculation, thereby giving assurance that water with a boron concentration below 4350 ppm would not be pumped into the core. The NRC's safety evaluation report [146] for the technical specification requirement for the containment building sump recirculation system addressed this boron concentration and resulting effective neutron multiplication.

3.4.6 Plenum Assembly Removal Preparatory Activities

- ***Purpose***. To confirm the adequacy of plenum removal equipment and techniques, the preparatory activities included: (●) video inspection of potential interference areas; (●) video inspection of the core void space; (●) video inspection of the axial power shaping rod (APSR) assemblies; (●) measurement of the loss-of-coolant accident restraint boss gaps; (●) measurement of the elevations of the APSR assemblies; (●) cleaning of the plenum and potential interference areas; (●) separation of unsupported fuel assembly end fittings; and (●) movement of the APSR assemblies.

- ***Evaluation: Criticality***. [147] The licensee's safety evaluation stated that manipulations within the plenum assembly and the core cavity would be performed with extreme care and within given constraints because of their potential to disturb the core. The dislodging of fuel assembly end fittings would be performed with the following prerequisites: (1) the core inspection revealed that the end fitting was unsupported (no full-length rods existed) and (2) all unsupported end fittings from Batch 1 and Batch 2 fuel were dislodged before dislodging any unsupported end fittings from Batch 3 fuel, which had a higher enrichment.

Implementation of these two operational constraints would ensure that the potential reactivity consequences would be negligible compared to the worst case core configurations analyzed for a postulated head drop accident, which were presented in the licensee's safety evaluation report (SER) [148] for the removal of the reactor vessel head. Core topography showed, and planned video inspections would verify, that the APSR assemblies were unsupported and could therefore be driven into the core region. Expectant core disruptions would be no more severe than the dislodging of unsupported end fittings. If the APSR assemblies could not be driven into the core, then attempts would be made to withdraw the assemblies into the plenum assembly, provided that they were observed to have no full-length rods; otherwise, their vertical travel would be limited to 5 inches. The inadvertent dropping of tooling was not expected to disrupt the core to an extent greater than the dislodging of unsupported end fittings.

The licensee concluded that, with a boron concentration of 3500 parts per million (ppm), the core would remain subcritical even for the worst case reconfiguration of fuel.

- ***Evaluation: Boron Dilution***. [149] The licensee's safety evaluation concluded that the stabilization of the core was assured since preventive measures to preclude a boron dilution event for inspection activities were taken. Measures specified in the licensee's SER [150] for the removal of the reactor vessel head and the licensee's SER [151] for the operation of the internals indexing fixture processing system would remain in effect to monitor reactor coolant system (RCS) boron content and to preclude inadvertent boron dilution of the RCS. Monitoring parameters would include water level inside the internals indexing fixture, RCS coolant sampling for boron concentrations, and mass balance of the RCS inventory.

- ***NRC Review***. [152, 153] Editor's Note: The NRC issued two SERs; the first SER covered the first five activities (see "Purpose" of this section), and the second covered the remaining three activities. The first SER was brief and did not specifically address this topic. However, the first safety evaluation stated that the first five activities were previously addressed in the NRC's SERs [154, 155] for Quick Look video inspection of the reactor core and for the reactor vessel underhead characterization. The NRC's safety evaluation stated that the licensee's prior experience conducting core and plenum video inspections and other in-vessel activities (e.g., radiation measurements, reactor coolant sampling) showed that these were benign activities (i.e., environmental impacts were very small), which posed little risk to the onsite workers or offsite public. The NRC further stated that the corresponding plenum inspection activities did not warrant further review.

- ***NRC Review: Criticality***. [156] The NRC's second safety evaluation of the remaining three activities stated that many of the safety issues, including criticality, relevant to plenum removal preparatory activities were addressed in the NRC's SER [157] for the reactor vessel head lift. The potential for criticality of the damaged reactor core for any fuel reconfiguration resulting from the proposed activities was effectively precluded by maintaining a sufficiently high boron concentration in the reactor coolant. The NRC concluded in the head lift evaluation that, at the current concentration of 5050 ppm ±100 ppm, there was virtually no potential for criticality for any postulated fuel configuration. For this reason, the NRC agreed that unsupported end fittings, including those associated with Batch 3 fuel assemblies, could be dislodged without a sequence constraint.

- ***NRC Review: Boron Dilution***. [158] The NRC's second safety evaluation of the remaining three activities stated that dilution of the boron concentration in the RCS could result in a criticality event if the dilution continued unchecked. The NRC's SER for the reactor vessel head removal and its SER [159] for the internals indexing fixture operations addressed the potential for boron dilution during previous TMI-2 cleanup activities. The NRC concluded in these evaluations that the licensee's measures for prevention, detection, and mitigation of a potential boron dilution event provided adequate assurance that subcriticality would be maintained for all postulated conditions. These measures, which would be in effect during the proposed activities, included prevention of a dilution event through the use of double isolation barriers for all potential dilution sources and the maintenance of all RCS makeup sources at the required RCS boron concentration.

Methods to detect a dilution event included periodic monitoring of the RCS boron concentration and RCS inventory measures. Potential dilution sources could be identified and isolated and, as necessary, borated makeup water would be injected into the RCS. Measurement of the boron concentrations at various elevations within the reactor vessel following makeup (or letdown and makeup) operations indicated that there was good mixing within the vessel to generate homogeneous boron solutions and that boron did not stratify, even under stagnant conditions.

The RCS boron concentration had recently been increased to about 5000 ppm from 3500 ppm, the value shown in the NRC's SER for head removal to be a sufficient concentration to prevent criticality under all credible postulated conditions. If a dilution event did occur, the frequent boron sampling, in conjunction with the large margin provided by the high RCS boron concentration, would provide sufficient time for the detection and mitigation of the dilution. The NRC concluded that preventive measures were adequate to make a boron dilution event during preparatory activities for plenum removal extremely unlikely.

3.4.7 Plenum Assembly Removal

- ***Purpose***. To remove the plenum assembly from the reactor vessel to gain access to the core region for defueling. The plenum was moved to the deep end of the fuel transfer canal, which contained reactor coolant for shielding.

- ***Evaluation: Criticality***. (160) The evaluation considered that a postulated mechanical failure of the polar crane or its rigging could result in a plenum assembly drop. The evaluation concluded that the postulated worst case plenum assembly drop would not uncover the fuel in the reactor vessel or cause criticality. The initial lift and the associated dislodging of fuel assembly end fittings and plenum cleaning had the potential to disturb the core. However, the analysis presented in the licensee's criticality evaluation report (161) for the reactor coolant system (RCS) showed that the core would remain subcritical with any core configuration, provided that the boron concentration in the RCS remained above the limit specified in the report. The licensee's hazard analysis report (162) on the potential for boron dilution of the RCS described the actions that should be performed to preclude a boron dilution. The total quantity of hydraulic fluid present in the hydraulic jacking system was about 8 gallons. Thus, it was determined that a postulated total loss of the hydraulic fluid into the RCS would not significantly dilute the boron concentration in the RCS and would not impact the subcriticality of the core.

- ***Evaluation: Boron Dilution***. (163) At the time of the evaluation, the RCS was borated to a level of about 5050 parts per million (ppm). The RCS criticality report showed that a boron level of 4350 ppm ensured subcriticality at any core configuration. This minimum concentration would be maintained during and following plenum assembly removal. Procedures governing the frequency of boron sampling, RCS level monitoring, isolation barrier checking, and boron dilution source checking, would be in force during the various plant operations (e.g., termination of internals indexing fixture water processing, fixture platform removal, and canal fill) to ensure against a boron dilution event that would cause the boron concentration in the RCS to drop below 4350 ppm.

- ***NRC Review: Criticality***. [164] The NRC's safety evaluation stated that the potential for criticality of the TMI-2 core due to a reconfiguration of damaged fuel was effectively precluded by maintaining a sufficiently high boron concentration in the reactor coolant. In the NRC's safety evaluation report [165] for reactor vessel head removal, the agency noted that boron concentration in the reactor coolant to 5000 ppm was increased to maintain the core subcritical for all credible core reconfigurations. The licensee performed analyses showing that, at a reactor coolant concentration of 4350 ppm, the core would remain subcritical in any postulated configuration. The NRC's safety criticality report [166] for the RCS concurred with this evaluation. The boron concentration in the RCS at the time of the evaluation was about 5050 ppm, and the licensee would administratively maintain the boron concentration at this level during and following plenum assembly removal.

- ***NRC Review: Boron Dilution***. [167] The NRC's safety evaluation stated that during plenum lift and transfer, as in earlier cleanup activities, procedures would be in place to prevent, detect, and respond to a potential boron dilution event. These procedures included periodic boron sampling and maintenance of all makeup sources at the RCS boron concentration (5000 ppm), periodic checking of double isolation barriers, and level monitoring of the RCS and potential dilution sources. In the NRC's previous safety evaluations for head lift, RCS criticality, and plenum removal preparatory activities, the agency concluded that these measures were adequate to minimize the potential for a boron dilution event. In the extremely unlikely event that boron dilution did occur, the large margin provided by the high RCS boron concentration would allow sufficient time to detect and mitigate dilution. Therefore, the NRC concluded that there was minimal potential that the reconfiguration of the fuel or a boron dilution event during plenum assembly lift and transfer could cause criticality.

3.4.8 Makeup and Purification Demineralizer Resin Sampling

- ***Purpose***. To obtain resin samples from the two makeup and purification demineralizers. Resin samples were required to characterize the present resin conditions for the development of a technically sound resin removal and disposal program.

- ***Evaluation: Criticality (Background)***. An annual report [168] from the DOE provided a detailed summary of the nondestructive assay of the fuel contents inside the demineralizers. The characterizations of the contents inside the demineralizer cubicles and demineralizer vessels were based on demineralizer drawings, accident operating history, and analyses of the condition of the demineralizers. Videotapes, radiation surveys, and contamination swipes were also performed in each cubicle. The DOE coordinated technical assistance provided by the DOE's Hanford Operations and Pacific Northwest Laboratories.

Makeup and purification system demineralizers were in use for about 18 hours during the accident. This system processed reactor coolant system (RCS) water and returned the coolant to various locations within the RCS and storage tanks. The licensee estimated that about 175 cubic meters of highly contaminated reactor coolant passed through the organic resin beds. The fission products and fuel debris resulted in high radiation levels in the demineralizer cubicles and prompted concern over the degree of subcriticality in the vessels. In 1982, robotic inspections of the cubicles and fuel assessments that used three independent techniques eliminated concerns over criticality, which permitted further characterization in 1983. Several national laboratories assisted in characterizing the fuel debris in the demineralizers.

A DOE laboratory report (169) provided an overview of fuel assessment techniques.

- *Gamma Ray Spectrometry*. Data obtained with the silicon lithium (Si(Li)) detector on the amount of cerium-144/praseodymium (Ce-144/Pr) present in the resin were used to calculate a fuel content for the "A" demineralizer vessel. Based on the similar chemical behavior of Ce-144/Pr and uranium dioxide fuel, and also based on the Ce-144/Pr-to-fuel ratios that were determined from samples taken at different locations in the TMI-2 plant, scientists determined a fuel content of 1.3 kilograms ± 0.6 kilograms in the "A" vessel. This estimate was significantly less than the criticality level of 70 kilograms. Although no quantitative fuel estimate could be made for the "B" vessel on the basis of the Si(Li) data, the one data point that was obtained indicated there was less fuel but more fission products in the "B" vessel than in the "A" vessel.

- *Neutron Dosimetry*. In the second characterization study, solid state track recorders (SSTRs) were placed in the "A" cubicle for 29 days to determine the presence of any fuel in the vessels. The SSTRs worked by providing a record for tracks of fission products generated from neutron-initiated fissions in the uranium-235 contained in the SSTR. The neutrons that initiated such fissions originated from spontaneous fissions from fuel debris in the demineralizers. The fission tracks in the SSTR were proportional to the calculated neutron flux. The measurement indicated the presence of an estimated 1.7 kilograms ± 0.6 kilograms of fuel in the "A" demineralizer.

- *Beryllium Detector*. The third characterization technique was based on the reaction that occurs when beryllium interacts with a gamma ray and releases a neutron. Using this method, fuel content in the "A" vessel was estimated at 11 kilograms ± 6 kilograms; the estimate for the "B" vessel was 3.9 kilograms ± 1.5 kilograms. Both estimates were well below the assigned critically safe fuel mass limits for resins in the demineralizers.

- ***Evaluation: Criticality***. (170) The licensee's safety evaluation concluded that there was insufficient fuel content in the resins to pose a criticality concern. Results from three independent measurements of the fuel content in the makeup and purification demineralizer vessels supported the evaluation. The DOE's Hanford Operations and Los Alamos Scientific Laboratory used nondestructive assay techniques to make the three independent measurements of the fuel content in the demineralizers. The DOE's Hanford Operations employed two different assay techniques, including gamma ray spectrometry that used a shielded Si(Li) Compton recoil gamma ray spectrometer and neutron dosimetry that used SSTRs.

- *Assay Techniques*. The first assay technique by the DOE's Hanford Operations involved the detection of the 2.18-megaelectronvolt gamma rays emitted in the decay of Pr-144 and its radioactive parent Ce-144. The quantity of fuel was then deduced from the ratio of Ce-144 to uranium in the fuel with the assumption of no partitioning. The second technique involved a more direct measurement of the fuel determined by the neutron flux arising from spontaneous fission. The Los Alamos group employed a third technique to determine the Ce-144 content. A beryllium (gamma-neutron reaction) detector, which was sensitive only to gamma rays above 1.67 megaelectronvolts, was used to estimate the quantity of Ce-144 in the demineralizers and thereby the fuel content. However, this technique was less sensitive to fuel and less accurate.

- *Conclusion*. These results indicated that no criticality concerns were evident during sampling activities. The measurements of the fuel content were well below the estimated 70 kilograms of fuel required for criticality.

- ***NRC Review***. Editor's Note: An NRC response to the licensee's safety evaluation was not located. However, refer to the NRC's safety evaluation report for makeup and purification demineralizer cesium elution.

3.4.9 Makeup and Purification Demineralizer Cesium Elution

- ***Purpose***. To remove most of the radioactivity from the resins while they were in the demineralizers to the extent that standard resin sluice procedures could complete the task. The scope of this evaluation included only the first phase of a three-phase process for disposition of the makeup and purification of resins. This first phase included the rinse and elution of the demineralizer resins. The latter two phases would include the sluicing, removal, solidification or other packaging, and disposal of these resins. Separate safety evaluations would address the latter phases.

- ***Evaluation: Criticality***. (171) The licensee's safety evaluation concluded that criticality was not a safety concern. The DOE's Hanford Operations reported (172) that there was a total of less than 4 kilograms of fuel in the demineralizers. The most reactive configuration (spherical) of TMI-2 type fuel (3 weight percent uranium-235) in an optimal moderator/reflector array would have required at least 93 kilograms of fuel for criticality to be possible.

- ***NRC Review: Criticality***. (173) The NRC's safety evaluation stated that nondestructive assay studies of the makeup and purification demineralizers showed less than 4 kilograms of fuel debris in the vessel. The minimum critical mass for the expected range of fuel enrichment was greater than 70 kilograms; therefore, the NRC concluded that no criticality potential existed.

3.5 Defueling Tools and Systems

3.5.1 Internals Indexing Fixture Water Processing System

- ***Purpose***. To provide reactor coolant water processing capability during head removal until plenum removal. The internals indexing fixture (IIF) processing system was selected based on the limited reactor coolant processing capability in the drained-down condition and the desire to provide adequate water cleanup capacity to minimize radiation dose rates around the IIF.

- ***Evaluation: Boron Dilution***. (174) The licensee's safety evaluation concluded that adequate measures to prevent a reactor coolant system (RCS) boron dilution event would be taken, adequate detection capability existed in the unlikely event of a boron dilution, and subcriticality of the core would be maintained.

  - *Introduction*. The TMI-2 core was maintained in a subcritical condition by virtue of the high concentrations of soluble boron in the reactor coolant.

    – *Previous Boron Concentration*. Conservative analyses, which were performed previously for the Quick Look video inspection of the reactor core (175) and the reactor vessel head removal, (176) showed that a criticality of the core would be prevented by maintaining a boron concentration of 3500 parts per million (ppm). These analyses bounded fuel configurations associated with planned work activities and credible accidents during head removal. These analyses were also expected to bound any possible configuration for the IIF processing system operation. Therefore, the evaluation of the worst credible boron dilution event determined that 3500 ppm provided an adequate poisoning for criticality prevention in the system across all activities during IIF processing system operation. Reevaluation of boron dilution concerns during operation of the IIF processing system would be required if the lower boron concentration safety limit of 3500 ppm was revised.

    – *Revised Boron Concentration*. To prepare for future defueling activities, the boron concentration was increased to an operating level of 5050 ppm ± 100 ppm. This concentration provided an even greater margin of safety during IIF processing system operation, and the reactor coolant temperature and chemistry would also be maintained within recovery technical specification limits. The only credible way that the RCS boron concentration could have been changed in an uncontrolled manner during IIF processing system operation was by the dilution of the reactor coolant with water that was either unborated or borated below the operating level.

    – *Preventive Measures*. To provide adequate assurance against a return to criticality, the preventive measures included: (●) prevention of boron concentration reduction; (●) detection of boron concentration reduction; and (●) restoration of the reactor coolant to the operating boron concentration.

    The RCS boron concentration could be reduced if water containing less than the operating boron limit was added to the RCS. The potential sources of this water were the various systems connected to the RCS, including the secondary system. Systems that potentially contained underborated water were reviewed and isolated. Two isolation boundaries were provided for potential in-leakage paths. An isolation boundary was defined as a closed tagged-out valve, a removed spool piece, a beat exchanger tube boundary, or an electrically locked-out pump. An electrically locked-out pump could be considered an isolation mechanism whenever the pump represented a pressure driving head. Where gravitational flow through a pump body had a potential for adding underborated water, a minimum of two additional isolation boundaries were provided.

The licensee's safety evaluation report (SER) or reactor vessel head removal procedures described the specific actions to prevent the addition of underborated water to the RCS from the various systems.

Operation of the IIF processing system presented additional potential deboration pathways and concerns. The IIF operation modes addressed in this SER included: (●) automatic level control; (●) manual level control; and (●) shutdown operations.

- *Automatic Level Control Mode*. When the IIF processing system was operating in the automatic level control mode, the inadvertent addition of underborated water to the RCS was considered extremely unlikely, because of the preventive measures that would be taken. Potential flowpaths of underborated water were double isolated. The position of isolation valves was confirmed visually or administratively every 24 hours, and the levels of tanks containing underborated water were logged every 24 hours. However, to further protect against a reduction of boron concentration, several methods of detecting a deboration event were used: (●) RCS level monitoring; (●) RCS boron sampling; and (●) RCS inventory monitoring.

  – *RCS Level Monitoring*. During this mode of IIF processing system operation, the RCS level was automatically maintained by valve HU-V9, which was controlled by the bubbler system (reactor coolant level monitoring system). The level was maintained at a given level, and alarm points were set above and below the desired level to prevent an unacceptable increase or decrease in the water level. If the RCS level reached an alarm point, makeup and letdown were both terminated, and alarms would sound in the control room, at the IIF, and on the submerged demineralizer system panel within the fuel handling building.

    Maximum letdown from the IIF through the submerged demineralizer system was calculated to be 15 gallons per minute (gpm), and normal flow rates were expected to be 10–12 gpm. Leakage into the RCS at flow rates greater than 15 gpm would increase the IIF level until the alarm point was reached. At this time, the IIF processing system would automatically shut down, and the source of in-leakage would be identified and terminated. The RCS level monitoring system could not detect in-leakage less than 15 gpm. Normal plant RCS level monitoring was available and was also equipped with alarms in the main control room to signal unacceptable water levels. The control room operators regularly monitored and logged the RCS level.

- *RCS Boron Sampling*. RCS sampling was performed once per week while the RCS was in the normal drained-down, depressurized condition, known as the "level control mode." While the plant was in this mode, in-leakage into the RCS would increase the RCS level, which would be indicated in the control room. However, during the operation of the IIF processing system in the automatic control mode, the RCS level was automatically maintained without operator action. Since any in-leakage into the RCS from potentially underborated sources could add to the makeup flow, the bubbler system could sufficiently throttle KU-V9 to permit the continued dilution without an observable increase in RCS level. Thus, in-leakage into the RCS from potentially underborated sources could be difficult for control room operators to identify quickly and would not depend on RCS level monitoring alone.

  Increasing the frequency of RCS sampling could provide adequate detection capability. The frequency of sampling was based on the time required for the boron concentration to be reduced from the operational level to the minimum level that would prevent criticality for postulated core configurations during head removal. These configurations would also bound activities for plenum removal.

  To define a sampling frequency based on normal operation of the IIF processing system, the dilution flow rate needed to be considered (i.e., the sampling frequency would be increased if there was a large dilution flow rate). Also, the maximum letdown rate through the submerged demineralizer system (SDS) dictated the maximum dilution flow rate since greater dilution flow rates would cause an RCS level increase. Control room operators could identify RCS level increases before significant dilution could take place.

  The calculation of the dilution time intervals for a range of SDS flow rates was discussed in detail in the SER (refer to Section 3.3.1.b).

- *RCS Inventory Monitoring*. To monitor the RCS inventory, the following calculations were performed at given time intervals:

  - *RCS Leak Rate Check*. In the level control mode, procedures required that an RCS leak rate check would be performed every 24 hours for a period of 4 hours. During the 4-hour period, all makeup and letdown to the RCS would be secured. Whereas, during the reactor coolant processing, leak rate checks would be performed every 72 hours for a period of 2 hours. During IIF processing system operation, the RCS leak rate monitoring would also be performed every 72 hours; however, the monitoring period could be increased to 4 hours to provide an adequate interval for leak rate measurement in the unpressurized condition.

  - *Boron Mass Balance*. Calculations for boron mass balance would be performed once every 24 hours during normal processing with the RCS pressurized or in the level control mode. These calculations were used to ensure that adequate boron concentration existed, even if any inventory discrepancy was unborated water. The frequency of the boron mass balance calculation would remain the same during operation of the IIF processing system.

    - *Tank Level Checks*. Tank level checks assessed the bleed tanks being used for makeup and receipt of reactor coolant to detect any discrepancy in RCS inventory. Reducing the reactor coolant boron concentration from 4950 to 3500 ppm required more than 10,000 gallons of unborated water to be added. Discrepancies greater than 10,000 gallons required a temporary termination of processing until the discrepancy was investigated. During operation of the IIF processing system, RCS inventory calculations would be performed at the same frequency as RCS boron sampling. These actions would detect a boron dilution event.

  – *Conclusion*. The evaluation concluded that a criticality due to a deboration event during operation of the IIF processing system was precluded based on the following: (●) Double isolation of potential pathways adequately prevented boron dilution. (●) Adequate detection of a boron dilution event was possible because of increased RCS sample frequency, in conjunction with level monitoring and RCS inventory checks. (●) In the unlikely event that boron dilution occurred, procedures were in place to terminate the dilution transfer. These procedures would include the mechanisms to return the RCS to the operating boron concentration.

- *Manual Level Control Mode*. The potential for boron dilution during operation of the IIF processing system did not differ significantly between the manual level control mode and automatic level control mode. The sampling and inventory calculation frequency during manual operation was specified to be the same as for the automatic level control operation.

- *Shutdown Operations*. The installation of the IIF processing system introduced flowpaths for underborated water to the RCS from the containment building basement pump system through the fuel transfer canal drain manifold to the discharge line of the IIF processing pump to the IIF and also from the fuel transfer canal drain pump through the fuel transfer canal drain manifold and discharge line of the IIF processing pump to the IIF.

  To provide double isolation of underborated water sources, quick-disconnect hoses and branch valves were provided at the fuel transfer canal drain manifold. When the IIF processing system was operating, the two other branch hoses would be disconnected, and the branch valves would be closed to provide double isolation from these sources. In addition, a check valve in the IIF processing system branch line at the manifold would provide added assurance that these two pathways were not credible sources for boron dilution.

  During the shutdown condition, the configuration of the RCS was not significantly different than in the level control mode before head removals, so the requirements for RCS boron sampling and inventory monitoring remain unchanged. Therefore, the RCS would be sampled once per week by a dedicated remote sampling system. This system would provide a means of obtaining a sample from outside the containment building if entry into the containment building was precluded or deemed undesirable. In addition, the RCS leak rate determination would be performed once every 24 hours.

- ***NRC Review: Boron Dilution***. [177] The NRC's safety evaluation concluded that additional system modifications were needed to detect potential boron dilution events. The NRC discussed its recommendations for system modifications in meetings with the licensee, and systems were modified to improve water inventory measurement and core sampling capability based on these discussions. The NRC also concurred with the licensee's system modifications and proposed use of the IIF water processing system detailed in the licensee's subsequent safety evaluation, [178] which proposed the use of a new RCS sampling system and improved water level indication in the reactor coolant bleed tanks.

o *RCS Boron Concentration*. Boron dilution of the RCS could result in an inadvertent criticality, so safeguards were established to prevent the occurrence of a dilution event. In the unlikely event that dilution occurred, the boron concentration in the RCS would be increased to 5050 ppm ± 100 ppm to provide a safety margin that addressed all criticality concerns. Additionally, sampling techniques and water inventory accounting procedures were established to provide early warning of a dilution event.

Dilution safeguards included procedural controls over all activities that involved the RCS. In addition, a physical, two-boundary isolation of the RCS from all potential dilution sources was provided. Conservative analyses indicated that criticality in the core would not occur if boron concentrations were maintained at 3500 ppm or higher. The evaluation concluded that a dilution event would be detected well before the boron concentration could decrease from the existing level, 5050 ppm ± 100 ppm, to 3500 ppm.

o *RCS Boron Sampling*. Boron sampling was one of two methods available to detect dilution. The licensee's initial SER proposed to periodically sample the IIF processing pump discharge to monitor for boron dilution. The SER addendum revised this concept to a more sensitive sampling technique. This technique included a separate sampling pump to obtain the sample from the area immediately above the core rubble bed. A conservative sampling frequency (once per 6 hours) was established to ensure early detection of a dilution event. The sampling frequency could be modified as needed, after core mixing data were collected and analyzed during the IIF processing. An online boron monitoring system was also being developed, and if successful, the system would provide a basis for reducing the sampling requirements.

o *Mass Balance*. A periodic mass balance to determine primary system water inventory could also serve as a technique to detect a dilution inflow. The licensee concluded that 12,480 gallons of unborated water would be required to dilute the 36,000-gallon reactor vessel/IIF volume from 4950 ppm boron to 3500 ppm boron. The NRC performed a more conservative analysis by assuming that the inflow of unborated water was restricted to the area outside the core barrel and to the lower hemisphere of the reactor vessel. This analysis was based on the concern that a critical fuel mass could have accumulated on the bottom hemisphere of the reactor vessel, and a dilution inflow could interact with this critical mass after limited mixing in the vessel. Structural baffles inside the reactor vessel and the core debris physically segregated the incoming makeup flow from the main body of water in the vessel. This physical boundary formed a 14,000-gallon water volume outside the core region. The NRC's analysis indicated that more than 4500 gallons of unborated water would be required to reduce the boron concentration in this region from 4950 to 3500 ppm.

- *Inventory Detection*. Based on the more conservative dilution scenario, the evaluation concluded that the IIF processing system water inventory detection system should be designed with sufficient sensitivity to detect a 4500-gallon inflow of potentially unborated water. To achieve a total system sensitivity of 4500 gallons, each of the two reactor coolant bleed tanks (RCBTs) should include instrumentation capable of detecting water inventory to at least ±750 gallons. RCBT sensitivity to ±750 gallons would be required to include mass balance calculations as a viable technique for detecting dilution events. Inventory calculations would need to be performed hourly to provide effective dilution detection.

- *Conclusion*. Based on the analysis described above, the RCBT inventory measuring capability was modified to increase the sensitivity to at least ±750 gallons. Operating procedures required hourly water inventory calculations. The boron dilution controls described above were incorporated into the IIF processing system and provided defense-in-depth assurance that minimized the potential for inadvertent criticality. The onsite NRC inspector would monitor IIF processing operations and would confirm the bases for this evaluation as operating data became available.

3.5.2 Defueling Water Cleanup

3.5.2.1 Defueling Water Cleanup System

- ***Purpose***. To remove organic carbon, radioactive ions, and particulate matter from the reactor vessel and fuel transfer canal/spent fuel pool "A" (FTC/SFP-A). The defueling water cleanup system (DWCS) actually comprised two independent subsystems (sometimes called trains): the reactor vessel cleanup system and the FTC/SFP-A cleanup system. The DWCS was totally contained within areas that had controlled ventilation and could be isolated. This limited the environmental impact of the system during normal operations, shutdown, or postulated accident conditions.

- ***Evaluation: Criticality***. [179] The licensee's safety evaluation stated that subcriticality of the core was maintained by a high concentration of boron in the reactor coolant system (RCS). The design ensured subcriticality of the fuel within the filter canister, which was addressed in the licensee's technical evaluation report [180] for the defueling canisters. The design of the FTC/SFP-A pumps and the reactor vessel cleanup pumps did not allow a significant quantity of fuel to accumulate. The system piping and the post filters were also designed to prevent a possible critical configuration of fuel debris and sized to be critically safe. [181] This was accomplished by restricting the size and configuration of components. The post filters were located downstream of both filter trains in the line to the ion exchangers and dilution water supply. So, the post filter would not accumulate significant quantities of fuel unless the filter media in the filter canister ruptured. If such a rupture occurred, the post filter would trap any fuel fines that would be transported past the filter canisters. Although the DWCS included an operations mode that would bypass the filter canisters, large amounts of fuel were not expected to accumulate on the post filters because of the restrictions on operating in this mode. Other system components were designed to preclude fuel accumulation, or the components would have critically safe dimensions.

- ***Evaluation: Boron Dilution***. [182] The licensee's safety evaluation stated that the only credible means of attaining criticality from the fuel contained in the vessel was through deboration of the RCS water. The approach described for prevention of deboration in the licensee's safety evaluation reports for the vessel head removal [183] and operation of the internal indexing fixture processing system [184] would be followed during DWCS operation. Specific system evaluations of deboration control would be performed before DWCS operation. The licensee's hazard analysis report [185] on the potential for boron dilution of the RCS had previously addressed boron dilution during defueling.

Other previously evaluated dilution events would be prevented by red-tagged isolation barriers if the DWCS was shut down for any reason. The X-Y bridge flush jets would be administratively controlled to ensure adequate mixing of water sources before returning the water to the lower vessel region (about the 301-foot elevation). The licensee's safety evaluation report [186] for the vessel head removal had previously analyzed mixing adequacy.

The Y trolley flush system had the capability to obtain its water supply from either the DWCS or the reactor vessel directly. When using the reactor vessel directly, the Y trolley flush system would be connected to a dedicated system with no external interfaces. When connected to the DWCS, the Y trolley would use the same supply and administrative controls as the X-Y bridge flush jets.

The cavitating water jet (called "cavijet") could be used at any elevation inside the reactor vessel. The cavijet would be isolated from the introduction of unborated water by establishing triple barrier isolation when the DWCS was shut down.

- ***NRC Review: Criticality***. [187] The NRC's safety evaluation concluded that the design of the DWCS provided reasonable assurance of preventing criticality during operation by maintaining a high concentration of boron in the system. The minimum boron concentration needed to ensure subcriticality was evaluated in the licensee's criticality evaluation report [188] for the RCS. The NRC review determined that the maintenance of the RCS chemistry within the constraints specified in the RCS criticality report would be adequate to ensure subcriticality in the reactor core and piping systems circulating RCS water.

The DWCS filters would remove fuel debris from the RCS. Following their dewatering, the filters would no longer be flooded with borated water. The filter design was such that the filters could contain up to 1000 pounds of core debris. Subcriticality of the loaded filters was ensured with installed boron poison rods. In addition, the design would maintain the debris in a geometrically safe configuration during all conditions for expected operations. The detailed canister criticality analysis was evaluated as part of the NRC's review of the licensee's technical evaluation report for defueling canisters.

The post filters, which were intended to prevent carryover of particulate matter to the ion exchangers, could also accumulate fuel material. Criticality was prevented in these filters by their physical design, in which the filters would reach the operational limits on differential pressure when about 4 pounds of debris had accumulated. This was well below the most conservatively predicted minimum of 70 kilograms of fuel needed for a critical mass. In addition, the post filters were designed to contain the fuel debris in a critically safe geometry.

3.5.2.2 Cross-Connect to Reactor Vessel Cleanup System

- ***Purpose***. To modify the fuel transfer canal/spent fuel pool "A" (FTC/SFP-A) cleanup system portion of the defueling water cleanup system (DWCS) to allow processing of the FTC/SFP-A water through the "B" train of the DWCS reactor vessel cleanup system. The purpose of this modification was to provide the capability to effectively process the FTC/SFP-A water in a similar manner to the reactor vessel cleanup process without the installation of additional body-feed and coagulant equipment in the fuel handling building. In addition, the proposed modification would authorize the use of FTC/SFP-A filtered effluent as a water source for the body-feed tank and as dilution water for the coagulant addition unit.

- ***Evaluation: Criticality***. [(189)] The licensee's safety evaluation considered the effect of a ruptured filter media canister on criticality, as well as the effect of adding body-feed and coagulants.

  - *Filter Media Rupture and Criticality Prevention*. In the proposed cross-connect modification configuration, the post filter was not normally used, and the reactor vessel filter canisters would be capable of processing fuel transfer canal (FTC) and reactor vessel water. The evaluation considered filter media rupture when the FTC water was being processed and returned to the spent fuel pool A (SFP-A) through a disconnected canister inlet in the reactor vessel filter canister. The potential existed to transfer reactor vessel fuel fines from the ruptured media of the reactor vessel filter canister directly to SFP-A. This canister also processed reactor coolant system (RCS) water since the post filter was not in service in this configuration. However, this condition would not create a safety concern since the boron concentration in the FTC and SFP-A was maintained between 4350 and 6000 ppm. This boron concentration ensured that any fuel fines would remain subcritical under all credible conditions.

    If the decision were made to process FTC/SFP-A water through the ion exchanges, the appropriate flowpath would be established, and the post filter would be placed in service.

Failure of filter canister media would introduce fuel fines into the system, but the fines would be collected by the criticality-safe post filter. This filter was criticality safe by design; therefore, the ion exchange would be protected.

- *Effect on Body-Feed and Coagulant Addition*. In the proposed configuration, a filter media rupture during filling of the body-feed tank, or while providing dilution water to the coagulant addition unit, could transfer fuel fines to either or both systems (reactor vessel cleanup system and the FTC/SFP-A cleanup system) since the post filter would be bypassed. This scenario did not create a safety concern since the body-feed tank and coagulant addition unit would contain RCS-grade borated water (4350 to 6000 ppm). This ensured that the fuel would remain subcritical under all credible conditions. The introduction of coagulant in the coagulant addition unit mixing chamber would dilute the borated water. The mixing chamber diameter of 5.5 inches ensured subcriticality. This conclusion was based on the criticality safety limit of 9.6 inches in diameter that assumed an infinite cylinder with optimal fuel-to-moderation ratio and a water reflector. In addition, the active polymer of the coagulant was melamine-formaldehyde, which was previously evaluated in the licensee's safety evaluation report [190] for the use of different coagulants and the associated criticality evaluation report. [191] These reports demonstrated that the use of this active polymer had no adverse effect on RCS chemistry and criticality control.

- ***Evaluation: Boron Dilution***. [192] The FTC/SFP-A cleanup system and the reactor vessel cleanup system were normally separated by double-valve isolation. However, double-valve isolation between these systems would not always be possible. For example, only single-valve isolation would exist when the DWCS was concurrently processing FTC/SFP-A water in the "B" train and reactor vessel water in the "A" train. In the proposed modified configuration, DWC-V384 and DWC-V385 would provide single-valve isolation. [t] The licensee's safety evaluation concluded that single-valve isolation was acceptable based on the following:

- *Water Source Isolation*. The safety evaluation report stated that boron dilution of the RCS in the proposed modified configuration would require a double failure of a nonborated water source or a water source borated to less than RCS grade and injection into the FTC/SFP-A cleanup system concurrent with leakage of system isolation valves. [u] Procedures to operate the proposed system would contain prestartup checklists to ensure that underborated water sources connected to the FTC/SFP-A process piping were isolated. An increase in FTC/SFP-A or internal indexing fixture water level would be detected, and the systems would be shut down if a failure occurred.

- *Limited Operation*. FTC/SFP-A processing was not intended to be a continuous process. Only the reactor vessel cleanup system would be in operation most of the time, and the probability of a double system failure was small. One processing cycle was established to

[t] Editor's Note: The safety evaluation did not provide a diagram of the new configuration of the cleanup systems.

[u] Editor's Note: This statement from the licensee's safety evaluation report was not clear. It is believed that two concurrent failures must occur in order to inject underborated water sources from the FTC/SFP-A cleanup system (train B) into the reactor vessel cleanup system (train A). The two failures include (1) underborated water sources inadvertently connected to the FTC/SFP-A process piping and (2) leakage of system isolation valves.

be 1 million gallons of water processed per quarter; at 50 gallons per minute, this cycle could be completed in about 16 days.

- *Boron Concentration*. The NRC's safety evaluation [193] attached to the technical specification amendment that increased the minimum required boron concentration recognized that the required boron concentration in the RCS, FTC, and SFP-A would essentially eliminate the possibility of boron dilution due to a leak or valve misalignment. Even when such a leak or valve misalignment occurred, boron dilution of the RCS was considered not credible since the technical specification required boron concentration of the FTC/SFP-A (i.e., 4350 to 6000 ppm) to be the same as that for the RCS. Therefore, the evaluation concluded that criticality due to a boron dilution event was not a concern.

- ***NRC Review: Criticality/Boron Dilution***. [194] The NRC's safety evaluation noted that the proposed modifications deviated from the flowpaths described in the original DWCS technical evaluation report. However, the NRC concluded that the likelihood of accidents and their consequences, with regard to the impact of system leakage, boron dilution potential, and potential for inadvertent criticality, were within the bounds of the analysis in the original approval.

3.5.2.3 Temporary Reactor Vessel Filtration System

- ***Purpose***. To restore and maintain the visibility in the reactor vessel to acceptable levels to ensure the continuation of the early defueling operations. Operation of the defueling water cleanup system (DWCS) revealed that a differential pressure across its filter canisters would increase rapidly as the result of micro-organism growth in the reactor coolant. Consequently, the DWCS was able to process only a relatively small amount of reactor coolant before the maximum design pressure was reached and the filter canister had to be replaced. These developments created the need to design and operate a temporary filter system while a permanent program to control this phenomenon was being developed.

- ***Evaluation: Criticality***. [195] The licensee's safety evaluation stated that any fluid system connected to the vessel that transported reactor coolant system (RCS) water had the potential to move fuel bearing materials. Consequently, the potential to accumulate fuel outside of the reactor vessel was addressed. The licensee stated that rigorous design controls were not required for the temporary reactor vessel filtration system (TRVFS). This conclusion was based on the temporary nature of the system and the unlikelihood of accumulating significant quantities of fuel, given the suction point for this system. However, the TRVFS design and operation provided separate assurances to preclude significant fuel accumulation and criticality because only suspended material in the RCS would be moved and any material trapped by the filter would always be in contact with borated water. The licensee's evaluation considered the following:

- *Suction Depths*. Various suction depths within the internals indexing fixture and the reactor vessel could be used with the TRVFS. A shallow suction depth at the 325.5-foot elevation (2 feet below the fixture water level) and deeper suction depths no lower than at the 313.5-foot elevation (1 foot above the normal core region) were planned. At these suction elevations, the TRVFS would pick up only suspended material in the RCS, which would minimize the quantity of fuel trapped on the filter.

- *Entrapped Fuel*. As a bounding evaluation, the quantity of uranium dioxide (UO_2) that could be trapped on the filter was determined based on the following assumptions: (●) flow rate of 100 gallons per minute; (●) gross alpha concentration of 10^{-2} microcurie per cubic centimeter (based on occurred concentrations during core drilling); (●) 150 microcuries of gross alpha per gram of UO_2; (●) 12 hours of continuous TRVFS operation; and (●) all fuel picked up by TRVFS being trapped on the filter.

  Based on these assumptions, about 18 kilograms of UO_2 could be deposited in the filter. This quantity was much less than the minimum critical mass for TMI-2 fuel (i.e., 93 kilograms).

- *Boron Concentration*. The licensee's criticality analysis report (196) for the RCS previously established that the core material could not go critical under any configuration postulated for defueling, provided that the surrounding water contained at least 4350 parts per million (ppm) boron. Since the TRVFS would be drawing water from the reactor vessel in a closed loop during normal operations, boron concentrations would be maintained at or above 4350 ppm. Therefore, any fuel material deposited in the filter would be effectively poisoned by the boron content of the water. Backflushing operations would be performed using a water supply from borated water storage tank water (also borated to at least 4350 ppm). This water supply would be routed to the TRVFS through existing flush wand connections.

- *Fuel Particle Size*. In addition, the licensee determined that, at a distance of 1 foot below the suction elevation, the flow velocity toward the suction was less than the settling velocity of fuel particles with sizes greater than 300 microns. Since the deepest suction elevation for TRVFS was greater than 8 feet above the core debris bed and at least 1 foot above any known area within the reactor vessel, fuel particles larger than 300 microns were not expected to be picked up by the TRVFS. As a further precaution, the licensee would not perform aggressive defueling techniques (e.g., shredder operation, clamshell debris removal) during TRVFS operation in order to limit the size of fuel particles that could be suspended. Therefore, filter canisters could be used with TRVFS operation without impacting the criticality evaluations. These evaluations were documented in the safety analysis report (197) of the Model 125-B shipping cask, which limited fuel particle sizes in the filter canisters to less than or equal to 850 microns.

- ***Evaluation: Boron Dilution***. (198) The licensee's safety evaluation considered the potential for criticality due to a boron dilution event. Diatomaceous earth consisted of about 88-percent silica and exhibited no propensity to remove or absorb boron. Operating experience with these filters in the fuel pool and reactor vessel resulted in no detectable dilution of either body of water. Therefore, significant boron dilution caused by removal of boron by the diatomaceous earth filters was not considered credible. Boron dilution of the reactor vessel or the filter vessel during normal operation was judged not credible because of the closed loop nature of the system, the small number of system interconnections, and the unavailability of unborated water sources in the vicinity of the suction connections. Administrative controls would ensure that there was not a significant probability of diluting the TRVFS or reactor vessel during backflush operations.

- ***NRC Review: Criticality (Revision 0)***. (199) The NRC's safety evaluation stated that to approach criticality, a minimum of 70 kilograms of UO_2 had to accumulate either at the filter assembly or at the waste drum. The NRC concluded that not even a small fraction of this amount would accumulate for the following reasons:

  - *Radiation Monitoring*. The filter assembly surface would be continuously monitored for gamma radiation. Filter changeout would take place before the radiation level reached a conservative setpoint (e.g., 3 roentgens per hour). For the amount of UO_2 to reach 70 kilograms, the corresponding radiation level would have reached 15,000 roentgens per hour.

  - *Suspended Solids Only*. The suction of reactor coolant was taken through a hose in the internal indexing fixture at no more than 2 feet below the normal reactor coolant level, which was more than 10 feet above the top of the core debris bed. At the maximum flow rate of 75 gallons per minute through the 1.5-inch inside diameter suction hose, there would be no velocity effects to pick up any significant amount of fuel debris from the debris bed. Only materials suspended in the reactor coolant would be removed. No more than about 0.2 kilogram of UO_2 was estimated to accumulate in the filter media after 12 hours of continuous operation. This estimate assumed that the reactor coolant contained 1 ppm of UO_2 and also assumed this UO_2 concentration was conservative, since analysis of the DWCS fluid had shown no detectable fissile material.

  - *Gross Alpha Analysis*. Sample analysis for gross alpha radioactivity would be performed on the waste drums. Although the amount of UO_2 in each spent filter media batch was expected to be no more than a few grams, analysis of the grab sample from the waste drums would further ensure that no significant quantities of UO_2 had accumulated in each waste drum.

- ***NRC Review: Criticality (Revision 1)***. (200) This revision reflected a change based on a more realistic and conservative approach to the calculations of the dose rates of fuel content in the TRVFS. The resulting dose rate at the filter housing, with the revised assumptions, was about 340 roentgens per hour. Assuming a radiation alarm setpoint of 3 roentgens per hour on the filter housing monitor, the result was a safety margin of about 110. A possible increase of the radiation alarm setpoint was proposed as an ALARA measure since, at that time, the filter was being replaced based on radiation levels as opposed to differential pressure. The frequent filter replacements were adding significantly to the exposure from defueling operations. The changes in the proposal would limit personnel doses based on operating experiences. [201]

The NRC's safety evaluation considered the proposal and approved continued operations of the system with certain restrictions and administrative controls. For example, only sources of water with boron concentrations greater than or equal to 4350 ppm were allowed to be processed through or stored in the TRVFS. Also, waste storage drums were to be sampled and analyzed for fissile material content before solidification or transfer to another container or system.

The NRC requested additional information for further review that could lead to lifting these restrictions. The only sources of water passing through or contained in the TRVFS, including the waste storage drum, would be borated to greater than 4350 ppm. These sources in the RCS and borated water storage tank would be maintained at nominal 5000 ppm boron with the technical specification's minimum of 4350 ppm boron. The licensee's criticality evaluation report [202] for the RCS and the NRC's related safety evaluation report [203] provided bounding criticality safety evaluations of a configuration (approved by the NRC) that was much more reactive than any attainable by the TRVFS. Subcriticality would be ensured when the minimum boron concentration in the sources of water exceeded 4350 ppm.

- ***NRC Review: Criticality (Revision 2)***. [204] This revision reflected changes that included the use of a new larger filter vessel that operated at higher flow rates (that had the potential to accumulate a larger quantity of fuel debris in the filter media) and the use of a defueling knockout canister as a receptacle for the discharged filter media. The safety evaluation reanalyzed the operation of the existing TRVFS for use with a new filter vessel and residue canister. [205]

The NRC's safety evaluation stated that the licensee's criticality evaluation report [206] for the RCS showed that there was no potential for achieving criticality in any credible fuel debris configuration in the presence of water borated to at least 4350 ppm. Since the TRVFS would process only water borated to no less than the recovery technical specification limit for minimum boron concentration, the NRC concluded that there was no potential for criticality in the filter or the attached piping. A previous NRC safety evaluation report [207] for the debris vacuum system demonstrated that the defueling knockout canister was a critically safe storage receptacle for fuel debris, regardless of particle size, when loaded to within its design weight limit.

- ***NRC Review: Criticality (Revision 3)***. This revision reflected changes that included the use of filter canisters and knockout canisters as receptacles for discharged diatomaceous earth, fuel debris, and backwash water, and the allowance of deeper suction within the reactor vessel. [208]

Editor's Note: The NRC's evaluation was not found.

3.5.2.4 Filter-Aid Feed System and Use of Diatomaceous Earth as Feed Material

- ***Purpose***. To add a feed material into the filter canisters to promote the buildup of cake on the filter media, thereby significantly improving the performance of the defueling water cleanup system (DWCS) filter canisters. A filter-aid feed system that used diatomaceous earth as the feed material was installed as an ancillary system to the DWCS.

- ***Evaluation: Criticality***. [(209)] The licensee's safety evaluation concluded that the neutron-moderating ability of diatomaceous earth was significantly less than the moderating ability of water. Therefore, if diatomaceous earth were added to the reactor coolant system (RCS), the effective neutron multiplication would not exceed 0.99, even if diatomaceous earth became intermixed with the fuel. Criticality control within the DWCS filter canisters was achieved by the placement of poison in the defueling canisters. The low moderating ability of diatomaceous earth would not affect the canister criticality evaluations.

- ***Evaluation: Boron Dilution***. [(210)] The licensee's safety evaluation stated that criticality control of the TMI-2 core was achieved by maintaining the boron level in the RCS greater than 4350 parts per million (ppm). Water used in the precoat and body-feed subsystems was taken directly from the charging water storage tank (SPC-T-4), which was borated to at least 4350 ppm. The water supplied from SPC-T-4 was obtained through a tie-in to the existing borated water flush line. Downstream of this tie-in, one branch supplied borated water to the precoat subsystem, and the other branch supplied borated water to the body-feed subsystem. During operations, administrative controls (i.e., approved procedures) ensured the use of SPC-T-4 as the water source. Since the RCS was maintained at a boron concentration of at least 4950 ppm, if unborated water was inadvertently used, then the small injection rate of water into the RCS through the filter-aid subsystems would not significantly reduce the boron concentration in the RCS before detection and correction. Each branch from the borated water flush line tie-in contained isolation valves to maintain double-barrier configuration control and to minimize the probability of a boron dilution event in the RCS when the subsystems were not in operation.

- ***NRC Review: Criticality***. [(211)] The NRC's safety evaluation concluded that the addition of the material would not affect the bounding conditions of the canister criticality analysis that were discussed in the licensee's technical evaluation report for the DWCS. Proper administrative controls would ensure that the body-feed precoat material was prepared with borated water and that operations remained within the bounds of the criticality evaluation detailed in an earlier revision of the technical evaluation report for the DWCS.

3.5.2.5 Use of Coagulants

- ***Purpose***. To demonstrate the use of coagulants and body-feed material to improve the performance of the defueling water cleanup system (DWCS) filter canisters in maintaining water clarity. Operating experience with the DWCS had not achieved the desired level of clarity in the reactor coolant system (RCS) water to support defueling operations within the reactor vessel. The DWCS filters required changeout because of high differential pressure without the expected high filter throughput. The root cause of shortened filter canister life was expected to be the presence of hydrated metallic oxides in colloidal suspension within the RCS that were plugging the filter media. The addition of the coagulant with body-feed was expected to agglomerate the colloids to filterable sizes, thus forming a filter cake on the filter media.

- ***Evaluation: Criticality***. [212] The licensee's safety evaluation noted that criticality safety of the RCS and defueling canisters was maintained by use of soluble boron at 4350 parts per million (ppm) and solid boron carbide, respectively. The use of coagulants and body-feed material in the RCS or canisters would not compromise the function or significantly increase the burden of these mechanisms for ensuring criticality safety. The following discussions were provided in the safety evaluation report (SER) that assessed the physics of the criticality safety issue. In this SER, the coagulant materials that were evaluated were referenced as vendor products 1182 and 1192. The body-feed material that would be used was diatomaceous earth.

  - *RCS Criticality*. Criticality safety in the RCS was maintained by the presence of water borated to a concentration of at least 4350 ppm. This boron concentration was adequate for maintaining the entire TMI-2 fuel inventory subcritical under bounding assumptions in the licensee's criticality SER [213] for the RCS. These bounding assumptions included optimum fuel moderation conditions and worst credible fuel configuration. As stated in the licensee's earlier revisions of the subject SER, [214, 215] coagulants 1182 and 1192 and diatomaceous earth were tested to determine their effect on soluble boron in the reactor coolant. These tests verified that the addition of the noted chemicals to the RCS would not cause precipitation of boron.

    The concentrations of coagulant injected into the RCS would be small. To enter the reactor vessel, the material must first pass through the DWCS filters. The accumulation of a nonborated mass of insoluble coagulant in the RCS that could coagulate and cause boron displacement was not credible. Since the coagulant material would not form a nonborated mass, the boron sampling requirements instituted for the RCS at the time would provide adequate warning if the boron concentration was below the administrative limit of 4950 ppm. Additionally, an excess of 5 tons of coagulant material that was dissolved in the reactor vessel water would be needed to pose a criticality concern. Therefore, given that the boron concentration of the RCS would be maintained above the administrative limit, the addition of these materials would not be a criticality safety concern.

  - *Defueling Canister Criticality*. Criticality safety of the defueling canisters was ensured by using fixed boron carbide material inside each canister. The coagulant and body-feed materials would collect primarily in the filter canisters; however, the following technical assessment was independent of the canister type (i.e., fuel, knockout, or filter). As with the RCS, the poison requirements for the defueling canisters were determined using bounding analytical assumptions. One of the key assumptions was the existence of optimum moderation conditions for the fuel. The moderator was assumed to be unborated water. The acceptability of the coagulant and body-feed material was demonstrated by assessing the impact on neutron multiplication of the canisters with and without the added materials.

– *Evaluation of 1192*. Vendor product 1192 was a cationic coagulant and was only 20 weight percent of the polymer $C_8H_{16}NCl$ and 80 weight percent unborated water. When the polymer was added to water, the chemical reaction caused the chlorine to disassociate from the molecule, which left a cation to coalesce the suspended solids. The evaluation would quantify the net effect of the increased moderation caused by the additional hydrogen and the increased neutron absorption due primarily to nitrogen.

  - *Assumption (Specific Gravity)*. The monomeric molecular weight (amu) of 1192 with the chloride atom was 161.7 and 126.3 without the chloride atom. Given that the polymer was 20 weight percent of the solution and the density of the solution was 1.03 grams per cubic centimeter, the polymer density was found to be 1.18 grams per cubic centimeter. Additionally, the evaluation assumed that the specific gravity of the polymer did not change with the loss of the chloride atom.

  - *Reactivity Effect from Nitrogen*. The number densities of the molecule and component atoms of the dry material were based on a molecular weight of 161.7 amu for the molecule without the chloride atom. Based on data [216] from Brookhaven National Laboratory, the macroscopic absorption cross section of 1192 was determined to be 0.03 per centimeter, while that of unborated water was 0.02 per centimeter, resulting in an excess amount of absorption. Using a macroscopic cross section of boron in water per ppm of 4.27×10^{-5} per centimeter, the excess absorption was correlated with an equivalent boron concentration of 225 ppm. The change (delta-k) in the infinite neutron multiplication ($k_{infinity}$) resulting from the additional equivalent boron was obtained using data from a report [217] by Oak Ridge National Laboratory. Based on these data, the $k_{infinity}$ was found to drop 0.044 delta-k, for a boron concentration value that changed from zero to 225 ppm.

  - *Reactivity Effect from Hydrogen*. The next step in assessing 1192 examined the increase in $k_{infinity}$ due to the increased moderation from the additional hydrogen. This was accomplished by using the “four factor formula” from nuclear reactor theory. Based on the values calculated for the four parameters, the increase in $k_{infinity}$ due to the addition of the 1192 polymer was found to be 0.007.

  - *Result (Net Reactivity Effect)*. The net effect on the $k_{infinity}$ (delta-k) caused by the presence of the 1192 polymer was estimated to be:
    [delta-k = (0.007 - 0.044) = -0.037].

  - *Sensitivity Analysis*. If the assessment provided above for 1192 was repeated using a molecular weight of 126.3 amu (i.e., without chloride) to obtain the elemental number densities, the negative effect on $k_{infinity}$ was estimated to be:
    [delta-k = (0.029 - 0.087) = -0.058].

- *Conclusion*. In both of the cases detailed above, the thermal neutron absorption by chlorine was conservatively excluded from the evaluation. Although the evaluation was made to determine the net change to $k_{infinity}$ of an infinite canister geometry, the conclusions apply to finite canister geometries (i.e., effective neutron multiplication). Therefore, the addition of the coagulant and body-feed materials would not have an adverse effect on criticality safety of the defueling canisters.

– *Evaluation of 1182*. The hydrogen number density for 1182 was determined using the amu of 151.2 and a polymer density of 1.52 grams per cubic centimeter. The range of bulk densities for 1182 (melamine-formaldehyde) was given as 1.47 to 1.52 grams per cubic centimeter. Based on these data, the number of hydrogen atoms per unit volume was more than 30 percent less than that of unborated water. Additionally, based on thermal cross sections for hydrogen and nitrogen of 0.332 and 1.9 barns respectively, the neutron absorption for 1182 per hydrogen atom was a factor of 5.9 larger than for water. Therefore, vendor product 1182 was both a poorer moderator and a stronger neutron absorber than unborated water. Thus, if 1182 were to replace water in defueling canisters containing fuel, a significant reduction in neutron multiplication would result.

– *Evaluation of Diatomaceous Earth*. Diatomaceous earth would be used as the body-feed material for the filter canisters. The material consisted primarily of silicon dioxide. The neutron moderation and absorption ability of the material was essentially zero; therefore, the material would not have an adverse effect on criticality safety.

o *Conclusion*. The use of coagulants 1192 and 1182 and diatomaceous earth in the RCS and defueling canisters was evaluated and found to have no adverse effects on criticality safety. The canister evaluation was applicable to all canister types and took no credit for the soluble poison that would reside in the canisters while stored at TMI-2. Therefore, the conclusion that there were no adverse effects on canister neutron multiplication also applied to canister shipment off site.

- ***NRC Review: Criticality***. (218) The NRC had previously approved (219, 220) the use of the coagulants and filter-aid material in the DWCS, but the agency had also specified that filters containing the materials could not be dewatered until the completion of a criticality analysis. The NRC performed independent calculations that confirmed the conclusion in the licensee's criticality evaluation report (221) for coagulants. The NRC's and licensee's evaluation reports concluded that the two chemical coagulants and diatomaceous earth would have no adverse effects on criticality safety. The NRC approved the use of the proposed material and the release of the defueling canisters containing them for dewatering and shipping, subject to the applicable constraints of the approved fuel shipping program.

3.5.2.6 Filter Canister Media Modification

- ***Purpose***. To reduce the potential for clogging of the filter canister's filter elements by increasing the pore size of the filter bundle in some of the filter canisters from 0.5-micron nominal size (2 microns absolute) to 16-micron nominal size (25 microns absolute).

- ***Evaluation: Criticality***. [222] The licensee's safety evaluation stated that the change in the filter media design was the only change to the internals of the filter canister. The filter bundle with the new media design was slightly heavier than the previous bundle. This additional weight, which resulted from an increase in the quantity of stainless steel within the canister, was expected to result in a greater neutron poisoning effect. This conclusion was based on previous canister evaluations that showed an increase in stainless steel within a canister would result in a lower effective neutron multiplication (k_{eff}). Therefore, in the normal configuration, canisters with the new media would have a k_{eff} less than the value calculated for the original media design.

In the design accident configuration, the internals of the filter canister were deflected to one side as a result of the dropping of the canister. The k_{eff} in the canister had been shown to increase with increasing deflection. The evaluation indicated that the new elements had greater load-carrying capabilities; therefore, less deflection would result. Thus, the potential increase in k_{eff}, resulting from the design accident, was determined to be bounded by the expected values for the original filter media design.

Based on the results reported in the evaluation, the new filter media was not expected to have an adverse impact on the existing analyses. Therefore, the normal k_{eff} and accident k_{eff} for the filter canisters as calculated for the original filter design (i.e., 0.839 and 0.892, respectively) were bounding for filter canisters containing the new filter media (i.e., 16 microns nominal).

- ***NRC Review***. Editor's Note: The NRC's safety evaluation report was not located.

3.5.2.7 Addition of a Biocide to the Reactor Coolant System

- ***Purpose***. To evaluate the effects of adding hydrogen peroxide as a biocide to the reactor coolant system (RCS) to aid the removal of biological contamination. This biological growth reduced water clarity and fouled the water processing system filters.

- ***Evaluation: Boron Dilution***. [223] The licensee's safety evaluation concluded that the addition of hydrogen peroxide to reactor coolant presented no concern from a criticality/boron dilution viewpoint. The hydrogen peroxide was added to reactor coolant system (RCS) water at a concentration of about 200 parts per million. The licensee's safety evaluation concluded that the volume of hydrogen peroxide required to provide a concentration of 200 ppm was small compared to the RCS volume; therefore, the dilution of the boron content in the RCS was of no consequence.

- ***NRC Review: Boron Dilution***. [224] The NRC's safety evaluation concluded that the biocide water solution would be borated to the RCS technical specification limit; therefore, the biocide would not present a potential for boron dilution or criticality.

3.5.3 Defueling Canisters and Operations

3.5.3.1 Defueling Canisters: Filter, Knockout, and Fuel

- ***Purpose***. To provide loading, handling, and storage of the canisters (filter, knockout, and fuel) for the long-term storage of core debris, ranging from very small fines to partial length fuel assemblies.

- ***Background***. Editor's Note: This background provides additional information on criticality prevention measures for defueling canisters (fuel, knockout, and filter), canister drop test, and experience with fuel canister filler material. References from other sources are noted.

  - *Criticality Control*. An internal shroud controlled the size of the internal cavity and provided a means of encapsulating the neutron-absorbing material used for criticality control in the fuel canisters. Each fuel canister was equipped with a square boron insert for criticality control. [225] An array of four rods around a larger central rod, where all rods contained boron carbide (B_4C) pellets, was included for criticality control in the knockout canisters. A center rod containing B_4C pellets was included in the filter canisters to ensure that the filter canister contents remained subcritical. [226]

  - *Canister Drop Tests*. In response to the NRC's questions on the safety analysis report for the criticality control for the *knockout* canister, the licensee and INEL conducted drop tests with a full-scale canister to confirm its structural integrity. [227] A series of four drop tests were conducted at the Drop Test Facility at Oak Ridge National Laboratory with a full-scale knockout canister. The tests would provide information to answer questions raised by the NRC about criticality control in canisters during a worst case hypothetical accident. These drop tests were designed to demonstrate that the internal poison rods in the canister would not be displaced beyond the values used in criticality calculations during the hypothetical drop accident conditions postulated in 10 CFR Part 71, "Packaging and Transportation of Radioactive Material." Test criteria for knockout canisters required the poison tube array to be maintained within the limits established by the criticality analysis. These limits included: (●) a maximum allowable lateral displacement of any of the five poison tubes of 1.9/2.5 centimeters from the tube's theoretical location; (●) no significant axial movement; and (●) no breach of the boundary of the poison tubes. [228]

    The defueling canisters, which were designed and fabricated according to American Society of Mechanical Engineers Section VIII coded pressure vessel specifications, were designed to withstand the effects of unrestrained drops of 6 feet 1.5 inches in air, followed by 19 feet 6 inches in water, or 11 feet 7 inches in air, while still confining fuel debris to a critically safe geometry. The evaluation determined that such performance would bound all postulated canister drops during handling, except for a potential drop from the fuel handling building's overhead crane in the truck bay. [229]

- *Lessons Learned*. Because the DOE required that the fuel debris be kept in a safe physical dimension for criticality safety, a filler material, such as a low-density concrete, was needed. However, the use of a LICON in the canister as filler material was not the best choice because the water in the LICON mixture was extremely difficult to remove. Many debris components included materials that acquired oxygen, leaving a net surplus of hydrogen that needed to be vented. So, the canister could not be permanently sealed during subsequent storage. Based on this experience, it was reported that other options were explored, such as aluminum or glass beads. [230]

- ***Revision Summary***. Editor's Note: The licensee issued four revisions of its technical evaluation report (TER). Given the importance of the defueling canisters, summaries of changes to the licensee's TER are provided below:

- *Revision 1*. [231] This revision incorporated the licensee's written responses [232] to the NRC's comments [233] on the original [234] TER. Attachment 1 on the criticality assessment of the fuel transfer system and Attachment 2 on the assessment of a drained pool scenario were included in this revision. This section presents the NRC's safety evaluation report (SER) [235] that documented the agency's review of Revision 1.

- *Revision 2*. [236] This revision reflected an update that allowed the introduction of fuel larger than standard fuel pellets into the knockout canisters. The NRC approval letter [237] stated that the agency evaluated the effects of particles larger than standard pellets on canister reactivity. The letter concluded that the effects of the larger size were negligible based on the results of a previous NRC safety evaluation. [238] This revision of the submittal included only the affected SER pages that were revised.

- *Revision 3*. [239] This revision reflected updates that included: (•) use of modified knockout canisters as dewatering filters in the canister dewatering system; (•) use of the defueling water cleanup system deep suction on canister loading; and (•) effects of filter-aid material and diatomaceous earth on canister safety. The NRC evaluated the details of this revision individually and found them to be acceptable as discussed in previous correspondence. [240] This submittal included the entire SER with attachments. Both attachments remained unchanged except for minor formatting changes and preambles added to both. Subsequent to Revision 1 of the SER, the vacuum system was added to the cleanup tool inventory that permitted fuel particle sizes greater than whole pellets. The original evaluations assumed that the most reactive fuel particle that could be in the knockout canister was an optimally moderated standard whole fuel pellet.

- *Revision 4*. [241] This revision reflected an increase in the pore size of the filter media in some of the filter canisters. In a brief letter [242] to the licensee, the NRC concurred with the licensee's conclusion that the change in filter pore size had no deleterious effects on any aspect of canister safety. This section presents this TER revision.

- ***Evaluation: Criticality (Canister Design)***. (243) The licensee's technical evaluation considered criticality issues associated with single canisters, canisters in the storage racks, and canister/storage rack interface. Criticality calculations were performed to ensure that individual canisters, as well as an array of canisters, would remain below the established effective neutron multiplication (k_{eff}) criterion under normal and accident conditions.

  - *Criticality Safety Criteria*. The established criticality safety criterion was that no single canister or array of canisters would have a k_{eff} greater than 0.95 during normal handling and storage at the TMI-2 site. For plant accidents (e.g., drained spent fuel pool), the established criticality safety criterion was a k_{eff} no greater than 0.99. These criteria were satisfied for all canister configurations.

  - *Canister Modifications*. Some modifications were made to defueling canisters after the safety evaluation of the original designs.

    - *Deep-Bed Filters*. Minor modifications were made to convert some of the knockout canisters into deep-bed filters for use in the defueling water cleanup system. The conversion to deep-bed filters consisted of changes in the internal plumbing, but these changes did not alter the placement of the poison rods in these knockout canisters. In addition, the diatomaceous earth, sand, or both added to these canisters had less moderating ability than water. Therefore, the criticality evaluations performed for the knockout canisters were determined to be bounding for the deep-bed filters. In addition, the evaluation concluded that the criticality evaluations performed on the knockout canisters with damage from a drop accident would be bounding for dropped deep-bed filters since the structural behavior of the deep-bed filters was similar to that of the knockout canisters.

    - *Coagulants and Diatomaceous Earth*. Coagulants and diatomaceous earth that were incorporated in the defueling water cleanup system to improve filter canister performance were considered in the licensee's criticality safety evaluation report (244) for coagulants. This evaluation showed that the addition of these materials in the canisters would not adversely impact the criticality evaluations for the defueling canisters. Additionally, the evaluation determined that the accumulation of coagulants and diatomaceous earth in the canisters would not adversely affect the conclusions on the subcriticality of the stored canisters in a postulated accident (dry storage pool).

  - *Criticality Models*. Criticality models were developed for each single canister, an array of canisters in a storage rack, and postulated canister drops. Further, the models were developed to analyze criticality margins for plant accidents such as loss of spent fuel pool water.

    - *Model Parameters*. The criticality models assumed the following parameter values for the canister contents: (●) Batch 3 fresh fuel only; (●) Batch 3 average enrichment plus 2 sigma (highest core enrichment); (●) no cladding or core structural material; (●) no soluble poison or control material from the core; (●) optimally moderated, stacked, standard whole fuel pellets; (●) canister fuel regions completely filled without weight restrictions; (●) uniform temperature of 50 degrees Fahrenheit (degrees F); (●) boron-10 surface density used in the Boral® [v] for the fuel canister assumed to be 0.040 gram per cubic centimeter (g/cm^3) (actual surface density would be 0.040 g/cm^3 with a 95/95 percent confidence level in the testing to provide at least a 2 sigma margin); (●) B_4C density used in the poison tubes for the filter and knockout canister assumed to be 1.35 g/cm^3 with the boron weight percent assumed to be 70 percent (actual B_4C density would be at least 1.38 g/cm^3 with a boron weight percent meeting requirements for ASTM International (ASTM)-C-750 Type 2 B_4C powder with minimum boron weight percent 73 percent).

Optimization studies were performed previously to determine the model parameter values that were used in the criticality evaluations for the canister contents and canister neutron poisons. These optimization studies were presented in the Babcock & Wilcox technical report [245] for the defueling canisters' final design, along with other parametric studies performed for special cases.

– *Analysis Cases*. The criticality analysis employed a fuel model that bounded all debris loading configurations. Three basic configurations were analyzed for each canister: (●) single standard canister configuration surrounded by water; (●) array of canisters in the storage pool; and (●) disrupted canister model resulting from a bounding (worst-case) drop.

– *Modeling Assumptions*. Key modeling assumptions included the following: (●) Standard canister configuration assumed that some minimum degree of damage could have occurred in the canisters during normal loading operations. (●) All canisters in an array analysis also included the minimum degree of damage. (●) A 17.3-inch center-to-center spacing was analyzed for the array cases. This spacing accounted for all storage rack tolerances and the minimum center-to-center spacing possible for any two canisters. (●) Canisters were loaded with fuel debris that consisted of whole fuel pellets enriched to 2.98 weight percent and optimally moderated with unborated water at 50 degrees F. (●) For accident conditions, optimized fuel was present in both normal fuel locations and in all void regions internal to the canister. Filling all void regions with fuel had the effect of adding fuel to the canister after a drop. (●) The canister shell, including the lower head, was identical for all three canister types; the concave inner surface was explicitly modeled but with the rounded corners squared off (this increased the volume of the lower head). (●) The protective skirt and nozzles on the upper canister head were not modeled. (●) The catalytic material and its structural supports, which were installed for hydrogen recombination in both the lower and upper heads of all three canisters, were replaced with fuel. (●) In the storage rack cases, the canisters were stored in unborated water in a 17.3-inch minimum center-to-center spacing.

[v] Editor's Note: Boral® is an aluminum boron carbide alloy used as a neutron absorber.

- *Additional Analysis Cases (Canister Drop)*. Three cases were examined for a dropped canister: (●) vertical drop; (●) horizontal drop; and (●) combined vertical and horizontal drop. The TER reported that shell deformation was essentially the same for all cases. The criticality analyses assumed that the cylindrical shell would not deform for these drops. Any deviation from the cylindrical shape would have increased the surface-to-volume ratio and increased the neutron leakage from the system.

- *Additional Modeling Assumptions (Canister Drop)*. Following a drop, a teardrop shape expansion was assumed to occur in the lower head region of the canister shell. The bottom head was modeled as a flat plate with the internal components resting on the plate. To bound all drop cases, the canister was assumed to rotate during the drop and land on its head. A similar teardrop shape would result. Both of these cases were merged into a single model that assumed the teardrop deformation at both the top and bottom with the internals displaced to the flattened lower head surface. For the combined vertical-horizontal drop, the radial displacement of the internal components was combined with the double teardrop model. This drop model bounded any conceivable drop configuration by exceeding conservative stress estimates of deformation.

o *Results*. Table 3-1 of the licensee's TER presents the results of criticality analyses that used basic three-dimensional canister models (filter, fuel, and knockout). These results represented bounding values for any configuration of the canisters at TMI-2 and showed that for any configuration, the k_{eff} would be less than 0.95 (ranging from 0.839 to 0.915), with uncertainties included. The highest value was from the standard array of knockout canisters (k_{eff} of 0.915). Because of the conservative assumptions built into the models, the k_{eff} of any actual configuration would be less than these bounding values.

  Sensitivity studies were performed on a nominal 18-inch center-to-center array spacing to determine the effect of a canister dropped outside of the rack. These analyses showed that k_{eff} was less than 0.95 for canisters dropped outside the rack, as long as the side of the dropped canister did not come within 2 inches of the side of the nearest canister in the rack. As stated in the licensee's technical evaluation report [246] for fuel canister storage racks, the storage rack design met this requirement.

o *Additional Sensitivity Analyses*. The licensee conducted a series of analyses to show the impact of changes to the following four modeling assumptions: (●) type of poison used in the filter and knockout canisters; (●) storage pool water temperature; (●) fuel particle size; and (●) change in filter media.

  - *Poison Type*. The values reported in Table 3-1 for the filter and knockout canisters assumed that the poison tubes for the canisters were filled with vibrapacked [w] B_4C powder. Actual fabricated filter and knockout canisters contained compressed sintered B_4C pellets. This change resulted in a small reduction to the diameter of the poison in

---

w Editor's Note: Sintered B_4C particles are vibrated into place in the tubes.

the canisters, which resulted in a small increase in the k_{eff} of the two canister types. Based on analyses, the net increase in neutron multiplication (delta-k) would not exceed 0.41-percent delta-k.

- *Temperature*. The values reported in Table 3-1 assumed a minimum temperature of 50 degrees F for all canister types. For canisters stored in the spent fuel pool, the temperature could be as low as 32 degrees F. Explicit criticality array calculations were not performed at this lower temperature. Rather, an evaluation was performed to determine the maximum increase in multiplication due to cooling from 50 degrees F to 32 degrees F. The maximum change in multiplication was determined to be an increase of 0.1-percent delta-k.

- *Particle Size*. The results reported in Table 3-1 were also based on the assumption that no single fuel mass greater than a whole fuel pellet existed in the TMI-2 core. Examinations of the core showed that fuel melting could have occurred. To assess the impact of this possibility, an evaluation was performed to determine the neutron multiplication for the most reactive Batch 3 fuel particle size. The neutron multiplication for the optimum size particle was only 0.07-percent delta-k higher than the neutron multiplication for the standard whole pellet. The corresponding increase in k_{eff} would be about the same magnitude. Therefore, there was no limit on the sizes of fuel particles that could be placed in the fuel and knockout canisters.

- *Filter Media*. The results reported in Table 3-1 for the filter canisters were based on the original 0.5-micron filter media design. Some filter canisters could be equipped with a filter media that removed particulates only down to about 16 microns. Since the filter canisters using the new filter media were otherwise identical to the original filter canisters, this change in media would not significantly affect the canister neutron multiplication. Therefore, the study concluded that the normal configuration results for the filter canister, as reported in Table 3-1, were bounding for filter canisters equipped with modules using the new filter media.

  - To address the accident configuration for filter canisters using the new filter media, the licensee's SER [247] for the filter canister media modification demonstrated that the new filter modules had greater load-carrying capabilities than the original filter modules. Thus, the deflections of the canister internals determined for canisters with the original modules bounded the deflections expected for canisters with the new modules. Therefore, the accident configuration filter canister results reported in Table 3-1 were bounding for canisters employing the new filter media.

- *Conclusion*. The licensee's sensitivity analyses showed that changes in k_{eff} stemming from the four modified assumptions would not result in exceeding the k_{eff} criterion of 0.95 for the cases reported in Table 3-1.

o *Conclusion*. The licensee's TER concluded that results of the criticality analyses, which used basic three-dimensional canister models, were bounding for any configuration of the canisters at TMI-2. The results showed that, for any configuration, k_{eff} would be less than 0.95, with uncertainties included. Because of the conservative assumptions built into the models, the evaluation concluded that the k_{eff} of any actual configuration would be less than the calculated bounding values.

- ***Evaluation: Criticality (Drained Spent Fuel Pool)***. [(248)] The licensee's safety evaluation addressed the criticality concern associated with a drained spent fuel pool. Attachment 2 of the report presented two drained spent fuel pool cases with different states of internal canister moderation. These cases were judged to be bounding with respect to the actual contents of the canisters during the unlikely event of loss of pool water. Case 1 assumed an optimal fuel volume fraction in 4350 parts per million (ppm) boron moderator, full density, and 50 degrees F. Case 2 assumed a realistic fuel volume fraction with pure water moderator, 100-percent humidity conditions, and 50 degrees F.

o *Calculation Models and Procedures*. In both cases, the basic canister model was the standard configuration knockout canister. For conservatism and to facilitate modeling in KENO standard geometry, the four satellite poison tubes and all lateral support plates were omitted, and their space was occupied by fuel. Additional conservatism was provided by assuming an infinite canister array and an enhancement of overhead reflection by concrete, which was modeled above the array with a 17.3-inch square pitch.

For Case 1, the optimal fuel volume fraction was determined by NULIF calculations to be 0.620 with an infinite neutron multiplication ($k_{infinity}$) of 1.02890. The cell weighted cross section for the KENO calculations was generated by NITAWL/XSDRNPM calculations. [(x)]

For Case 2, a measured fuel volume fraction for randomly packed whole fuel pellets was used. This volume fraction was 0.624, which by coincidence was close to that of Case 1. NULIF calculation for this volume fraction with saturated steam (nonborated pure water) moderator gave a $k_{infinity}$ of 0.65706. Further NULIF calculations at this fuel volume fraction versus increasing water density gave a monotonically increasing $k_{infinity}$ up to 1.21412 at 100-percent water density. Case 1 covered subcooled water that remained after the dewatering process. This water would be borated.

o *Results*. For Case 1, the calculated maximum k_{eff} was 0.964, which included a 0.02 benchmark uncertainty and the 2-sigma KENO uncertainty. This was for an infinite X-Y array with no concrete side reflection. The effect of concrete reflection on the side rather than an additional knockout canister was shown to be negative with respect to reactivity.

For Case 2, the very low value of $k_{infinity}$ ensured that k_{eff} for an array of canisters would be well below that for Case 1. This was verified by a KENO calculation using an infinite

[x] The NULIF code was used primarily for fuel optimization studies in a 111-energy group representation. NITAWL and XSDRNPM were used for processing cross sections from the 123-group AMPX master cross-section library. NITAWL provided the resonance treatment and formatted the cross section for use by either XSDRNPM or KENO-IV. In most cases, XSDRNPM cell-weighted cross sections were used in the KENO-IV calculations, but for some comparative fuel optimization runs, KENO-IV directly used the NITAWL output library.

17.3-inch pitch array, which yielded a k_{eff} of 0.632 including uncertainties. The effect of concrete reflection was found to be negative for this case also.

The results of this analysis assumed that the most reactive fuel particle that could be in the knockout canister was an optimally moderated, standard whole fuel pellet. However, this assumption was no longer appropriate because of the subsequent change to the vacuum system that permitted fuel particle sizes greater than whole pellets to be loaded into a knockout canister. The analysis in the original evaluation used conservative assumptions (e.g., four satellite poison tubes were not modeled). Additionally, the probability of a drained pool scenario occurring was small. Therefore, even though the analysis using optimum size fuel particles was not repeated, the conclusion that k_{eff} would not exceed 0.99 for a drained pool scenario was considered adequate.

- *Conclusion*. The evaluation concluded that no realistically conceivable condition could occur during a spent fuel pool drainage event to cause a k_{eff} value to exceed the specified 0.99 acceptance criterion. This assumed that administrative control precluded diluting or reflooding the canister contents with pure water.

- ***Evaluation: Criticality (Lead Shielding)***. Editor's Note: The defueling canisters were transferred to locations within the reactor and fuel handling buildings using a transfer shield containing lead. Transfer of canisters to the shipping cask used a different device, called a "transfer cask." SER [(249)] Attachment 1 addressed the criticality concerns associated with the use of lead shields around the canisters. Given that this criticality analysis was extensive, it is summarized in Section 2 of this chapter and not repeated in its entirety here.

- ***Evaluation: Criticality (Load Drop)***. [(250)] The licensee's safety evaluation stated that canisters were capable of withstanding bounding (worst case) accidents. Vertical drops of about 6 feet in air followed by 19.5 feet in water, or about 11.5 feet in air, were considered, along with a combination of vertical and horizontal drops. These drops were analyzed to bound a drop in any orientation. For these cases, the structural integrity of the poison components must be maintained, and the canister must remain subcritical. Deformation of the canister was acceptable. Leakage of the entire contents of core material from the canister was unlikely but was allowed, provided that the contents left in the canisters remained subcritical. Refer to NUREG/KM Chapter 7 on load drop evaluations for the complete evaluations.

- ***NRC Review: Criticality (Summary)***. [(251)] The NRC's safety evaluation stated that the defueling canisters were designed to ensure that their contents remained subcritical under all normal operational conditions and during all postulated accident conditions. The conditions analyzed included both a single-canister configuration and an array of canisters on a 17.3-inch center-to-center spacing, which was the minimum spacing for all onsite storage rack locations. Both an intact canister and a canister deformed by the worst case drop accident were modeled. The deformed geometry used in the calculations was that predicted by the structural analysis with additional conservative assumptions for the poison structure location. The canisters were modeled using computer codes generally recognized as acceptable by the NRC.

- *Licensee's Modeling Assumptions and Results*. The NRC's safety evaluation summarized the licensee's modeling assumptions and results:

  – *Modeling Assumptions*. The calculational model used the following conservative assumptions: (●) The canister's contents consisted of Batch 3 fuel only, with the average Batch 3 enrichment of 2.93 percent plus 2 standard deviations and were in the highest enriched region of the core. (●) No fissile burnup or fission product inventory would produce a negative reactivity. (●) The canister did not contain any cladding, core structural material, soluble poison, or control material (i.e., control rod debris or burnable poison) from the core. (●) The contents were fuel in the optimal lump size with an optimal fuel-to-moderator ratio and no boration in the entrained water. (●) All void regions in the canister were filled with fuel without regard to the weight restrictions on a loaded container. (●) All three types of canisters contained catalytic recombiners in the upper and lower heads. The criticality analysis assumed that the regions occupied by the recombiners were filled with fuel. (●) The loading of fixed poison material was the lowest possible loading of fixed poison material. (●) The canister geometry was conservatively modeled to account for the internal configuration, the structural members of canister internals, and the closure heads.

  – *Results (Fuel Canister)*. The fuel canisters were analyzed for a single canister infinitely reflected by water, an infinite array of canisters in unborated water, and a canister deformed by the bounding drop case. The deformed case assumed fuel had migrated into the bulged lower and upper heads. All cases yielded a maximum k_{eff} of 0.877.

  – *Results (Knockout Canister)*. The evaluation considered two knockout canister configurations. These included the standard undamaged configuration and the damaged configuration that assumed the worst deformed geometry. The damaged configuration did not assume that fuel had migrated into the ends of the canister, which differed from the assumption used for the fuel and filter canisters. In addition, the loss of the B_4C pellets was not assumed for the knockout canister as it was for the filter canister. Oak Ridge National Laboratory performed a drop test of an as-built knockout canister and demonstrated that the bottom support plate and the poison rods remained intact following the maximum predicted impact loads from a drop accident. These configurations were analyzed as a single canister infinitely reflected by water, an infinite array of undamaged canisters in unborated water, and a single dropped canister. The maximum calculated k_{eff} was 0.915.

  – *Results (Filter Canister)*. Two filter canister configurations were also considered. The licensee assumed fuel above the lower support plate and a second configuration with fuel in the lower head plus fuel filling the filter element drain tubes (i.e., ruptured filters). The maximum calculated k_{eff} when considering the single canister, the array of canisters, and the single dropped canister was 0.892.

- *The NRC's Conclusion*. The NRC performed independent calculations to verify the licensee's criticality analyses. These included computer code analysis of several test cases, as well as an evaluation of the assumptions and the computer codes used by the licensee. The NRC's results agreed with those of the licensee.

  The licensee presented additional analysis to determine the effects on criticality of the canister transfer shielding. The NRC determined that the analysis used acceptable analytic techniques with appropriate conservative assumptions. This analysis showed that handling an undamaged full canister in the proposed transfer shield would not result in a k_{eff} of greater than 0.95. Analysis of a damaged canister would be performed on a case-by-case basis as needed.

  Details of the NRC's criticality analyses were included in Appendices 1 and 2 to the NRC's SER. The NRC's evaluations considered the loading of a single defueling canister (Appendix 1) and the accidental drop of the contents of a loaded canister into a similar loaded canister (Appendix 2).

- ***NRC Review: Criticality (Single Canister Loading)***. (252) Appendix 1 of the NRC's SER (253) presented the criticality evaluation for the loading of defueling canisters. The licensee's criticality analysis, which was performed by Babcock & Wilcox (B&W), was reviewed for each of the three defueling canister designs (fuel, knockout, and filter), as well as the associated nuclear data and geometric details. Based on the review of these documents, the NRC found that the criticality calculation method, physical and geometric assumptions, atomic number densities (giving mass loadings of nuclides per region), and description of canisters analyzed were accurate and representative of the intended case analysis.

In addition to the document review, the NRC performed independent KENO-IV Monte Carlo calculations of the knockout and fuel canisters. These independent calculations agreed with the results obtained by B&W as given in the licensee's TER. Table 1 of Appendix 1 to the NRC's SER showed a comparison of k_{eff} results by the NRC and B&W under various conditions. The NRC did not evaluate the filter canisters since they contained a 2-inch diameter B_4C central poison rod similar to that in the knockout canister and also contained about 10 times more internal steel than in the knockout canister; therefore, the filter canister was considered less reactive than the knockout canister.

- *Results*. The NRC recommended acceptance of the criticality analysis portion of the licensee's TER and concurred with the licensee that there was at least a 5-percent shutdown margin (k_{eff} no greater than 0.95) for all three canisters under normal and postulated accident modes.

  The NRC noted that B&W did not report any k_{eff} results for B_4C replaced by water or replaced by a void. The NRC calculated a single knockout canister to have a 4.3-percent shutdown when the B_4C was replaced by water, and a 3.8-percent shutdown when the B_4C was replaced by water with the remaining steel tubes deflected off-center by 1.2 inches. The NRC noted that these two cases were supercritical (1.033 and 1.041, respectively) for the infinite array calculation as given in Table 1 of the appendix. Therefore, if the above scenarios could be realized in the postulated accident modes, the 5 percent shutdown margin would be compromised. Further, if a void replaced the B_4C, the shutdown margin was further reduced from 4.3 percent to 3.4 percent.

- *Conclusion*. In summary, the NRC concluded the following: (●) B&W's calculational methodology (KENO IV-123 Group GAM/THERMOS cross-section library) represented one of the best state-of-the-art approaches, which successfully calculated many appropriate benchmark criticality analyses. In particular, the NRC noted that the B&W fuel-water homogenization procedure, which was fundamental to the B&W approach and results, was performed correctly. (●) B&W's criticality analyses used the most (neutronically) reactive fuel/water mixture in representing the core debris in each canister. (●) Some conservative assumptions used by B&W included: each canister was loaded up to a height of 14 feet, which was about an extra 3 feet of reactive material; the density of B_4C was taken as 1.35 g/cm^3; the area density of boron-10 was taken as 0.04 g/cm^3; and a minimum amount of steel was credited to the knockout canister (about 1.5 volume percent) and the filter canister (about 14 volume percent).

  The independent calculations performed by the NRC corresponded with the B&W results for the cases considered.

- *Modeling Assumptions*. Appendix 1 to the NRC's SER discussed the criticality methods and assumptions used to establish the conservative parameters fundamental to both B&W's and the NRC's criticality evaluations.

  – *Canister Contents*. Both B&W and the NRC assumed that the fuel debris contents for all three canister types contained: (●) uranium dioxide; (●) unclad pellets; (●) unborated water moderator; and (●) a volume fraction of 0.30 of fuel and 0.70 of water. This content combination was established through many independent calculations to constitute the most reactive mixture.

  – *Moderator*. For a borated water system with a range of 3000 to 5000 ppm boron, the most reactive mixture turned out to be a volume fraction of 0.60 of fuel and 0.40 of water. However, for these borated systems, the k_{eff} was on the order of 30 percent less than that for any corresponding system moderated by unborated water. Thus, the delta-k of 0.3 bounded the most reactive mixture to be fuel moderated by unborated water. Therefore, all criticality calculations used unborated water as the moderator.

  – *Boron Content*. Both B&W and the NRC assumed a very conservative density for B_4C of 1.35 g/cm^3 versus 2.43 g/cm3 given in the handbooks. In addition, an area density of 0.04 g/cm^3 for boron-10 was assumed for the Boral® plates.

– *Criticality Codes*. Both B&W and the NRC used the KENO-IV Monte Carlo computer program with the 123 group GAM/THERMOS neutron cross-section library that adjusted the resonance nuclide (uranium-238) with the NITAWL [y] program.

– *Criticality Models and Results*. The NRC performed criticality analyses of a single canister (water flooded and reflected) and an infinite array of canisters (17.3-inch center-to-center spacing in water pool) for each of the knockout and fuel canisters. The undamaged knockout canister was modeled for three cases: (●) B_4C in place; (●) B_4C replaced with water; and (●) B_4C replaced by a void. The knockout canister internals contained an array of four outer poison rods and one central poison rod. The damaged knockout canister model assumed that all B_4C rods were replaced by water and the stainless-steel rods were displaced off center by 1.2 inches. The undamaged fuel canister was modeled with the surrounding Boral® plates in place.

  The filter canister contained a 2-inch-diameter B_4C central poison rod similar to that in the knockout canister but did not contain the four outer rods. In addition, it contained about 10 times the amount of internal steel of that in the knock-out canister. The NRC determined that the filter canister was considered less reactive than the knock-out canister because of the center rod and extra internal steel; therefore, the NRC did not analyze the filter canister.

  - *B&W Models*. B&W homogenized the uranium dioxide and the associated water (30/70 mixture) using an XSDRN cell group-spatial weighting into a debris mixture. Using generalized geometry, this homogenized water-fuel mixture occupied all the space: (●) within the Boral® plates of the fuel canister; (●) inside the knock-out canister not occupied by the five B_4C stainless-steel clad rods; and (●) inside the 17 filter elements of the filter canister.

  - *Fuel Canister Model (Undamaged)*. As a check on the B&W homogenization procedure, the NRC's model required that the uranium dioxide pellet be described as a discrete cylinder surrounded by the cell (30/70 volume ratio) water. This restricted the NRC's canister's geometry to a square cylinder. The pellet-water constituted a box-type in the KENO-IV geometry, and since the fuel canister possessed a square internal region (surrounded by Boral® plates) that would contain the debris, this geometry represented the ideal case to check the homogenization process fundamental to B&W's calculated procedure. Results (refer to Table 1 of Appendix 1 to the NRC's SER) showed the homogenization procedure of B&W and the discrete procedure of the NRC to be equivalent. Both calculated the same k_{eff} for the single fuel canister and for an array.

  - *Knockout Canister Models (Undamaged)*. For the knock-out canister, the square cylinder geometry of the NRC maintained the exact masses of uranium dioxide, water, steel, and B_4C that existed in B&W's cylindrical geometry. Results of the

[y] Editor's Note: Nordheim Integral Treatment and Workline Library (NITAWL) uses the Nordheim Integral Treatment to perform neutron cross-section processing in the resolved resonance energy range.

undamaged single knock-out canister with B_4C in place showed excellent agreement between the two methods. For the infinite array, the NRC value of $k_{infinity}$ was higher by about 4.5 percent, since in this geometry, the square box ends came much closer to neighboring boxes, whereas the cylinders remained effectively further apart from one another.

- *Knockout Canister Models (Damaged)*. The damaged cases for the NRC assumed the B_4C was replaced by water, whereas B&W assumed only a displacement of the B_4C stainless-steel rod. Although the NRC's condition was more severe, the single damaged knock-out canister was still subcritical, but the infinite array of such damaged canisters was supercritical.

– *Sensitivity Analyses*. The NRC performed a series of sensitivity analyses for the undamaged knockout canister that assumed 3000 ppm of boron in place of unborated water. Calculation results showed a k_{eff} that ranged from 0.582 to 0.618 for the undamaged single canister with flooded and reflected borated water and for the undamaged infinite array of canisters in a borated water pool.

- ***NRC Review: Criticality (Double Canister Loading)***. [254] Appendix 2 of the NRC's SER [255] provided the criticality evaluation for the accidental drop of a loaded canister and its contents onto another similarly loaded stored canister. (The criticality analyses presented in Appendix 1 assumed only one canister loading.) Results showed that the k_{eff} shutdown margin was between 32 percent (maximum) to 13 percent (minimum). These analyses conservatively considered loading limitations per canister, maximum storage volume per canister available, and 4350 ppm boron in water. The NRC concluded that such an accident posed no criticality hazard based on the analysis of seven KENO Monte Carlo (123 groups) cases.

o *Modeling Assumptions*. The NRC's criticality evaluation included the following assumptions in the analyses:

– *Storage Volume*. The stored canisters configured in a parallelepiped (where each face is a parallelogram), borated (4350 ppm) water region (storage volume) of dimensions 18 inches by 18 inches by 14 feet with a volume of 802,000 cubic centimeters.

– *Canister Contents*. Each canister had a maximum capacity of 900 kilograms dry uranium dioxide pellets with a density of 10 g/cm^3. This nominal value was 4.5 percent higher than the greatest payload (861 kilograms total) for a knockout canister.

– *Neutron Absorbers*. No structural material or solid poison material was present in the storage volume of the canisters. Cases with borated moderators were analyzed for three concentrations: 0 ppm, 3000 ppm, and 4350 ppm.

– *Neutron Reflectors*. Storage volume was surrounded by a 1-foot-thick borated water reflector.

- *Fuel Geometry*. The as-built pellet was in the form and geometry that presented an optimum fuel volume to water volume ratio for both unborated and borated water cases.

- *Fuel/Moderator Volume Ratios*. Previous criticality studies (refer to Appendix 1 of the NRC's SER) were performed for uncladded, cylinder fuel pellet that was surrounded by a cell of water of various boron concentrations. For unborated water, the maximum reactivity was a fuel pellet with a 30/70 fuel/water volume ratio. In this case, water was more important than fuel (more fuel was less reactive). For borated water, maximum reactivity for the fuel pellet shifted to 60/40 fuel/water volume ratio over the boration of 2500 ppm to 4500 ppm. In this case, fuel was more important than the borated water. Since the actual ratio was from 58/42 to 62/38 over the boration range, which showed small dependence on boron concentration, the double loading analyses assumed an average value of 60/40. [z] For these borated systems, the k_{eff} was of the order of 30 percent less (or delta k_{eff} of 0.3) than any corresponding system moderated by unborated water.

- *Fuel Loadings*. Simple calculational results (refer to Table 1 in Appendix 2 of the NRC's SER [256]) showed that the contents of about six canisters would be needed to approach the optimum 60/40 ratio for borated systems, whereas the contents of about three canisters were needed to approach the optimum 30/70 for unborated systems. It should be noted that these optimum ratios conservatively contained more fuel than the one canister dropping into another canister.

  The criticality analysis of the various cases was modeled in cells of a discrete pellet region, surrounded by its associated moderator close-fitting into the 18-inch by 18-inch cross-sectional area of the canister storage volume. This gave a uranium dioxide mass loading of 2764 kilograms (versus 2700 = 3 x 900) for the 30/70 ratio and 5678 kilograms uranium dioxide (versus 5400 = 6 x 900) for the 60/40 ratio because of the arithmetic discrepancies from fitting prescribed volume fractions into a fixed region. The 30/70 case was slightly nonconservative, whereas the 60/40 was conservative since there was more fuel to contribute to a possible reactivity event.

o *Results*. Table 2 of Appendix 2 of the NRC's SER [257] presented the NRC's results.

- *Unborated Water Cases*. For the unborated water cases, a comparison of Case 1 (30/70 volume ratio) and Case 4 (60/40 volume ratio) showed that k_{eff} decreased by 0.14 (1.230 and 1.099, respectively) by increasing the fuel by a factor of 2 (30 to 60 percent). The more reactive 30/70 mixture Case 1 was in agreement with the results that was reported in previous studies for a single fuel pellet model.

- *Borated Water Cases*. For the 3000 ppm borated water cases, a comparison of Case 2 (30/70 volume ratio) and Case 5 (60/40 volume ratio) resulted in an increase in k_{eff} of 0.14 (0.775 and 0.918, respectively) by increasing the fuel by a factor of 2 (30 to

[z] Editor's Note: Single loading was represented by the 30/70 ratio (30 percent fuel) and the double loading had the 60/40 ratio; thus doubled the amount of fuel (60 percent fuel).

60 percent). For the 4350 ppm borated water cases, a comparison of Case 3 (30/70 volume ratio) and Case 5 (40/60 volume ratio) resulted in an increase in k_{eff} of 0.19 (0.677 and 0.871), by increasing the fuel by a factor of 2 (30 to 60 percent). The more reactive 60/40 mixture cases (Case 5 and 6) were in agreement with previous studies for a single fuel pellet model.

Results indicated that the k_{eff} shutdown margin (1 minus k_{eff}) with a 4350 ppm boron moderator concentration was between the maximum of 32 percent (30/70 ratio) to a minunum of 13 percent percent (60/40 ratio).

Case 6 represented approximately six canister loadings that filled the storage volume at the most reactive mixture 60/40 with 4350 ppm boron in the storage water. If the canister components (structural and solid neutron absorbers) and canister core debris contents (control rod poisons, fixed poisons, core structure material, fission products, and lower average core enrichment) were considered in the analyses, then the k_{eff} of all cases could be decreased by at least 0.10. Since only two canister contents (fuel and moderator) that represent accident conditions were included in the model, subcriticality was ensured by a large margin.

- *Sensitivity Analyses*. The NRC performed three sensitivity analyses with borated water systems.

  - Case 7 represented a 14-foot-deep infinite slab (X-Y direction) with Case 6 contents (4350 ppm boron and 40/60 volume ratio with six loadings) that resulted in a k_{eff} of 1.085.

  - Case 3 (4350 ppm boron and 30/70 volume ratio) was reanalyzed as an infinite system (X-Y-Z direction) that resulted in a $k_{infinity}$ of 0.8021.

  - Case 7 was reanalyzed as an infinite system (X-Y-Z direction) that resulted in a $k_{infinity}$ of 1.095

- ***NRC Review***. [258] Refer to NUREG/KM Chapter 7 on load drop evaluations for the NRC's safety evaluation of this topic.

3.5.3.2 Replacement of Loaded Fuel Canister Head Gaskets

- ***Purpose***. To replace head gaskets on the loaded fuel canisters in spent fuel pool "A" (SFP-A) because of excessive leakage. The replacement of gaskets involved: (●) moving a loaded canister from the SFP-A storage rack to the dewatering station rack with the canister handling bridge; (●) removing the canister head; and (●) transferring the head to a worktable. At the worktable: (●) the head was rotated for access to the gaskets; (●) the metallic gaskets were removed; (●) new synthetic gaskets were installed; and (●) the head was inspected before reinstallation.

- ***Evaluation: Criticality***. [259] The licensee's safety evaluation stated that various measures were in place to ensure that a fuel canister with its head removed maintained an effective neutron multiplication (k_{eff}) below the licensing criteria for both planned operations (k_{eff} no greater than 0.95) and accident conditions (k_{eff} no greater than 0.99). First, the technical evaluation report [260] for the defueling canisters analysis demonstrated that the maximum k_{eff} for a single loaded fuel canister moderated with unborated water was 0.857. The removal of the canister head was not expected to appreciably affect this value. Additionally, the canister would remain in SFP-A during the gasket replacement. As the recovery technical specification required the water in SFP-A to be borated to at least 4350 parts per million (ppm), the water within any open fuel canister would also be borated. Taking credit for the borated water within the canister would have reduced the k_{eff} to a value well below 0.857. Consequently, this evaluation concluded that the planned activities associated with the head gasket replacement would not result in a canister k_{eff} exceeding the licensing criterion of 0.95 for normal conditions.

Three postulated accident conditions were also evaluated. The first postulated accident was the emptying of an open canister's contents into SFP-A. The licensee's criticality safety report [261] for the reactor coolant system evaluation determined that the 4350-ppm boron concentration in SFP-A would ensure a k_{eff} no greater than 0.99. The second postulated accident was an inadvertent filling of an open fuel canister with unborated water. As demonstrated in the previous paragraph, the k_{eff} for an open fuel canister filled with unborated water was considerably less than 0.99 (i.e., about 0.857). Finally, the last postulated accident was the drained pool condition. In this case, any water remaining in the open canister would be borated to at least 4350 ppm, thus ensuring that the k_{eff} would not exceed 0.99. The evaluation noted that the presence of the boron material shroud inside the fuel canister would tend to reduce the dry pool k_{eff} even further. Therefore, the evaluation concluded that, if the postulated accident conditions were to occur during the head gasket replacement activities, the resultant k_{eff} would not exceed the accident condition licensing criterion of 0.99.

- ***NRC Review: Criticality***. [262] The NRC's safety evaluation concluded that the heavy load handling and criticality control aspects of the proposed activity were bounded by the NRC-approved safety evaluations for the defueling canisters and SFP-A fuel canister storage racks.

3.5.3.3 Use of Debris Containers for Removing End Fittings

- ***Purpose***. To use modified fuel canisters as "debris containers" for removing fuel assembly upper end fittings, control component spiders, or other structural material from the reactor vessel. This activity was performed to expedite access to the vacuumable fuel and debris in the core. The modified fuel canister did not have internal neutron-absorbing plates, concrete filler, recombiner catalyst, dewatering capability, or a relief valve. After the debris containers were loaded, they would be closed and stored in the spent fuel pool "A" (SFP-A) racks until final dispositioning of the containers and their contents. There were no plans to use these debris containers for shipment. Since these canisters would not have relief valves installed (a prerequisite for shipping), they could be easily identified.

- ***Evaluation: Criticality***. [263] The licensee's safety evaluation stated that the potential criticality concerns were evaluated and found to be bounded by the previous criticality evaluation report [264] for the reactor coolant system (RCS) and the criticality evaluation report [265] on limits of foreign materials allowed in the RCS during defueling activities. The licensee's evaluation considered potential criticality concerns while the container was being loaded in the reactor vessel, transferred to SFP-A inside the canister transfer shield (CTS), and stored in the SFP-A racks.

  - *Loading and Storage*. When the containers were loaded in the reactor vessel or temporarily stored in either the fuel transfer canal or SFP-A, they were submerged in water with a boron concentration of at least 4350 parts per million (ppm). Since the containers were vented, the evaluation assumed that the boron concentration of the water within the containers was equal to the boron concentration of the surrounding water. The previous RCS criticality analysis demonstrated that the core would remain shut down, with an effective neutron multiplication (k_{eff}) no greater than 0.99 when the RCS water was borated to a concentration of at least 4350 ppm. Although differences existed between the assumptions used in the previous RCS criticality analysis and those that would actually have been used for an explicit analysis of submerged containers, direct application of the RCS criticality results to this evaluation was determined to be conservative in two ways.

    First, the reference model in the RCS criticality analysis included the entire TMI-2 core, which meant less neutron leakage than for the debris containers with their significantly smaller fuel quantities. Second, the reference model in the RCS criticality analysis did not include neutron-absorbing structural materials. These materials constituted the majority of the contents in the debris containers. Therefore, based on the results of the RCS criticality analysis and these two conservative assumptions, the evaluation concluded that the containers would be subcritical (k_{eff} no greater than 0.99) when one or more containers were submerged in borated water and the containers were filled with borated water, both with a boron concentration of at least 4350 ppm.

  - *Canister Transfer Shield*. When a container was inside the CTS, the lead and steel walls of the CTS acted as an additional neutron reflector, tending to increase k_{eff}. The results from the previous criticality evaluation reports (on RCS and foreign materials) were used in this present evaluation to demonstrate that the containers were subcritical when inside the CTS. The present evaluation identified conservative aspects of the criticality models used in the previous analyses, while considering the expected configuration of the container in the CTS and expected operating conditions (e.g., thicker reflectors surrounding the core and lower boron concentration). When considering the conservatisms and actual considerations, the k_{eff} would be reduced. Based on these considerations, the evaluation concluded that the resultant k_{eff} would be below 0.99, and the containers would remain subcritical when within the CTS.

- *Loading Restrictions*. The above conclusions were reached independently of the containers' fuel inventory; therefore, no restrictions were needed on the amount of fuel loaded into the containers. However, efforts made to limit the amount of fuel entering the containers included limiting fuel rod end stubs to about 2 inches and limiting debris to structural materials with no significant quantities (i.e., chunks or agglomerations) of unidentifiable material attached. Therefore, the containers were expected to contain no significant quantities of fuel. Additionally, to ensure that the k_{eff} of the defueling canisters located in the storage racks remained below the licensing criteria, debris containers would be segregated from defueling canisters in the SFP-A storage racks by at least one space in all directions.

- ***NRC Review***. (266) Editor's Note: The NRC's safety evaluation report did not identify any specific safety considerations. The NRC's review concurred with the licensee's assessment that the safety consequences of the proposed activity were bounded by the previously approved technical evaluation report for the defueling canister and safety evaluation report for early defueling. However, the NRC required that procedures for the use of the debris container include restrictions such as the efforts to limit the amount of fuel entering the containers, as discussed in the licensee's letter (267) on the use of debris containers.

3.5.3.4 Fuel Canister Storage Racks

- ***Purpose***. To provide storage for the three different types of canisters (fuel, filter, and knockout) filled with debris material from the reactor vessel. Storage for 263 canisters was available in the racks located in spent fuel pool "A" and in the deep end of the fuel transfer canal.

- ***Evaluation: Criticality***. (268) The licensee's safety evaluation stated that the scope of its evaluation included the design of the fuel canister storage racks and the activities associated with the use of the racks during defueling. The criticality analysis was discussed in the licensee's technical evaluation report (TER) (269) for defueling canisters. The fuel canister storage racks were designed to withstand the impact energy of a postulated defueling canister drop. The evaluation determined that damage caused by an accidental drop of a canister would be local and would not reduce the spacing between canisters to less than the 17.3 inches used in the criticality analyses for the defueling canisters. The dropping of the canister transfer shield was not considered a credible event as the shield was an integral part of the canister handling bridge.

- ***NRC Review: Criticality***. (270) The NRC's safety evaluation stated that the fuel canister storage racks were designed to maintain the loaded fuel canisters in a geometrically safe subcritical array. The specifics of the canister criticality analysis were not included in the evaluation of the storage racks but were evaluated during a concurrent review of the licensee's TER for the defueling canisters (to be issued later). For the purposes of the storage rack evaluation, the NRC assumed that maintaining a minimum center-to-center spacing of 18 inches between loaded fuel canisters was adequate to ensure a subcritical array. This assumption would be reevaluated in the concurrent NRC review [(271)] of the fuel canister TER.

The canister storage racks were designed so that the minimum center-to-center spacing of adjacent fuel canisters was 18 inches. The racks were designed structurally so that a postulated earthquake or postulated load drop on the rack would not compromise the 18-inch spacing. The design also prevented inadvertent insertion of a fuel canister into any location not intended as a storage location. If a canister was dropped from the canister transfer system, such that the canister fell vertically alongside the rack modules and leaned against the side of a storage cell, the center-to-center spacing between the dropped canister and an adjacent stored canister would always be greater than 13 inches. However, the three center-most storage rack cells that were adjacent to the reactor plenum storage rack (located in the deep end of the fuel transfer canal inside the containment building) were designed such that the 18-inch spacing for a dropped canister could be assured only if the reactor plenum was in place on the plenum storage stand. If for some unforeseen reason, the plenum was not placed on the storage stand, the NRC would require additional controls to ensure the required minimum spacing.

The NRC approved the installation of the fuel canister storage racks in the fuel transfer canal and the spent fuel pool "A" but not the storage of fuel material. Approval of the use of the racks for storage of fuel canisters loaded with fuel material was contingent on the NRC's approval of the criticality and load drop analysis in the fuel canister TER.

3.5.3.5 Canister Handling and Preparation for Shipment

- ***Purpose***. To transfer defueling canisters from spent fuel pool "A" (SFP-A) to the shipping cask for offsite shipment. Defueling canisters were transferred to locations within the containment building and fuel handling building using a transfer shield. The transfer of canisters to the shipping cask used a different device called a "fuel transfer cask."

- ***Evaluation: Criticality***. [(272)] The analyses presented in the licensee's criticality evaluation report for the reactor coolant system [(273)] (RCS) and its hazard evaluation report [(274)] for the potential for boron dilution in the RCS demonstrated that any fuel debris configuration would remain subcritical if the debris was in water at a boron concentration of 4350 parts per million or greater. Since SFP-A was maintained at a boron concentration of greater than 4350 parts per million, any postulated accident that resulted in a reconfiguration of the fuel debris (e.g., canister damage) would not cause criticality in SFP-A. Since each canister was transferred individually, only an accident in SFP-A could result in damage more than one canister.

The use of demineralizer water for canister decontamination during the transfer of a loaded canister into the fuel transfer canal was evaluated in the licensee's safety evaluation report [(275)] and the NRC's related safety evaluation report. [(276)] Both reports determined that such use would be within the guidelines of the licensee's hazard evaluation report for the potential of boron dilution in the RCS and would not increase the criticality potential of SFP-A.

The licensee's technical evaluation report [277] for the defueling canister demonstrated that an undamaged canister could be transferred in the fuel transfer cask (surrounded by a lead reflector) without causing the effective neutron multiplication of the canister contents to exceed 0.95.

- ***NRC Review: Criticality***. [278] The NRC's safety evaluation stated that the defueling canister design was expected to remain subcritical under all onsite conditions and, when used in combination with the shipping cask, during normal and accident transportation conditions.

  - *Load Drop*. The defueling canisters were designed and fabricated according to American Society of Mechanical Engineers Section VIII pressure vessel specifications. They were designed to withstand the effects of unrestrained drops of 6 feet 1.5 inches in air, followed by 19 feet 6 inches in water or 11 feet 7 inches in air, and still maintain fuel debris confinement in a critically safe geometry. The NRC determined that such performance would bound all postulated canister drops during handling, except for a potential drop from the fuel handling building's overhead crane in the truck bay.

  - *Canister Preparation*. The final canister weights were verified to ensure that the canisters conformed to the design limits that factored in the structural and criticality analysis and also to ensure that cask loading conformed to the requirements of the certificate of compliance [279] for the Model 125-B shipping cask.

3.5.3.6 Canister Dewatering System

- ***Purpose***. To remove and filter the water from submerged defueling canisters and to provide a transfer path to the defueling water cleanup system for processing. The dewatering system also provided the cover gas for canister shipping.

- ***Evaluation***. Editor's Note: The licensee's safety evaluation of the canister dewatering system was provided in the safety evaluation reports [280, 281] for canister handling and preparation for shipment.

- ***NRC Review***. Editor's Note: The NRC's safety evaluation of the canister dewatering system was provided in the safety evaluation reports [282, 283] for canister handling and preparation for shipment.

3.5.3.7 Use of Nonborated Water in Canister Loading Decontamination System

- ***Purpose***. To use nonborated water for canister decontamination before shipment in order to stabilize the boron concentration in the fuel transfer canal/spent fuel pool "A" (FTC/SFP-A). Boron concentration in SFP-A was increased by adding borated water and by water evaporation.

- ***Evaluation: Criticality***. [284] The licensee's safety evaluation concluded that subcriticality was ensured by: (●) establishing the boron concentration required for the recovery technical specification during the defueling process; (●) maintaining and monitoring the boron concentration and inventory levels; and (●) isolating potential deboration pathways. The boron concentration and inventory level of the FTC/SFP-A were monitored in accordance with approved procedures to ensure that recovery technical specification limits for boron concentration and inventory level were satisfied. Additionally, subcriticality was maintained by the canisters' engineered safeguards and their storage arrays.

- ***Evaluation: Boron Dilution***. [285] The licensee's safety evaluation stated that physical isolation was provided between the canister loading decontamination system (CLDS) and defueling water cleanup system (DWCS) when using nonborated water. This physical isolation consisted of cutting and capping all lines associated with the CLDS that connected to the DWCS. This isolation eliminated the possibility of nonborated water from the CLDS being inadvertently introduced into the reactor coolant system (RCS). Additionally, existing administrative controls and a physical walkdown before system operation would minimize the potential for inadvertent hose connections.

During canister decontamination with nonborated water, a localized deboration of SFP-A would occur in the vicinity of the cask-loading station. The only credible means of obtaining a critical mass of fuel in this location during canister decontamination would be the result of a canister drop, while lifting a canister from the cask-loading station into the fuel transfer cask. The results of a canister drop during cask loading were analyzed in the licensee's safety evaluation report [286] for canister handling and preparation of shipment (refer to Section 6.1 of that report). This analysis stated that if the canister and grapple were to drop while lifting a canister into the cask, the canister would fall back into the loading station canister guides and no unacceptable consequences to the equipment would occur. Furthermore, the drop height when the canister was at the top loaded position in the cask was less than the designed drop limits for the canister. Therefore, should a canister drop occur in this position, the canister would remain intact, and fuel debris and poison material would remain in a stable configuration within the canister.

- ***NRC Review: Criticality/Boron Dilution***. [287] The NRC's safety evaluation concluded that the proposed system modifications were adequate to prevent inadvertent communication of the nonborated water source with the RCS. Therefore, the probability of an RCS boron dilution transient did not increase as a result of the activity. The NRC also evaluated the potential for inadvertent criticality in SFP-A due to boron dilution. The licensee's technical evaluation report [288] for the defueling canisters included an analysis of the criticality safety of the defueling canisters. This analysis showed that defueling canisters stored in the design configuration of SFP-A storage racks would be subcritical for the worst case geometric deformation after a drop accident with a completely deborated SFP-A. The analysis also showed that a loaded canister dropped on top of another loaded canister in the storage racks would remain subcritical, provided that the water in SFP-A was borated. The analysis did not support a critically safe configuration in the case of nonborated water. Even though this presented an extremely unlikely scenario, the action required to prevent such an event is described below.

The volume of nonborated water used for canister decontamination between weekly SFP-A sampling intervals was small in comparison to the SFP-A volume; therefore, sufficient deboration of SFP-A to present a criticality hazard was not likely to occur during the weekly SFP-A sampling intervals. Localized deboration, however, was a potential concern because of the stagnant nature of SFP-A, along with the density difference between the bulk SFP-A water and the heated, nonborated water that would be used for canister decontamination. Operation of the DWCS in SFP-A at sufficient flow rates ensured a mixing of the water to preclude a local deboration transient. However, the licensee did not intend to continuously operate the DWCS during canister decontamination. Therefore, the NRC required daily sampling of SFP-A water in the vicinity of the decontamination station to verify that the boron concentration was not reduced when the DWCS was not operating in SFP-A.

3.5.4 Testing of Core Region Defueling Techniques

- ***Purpose***. To use hydraulic heavy-duty defueling tools for limited bulk defueling operations on the hard crust layer of the damaged core.

- ***Evaluation***. (289) Editor's Note: The licensee's safety evaluation report did not specifically address this topic. However, the use of the light-duty tong tool and the light-duty spade bucket was previously addressed in Revision 4 of its safety evaluation report (290) for early defueling. The safety evaluation report (291) for the use of the hydraulic impact chisel to separate fused material addressed limited use of the hydraulic impact chisel.

- ***NRC Review: Criticality/Boron Dilution***. (292) The NRC's safety evaluation stated that its previous safety evaluation report (293) for early defueling concluded that breaking up the crust would not cause an inadvertent criticality. This conclusion was based on the boron concentration maintained in the reactor coolant system, which would provide adequate margin to ensure subcriticality for any postulated fuel configuration. The manipulation of partial fuel assemblies was similarly bounded by the activities approved for early defueling. The design of the fuel canisters would prevent criticality during loading, transfer, and storage of the partial fuel assemblies or pieces from the hard crust layer of the core. The NRC determined that the conclusions of early defueling regarding the limited potential for boron dilution also applied to the proposed activities.

3.5.5 Fines/Debris Vacuum System

- ***Purpose***. To modify the fines/debris vacuum system using a knockout canister and a filter canister in series. Modifications included: (●) use of a vacuum nozzle to allow larger debris particles to be vacuumed into the knockout canisters; (●) use of mechanical probes and water jets on the end of the vacuum nozzle to loosen the packed rubble; (●) use of a larger vacuum tool to allow debris removal from the lower head; and (●) temporary use of the vacuum system without a filter canister. The safety evaluation report [294, 295] for early defueling had previously approved the initial use of the fines/debris vacuum system.

- ***Evaluation: Criticality***. [296] The licensee's safety evaluation considered a new nozzle, which would allow larger debris particles to be vacuumed into the knockout canister, with respect to the canister design criterion. The criterion required maintenance of an effective neutron multiplication (k_{eff}) no greater than 0.95 during all phases of defueling operations (i.e., loading, transfer, storage, and shipping) and for postulated canister drops. The licensee's technical evaluation report [297] for the defueling canisters stated that an optimal fuel lump size would increase k_{eff} by about 0.07 percent, which was very small relative to the margin between reported calculated k_{eff} values and the k_{eff} criterion of 0.95. Thus, the presence of larger fuel debris sizes in the knockout canister was determined to have minimal impact on the then-current criticality evaluations and so would not compromise the canister criticality design criteria.

Criticality evaluations for a dropped knockout canister were based on analytical structural deformations within the canister (later verified in actual drop tests), which were independent of debris size. The evaluation concluded that increasing debris size in the knockout canister would have no impact on the structural and criticality evaluations that had been performed for a postulated drop of a knockout canister. In addition, the evaluation also concluded that the operation of the fines/debris vacuum system with the new nozzle would not impact the particle size range in the filter canister. The full flow outlet screen in the knockout canister was designed to withstand the maximum pressure differential across the screen that could be developed by the vacuum system.

- ***Evaluation: Boron Dilution***. [298] The licensee's safety evaluation considered local deboration during the use of the modified nozzle. The water source for the water jet agitation of the debris pile was at the discharge of the submersible pump. The pump, which was located below the defueling work platform and within the internals indexing fixture, took suction from the reactor vessel water. Administrative and/or physical controls and compliance with relevant operating procedures would preclude the connection of any water source other than the discharge of the submersible pump. Proper lineup would be verified before pump operation. Therefore, the introduction of water sources other than reactor vessel water by use of the modified nozzle was precluded.

- ***NRC Review: Criticality/Boron Dilution***. [299] The NRC's safety evaluation considered the effects of vacuuming larger particles into the knockout canister on the criticality analysis and structural analysis of the canister. The structural analysis previously approved in the NRC's safety evaluation report [300] for the defueling canister was independent of the size of particles loaded into the canister, and the criticality analysis was based on fuel-pellet-size particles. However, independent calculations performed by the NRC in support of the licensee review of the Model 125-B shipping cask determined that fuel-pellet-size particles provided the optimum size for maximum reactivity. Therefore, the NRC concluded that loading of larger debris would have a negligible effect on canister reactivity, and the analysis in the NRC's safety evaluation of the defueling canisters would bound this situation.

3.5.6 Hydraulic Shredder

- ***Purpose***. To use a hydraulically powered shredder to reduce the size of fuel pins and other core debris and to facilitate the loading of fuel canisters or debris buckets.

- ***Evaluation: Boron Dilution***. (301) The licensee's safety evaluation stated that the core was kept subcritical by ensuring against a boron dilution event that could lower the boron concentration below 4350 parts per million and against a localized deboration within the core. The shredder was hydraulically powered using a working fluid that was borated to at least 4350 parts per million to preclude boron dilution. The shredder also contained a quantity of unborated lubricating oil within the gear housing. This quantity of oil would be limited to 2 gallons, which was consistent with the guidelines of the licensee's safety evaluation report (302) for foreign materials allowed in the reactor coolant system, to preclude a localized deboration event.

- ***NRC Review***. (303) Editor's Note: The NRC's safety evaluation report did not specifically address this topic.

3.5.7 Plasma Arc Torch

3.5.7.1 Use of Plasma Arc Torch to Cut Upper End Fittings

- ***Purpose***. Use of a plasma arc torch to cut the upper end fittings inside the reactor vessel to allow the fittings to be placed directly in fuel canisters for transfer from the reactor vessel to the fuel canal. This was the first use of the plasma arc torch inside the reactor vessel.

- ***Evaluation***. (304) Editor's Note: The licensee's safety evaluation report did not specifically address this topic.

- ***NRC Review***. (305) Editor's Note: The NRC's safety evaluation report did not specifically address this topic.

3.5.7.2 Use of the Plasma Arc Torch to Cut the Lower Core Support Assembly

- ***Purpose***. To use the plasma arc torch to cut the lower core support assembly (LCSA), including the flow distributor head.

- ***Evaluation: Criticality/Boron Dilution***. [306] Even though the unborated water inventory in the torch coolant system exceeded the 2-gallon limit established in the licensee's safety evaluation report (SER) [307] on the limits of foreign materials allowed in the reactor coolant system (RCS), the licensee's criticality safety evaluation concluded that the plasma arc torch could be used to cut the LCSA without causing a criticality safety concern within the reactor vessel. The evaluation described: (●) conservatisms inherent in previous criticality studies; (●) base case criticality model, assumptions, and conservatisms; (●) base case model analyses results; and (●) quantification of conservatisms.

  - *Previous Studies*. The licensee's safety evaluation summarized the two previous criticality evaluation reports [308, 309] for the RCS and the limits of foreign materials allowed in the RCS. The evaluations performed for both studies contained assumptions that were considered overly conservative when applied to the specific activity of using the plasma arc torch to dismantle the LCSA. Consequently, the reduction of some of these conservatisms was considered necessary to realistically model the conditions that would exist during the cutting of the LCSA. Justification for reducing these conservatisms is addressed below.

    – *Analysis Scope*. The evaluations completed for both studies were performed with the intent that the results would be bounding during all credible situations during the entire defueling process. No attempt was made to define assumptions for a particular defueling activity or phase. However, the scope of this evaluation (LCSA cutting) limited the use of the plasma arc torch (unless evaluated separately at a later date) to the cutting of the LCSA, including the flow distributor head. The assumptions and criticality safety models that were developed for this evaluation could be tailored to the specific activities and possible accident configurations associated with the cutting of the LCSA.

    – *Core Damage*. At the time both studies were conducted, there was limited knowledge of the spatial distribution of fuel within the reactor vessel. Subsequently, data became available from debris samplings, video inspections, and defueling records, and understanding of the accident scenario improved, which allowed a more realistic modeling of the fuel debris spatial distribution in the current criticality safety analyses.

    – *Fuel Burnup*. The previous studies took credit for fuel burnup in Batch 3 fuel only. The rationale for this assumption was the small reactivity effect that was seen when Batch 1 and 2 fuel were added to the periphery of the Batch 3 fuel. Therefore, any credit for burnup of Batch 1 and 2 fuel would essentially have a negligible effect on effective neutron multiplication (k_{eff}). This effect was encountered because previous analyses had placed the entire initial inventory of the highest enriched (Batch 3) fuel in the center of the fuel arrangement. However, with the placement of a smaller amount of Batch 3 fuel in the central fuel region, as done in the plasma arc torch analyses, the reactivity worth of the other fuel batches increased. With the higher reactivity worth of the Batch 1 and 2 fuel, the burnup worth of these fuel batches also became more important. Therefore, burnup of Batch 1 and 2 fuel was included in the plasma arc torch criticality safety analysis.

- *Base Case Criticality Model*. Considerations in the development of the base case criticality model included the following:

  - *Geometrical Considerations*. The licensee determined the most likely geometry between the unborated coolant and the debris. Geometrical considerations included the following:

    - *Damage Prevention*. If flushing was performed with the torch in the vessel, no load handling activities would be allowed in or over the reactor vessel to minimize the potential for damage to the flush system.

    - *Leakage Mixing (Reactor Coolant)*. A mixing analysis for the torch coolant system concluded that mixing would occur rapidly during an inadvertent release of torch coolant. The evaluation concluded that any inadvertent leakage of the coolant during system flushing would adequately mix with the reactor vessel's borated water; therefore, the torch coolant would not pose a criticality safety concern.

    - *Limited Fuel Debris*. Before the use of the plasma arc torch for cutting the LCSA, all significant fuel masses above the LCSA, except the fuel behind the core former plates, would be removed. However, the core former plates were expected to prevent any significant quantities of fuel behind them from falling into a region where the torch was operating. Readily accessible debris would be removed during the cutting of the various plates of the LCSA. Also, the safety features inherent in the torch design would prevent the torch from becoming embedded within the fuel debris during normal operations.

    - *Leakage Intermix (Fuel)*. Given that unborated water was less dense than the borated water in the reactor vessel water, [310] any torch coolant leakage was expected to rise rather than sink into the debris. Therefore, the evaluation assumed that water would intermix with the debris pile at or near the surface of the debris accumulations. The evaluation concluded that any substantial amount of unborated water deeply intermixing within the debris was highly unlikely.

      The most likely geometry between the unborated coolant and the debris was when unborated water formed a layer on, or slightly penetrated into, the debris bed. However, for conservatism, the analysis assumed that the entire volume of unborated water would be totally submerged within the fuel, and to maximize reactivity effects, the unborated region was placed in the center of the fuel model (i.e., the most reactive location). Given that an accurate prediction of the shape of any unborated water region was essentially impossible, the analysis assumed a spherical configuration, which minimized the ratio of surface area to volume.

  - *Geometrical Model*. Based on the above considerations, the plasma arc torch base case model was developed for the criticality analysis. The model included two concentric spheres (refer to Figure 4 of the SER). The inner smaller sphere represented the unborated water mixing with fuel debris. The size of this region was based on the volume (i.e., 3.0 gallons) of unborated water that was assumed to leak into the vessel. The outer and larger spherical fuel volume estimate was based on the balance of the initial fuel inventory, optimally moderated with the borated reactor coolant at 4950 parts per million (ppm). These two spheres were then surrounded by a thickness of borated water that represented an infinite reflector layer.

– *Fuel Enrichment and Burnup Worth*. The analysis model included two concentric spherical fuel zones (refer to Figure 4 of the SER). The inner fuel zone, which was moderated by unborated water and appropriately sized for the amount of unborated water included in the analysis, was modeled with Batch 3 fuel (2.67 weight percent enriched) and burnup effects. For the larger, outer fuel zone, the model used fuel enrichments and burnup effects that were based on the average fuel (i.e., a homogeneous mix of the three fuel enrichment batches). This average fuel was determined to be a conservative choice, based on actual enrichment data from available fuel debris samples and the following considerations: (●) Most of the Batch 3 fuel was removed from the vessel. (●) Debris in the LCSA was expected to be primarily Batch 1 and 2 fuel in relatively equal amounts. This fuel content was based on the initial loading patterns, earlier defueling activities, and the then-existing understanding of the accident scenario. (●) Average initial enrichment without burnup of Batch 1 and 2 fuel was 2.31 weight percent. (●) There was an apparent lack of Batch 1 fuel in the sample data (i.e., only one data point had an enrichment lower than the initial Batch 1 enrichment of 1.96 weight percent).

– *Lattice Structure*. As reported in previous criticality evaluation reports [311, 312] on limits of foreign materials allowed in the RCS and for the RCS, the fuel was represented as a homogeneous medium where the neutronic data corresponded to a dodecahedral lattice

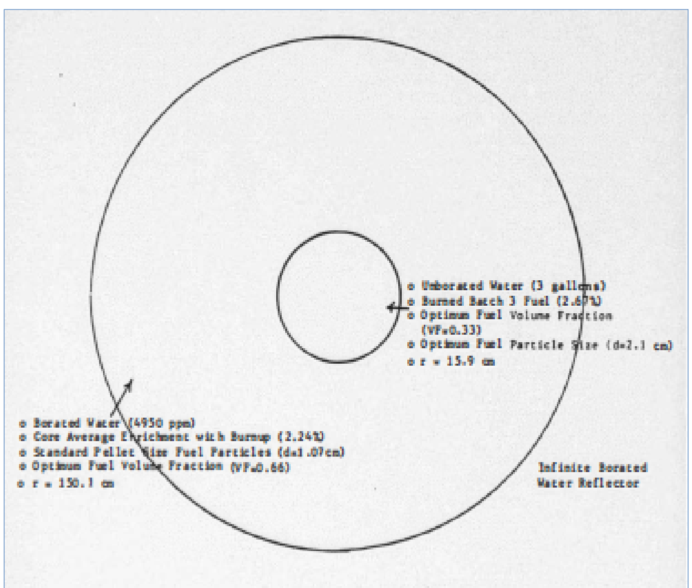


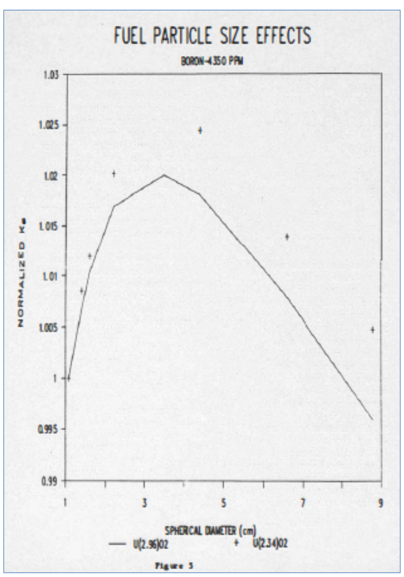
Figure 4: Base Case LCSA Criticality Safety Model

SER Figure 4. Base case LCSA criticality safety model.

structure (polyhedron with 12 flat faces) of spherically shaped fuel pellets. [aa] The previous analyses had limited the maximum size of the fuel particle to the equivalent of a standard fuel pellet with the presence of melted fuel; therefore, larger pellets were considered for the plasma arc torch analyses.

- *Optimum Fuel Particle Size (Inner Sphere)*. Based on the relatively small size of the inner sphere, the analysis assumed that the entire fuel mass that would mix with the unborated water consisted of Batch 3 fuel. According to the damage assessments at that time, fuel melting was not initiated in any Batch 3 fuel. Rather, any dissolution of Batch 3 fuel occurred as a result of melted Batch 1 and 2 fuel flowing past the Batch 3 fuel rods. Consequently, the fuel in any particles that were larger than standard pellets was highly unlikely to comprise solely Batch 3 fuel. If Batch 3 fuel was present in the large particles, it was likely to be mixed with Batch 1 and 2 fuel. Additionally, based on the available sample data, it would be unlikely that any large fuel particles could be pure uranium dioxide (UO_2). Some of the impurities present in the sampled debris were quantified, such as fuel cladding, control rod, burnable neutron poison, and structural material (refer to Table 1 of the SER).

  The effects of these impurities were evaluated in the SER (see below). However, an optimum (i.e., most reactive) fuel particle size of pure UO_2 was used to represent the fuel within this region to maximize the reactivity effects of fuel melting. The optimum fuel particle size was determined by performing an extensive series of lattice cell calculations. These calculations varied particle size and fuel volume fraction until a most reactive particle size and volume fraction combination was found. The



SER Figure 5. K_{eff} versus fuel particle size for two fuel enrichments.

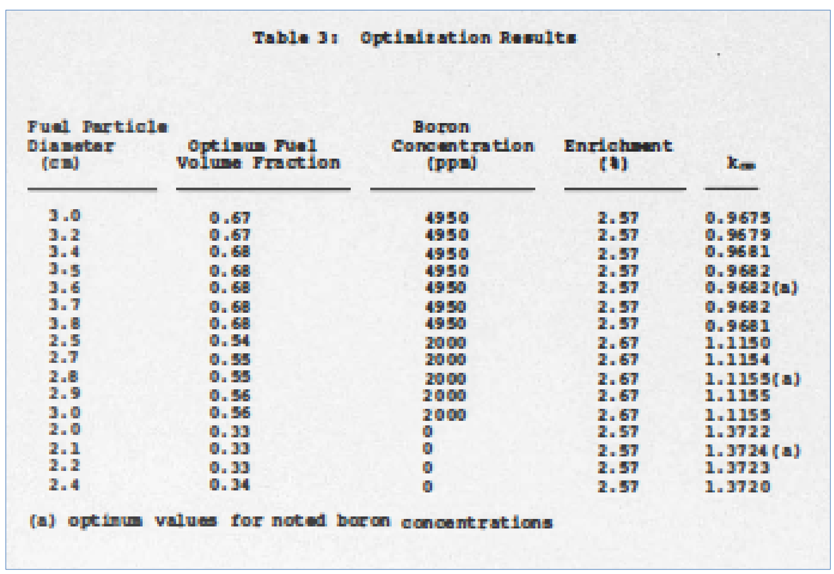


Table 3: Optimization Results

Fuel Particle Diameter (cm)	Optimum Fuel Volume Fraction	Boron Concentration (ppm)	Enrichment (%)	k_{∞}
3.0	0.67	4950	2.57	0.9675
3.2	0.67	4950	2.57	0.9679
3.4	0.68	4950	2.57	0.9681
3.5	0.68	4950	2.57	0.9682
3.6	0.68	4950	2.57	0.9682(a)
3.7	0.68	4950	2.57	0.9682
3.8	0.68	4950	2.57	0.9681
2.5	0.54	2000	2.67	1.1150
2.7	0.55	2000	2.67	1.1154
2.8	0.55	2000	2.67	1.1155(a)
2.9	0.56	2000	2.67	1.1155
3.0	0.56	2000	2.67	1.1155
2.0	0.33	0	2.57	1.3722
2.1	0.33	0	2.57	1.3724(a)
2.2	0.33	0	2.57	1.3723
2.4	0.34	0	2.57	1.3720

(a) optimum values for noted boron concentrations

SER Table 3. K_{eff} versus optimum fuel particle sizes.

[aa] Editor's Note: More recent versions of computer codes for criticality safety analyses include enhanced geometric and physics capabilities that can be used for analyzing the concerns described in this report. Furthermore, these codes include updated cross-section libraries.

dodecahedral unit cell of the previous criticality analyses (spherical fuel particles surrounded by water) was used for these calculations. The 27-group END/B-IV cross-section library was applied in the SCALE computer code [(313)] to provide resonance-shielded (NITAWL-S module) and cell-weighted (XSDRNPM-S) cross sections. The optimum particle size was determined to have a diameter of 2.1 centimeters for the unborated region.

- *Optimum Fuel Particle Size (Outer Sphere)*. Similarly, a series of lattice cell calculations was performed to determine the optimum fuel volume fraction for the core average fuel mixed with borated water at 4950 ppm. However, the use of an optimum particle size for the outer fuel zone was considered unnecessarily conservative for the plasma arc torch analyses. This conclusion was based in part on the following considerations: (●) Core damage assessments indicated that a large percentage (greater than 60 percent) of the debris in the LCSA/lower head was either fines (less than pellet size) or large fused masses (greater than about 20 centimeters diameter). (●) It was unlikely that any melted fuel particles would be pure UO_2 as was assumed in the optimization calculations. (●) Most of the Batch 3 fuel would be removed from the vessel before plasma torch usage; this fuel region would consist mostly of Batch 1 and 2 fuel. (●) The range of pure UO_2 fuel particle size was somewhat narrow where infinite neutron multiplication ($k_{infinity}$) value exceeded the size of standard pellets.

  The last consideration was demonstrated by the data presented in Figure 5 of the SER. This figure provided the relative relationship between $k_{infinity}$ and fuel particle size for two different fuel enrichments (2.96 percent and 2.34 percent). [(314)] Although the boron concentration used to develop the data for Figure 5 was 4350 ppm, the general conclusions derived from this curve would not change the boron concentration of interest in this analysis (i.e., 4950 ppm). Backup for this assumption was provided by the 3.6-centimeter diameter optimum fuel particle size shown in Table 3 of the SER with a 4950-ppm boron concentration.

- *Optimal Fuel Volume Fractions*. An optimal fuel volume fraction was used for each of the different particle size calculations shown in Figure 5 of the SER. Pure UO_2 particles were assumed for the analysis. The $k_{infinity}$ values presented in the figure were normalized to the $k_{infinity}$ value at the spherical diameter corresponding to standard pellets (1.07 centimeters). This normalization was performed for the two enrichments analyzed. A review of the figure showed that optimally moderated particles with diameters in the narrow range of greater than the equivalent of standard pellets to less than about 10 centimeters would have a $k_{infinity}$ value that exceeded the $k_{infinity}$ value for standard pellets. Consideration of the presence of impurities in the melted fuel, along with the use of actual fuel volume fractions, would result in decreased values of $k_{infinity}$ for the melted fuel.

- *Conclusion*. The evaluation concluded that the use of fuel particles of a size corresponding to the equivalent of standard pellets would be an appropriately conservative representation of fuel in the outer fuel zone.

– *Unborated Coolant Volume*. The maximum unborated coolant inventory in the plasma arc torch cooling system was, by design, less than 4 gallons. However, based on the physical characteristics of the coolant system (e.g., the system was vented to the atmosphere), the draining of the entire inventory following a line break or a torch tip blowout was determined to be hydraulically impossible when the torch was operating in the reactor vessel.

To evaluate the maximum amount of unborated coolant leakage that would occur during torch operation, a draindown test was performed. In this test, the system pump was permitted to operate throughout the duration of the test. In reality, a float switch (disabled in the test) would shut off the pump at a low inventory level. The measured leakage from the test was about 3.45 gallons. As the test was performed with the hoses in open air (to assist in measuring leakage quantity), this volume was reduced by the amount of the coolant inventory that would not drain since the torch would actually be in the reactor vessel (about 0.47 gallons).

The evaluation concluded that the maximum amount of unborated water that would drain from the torch coolant system during torch operations was limited to less than 3 gallons. This volume (3 gallons) was used as the volume of unborated water in the base case model.

– *Conservatisms*. In the development of the plasma arc torch base case criticality safety model, the following conservative assumptions were used: (●) no credit for presence of steel plates in LCSA; (●) no credit for large amounts of structural or solid poison materials existing in debris (refer to Table 1 of the SER for a list); (●) optimized fuel particle size in unborated fuel region; (●) optimized fuel/moderator ratio in all fuel regions; (●) no credit for mixing of unborated cooling water with borated vessel water; (●) minimum allowable boron concentration of 4950 ppm assumed in borated regions of the model; and (●) unborated water region placed in most reactive configuration (center of fuel model).

The SER (see below) provided quantification of the reactivity worth of some of these conservatisms. The evaluation determined that isolated regions within the debris bed could have greater average enrichments or more reactive particle sizes than those used in the large sphere of the base case model. However, considering the base case model as a whole, including the inherent conservatisms outlined above, the evaluation concluded that the base case model was a conservative representation of any credible configuration that could be experienced while using the plasma arc torch to dismantle the LCSA. Therefore, the base case model was appropriate for use in this evaluation.

- *Results (Base Case)*. The licensee's safety evaluation provided the following results of its criticality analyses:

  – *Optimization Results*. An extensive series of calculations was performed to determine the optimum fuel particle size and corresponding optimum fuel volume fraction for the various boron concentrations of interest. Table 3 of the SER gives the results of these calculations. Preliminary investigations found that the optimum size and volume fraction were mainly a function of boron concentration; therefore, a change in enrichment had little effect on these parameters. Optimization was not performed at every combination of enrichment and boron concentration but rather at one enrichment for each boron concentration of interest. Optimum ($k_{infinity}$) values at noted boron concentrations from Table 3 were $k_{infinity}$ of 0.9682 at 4950 ppm, $k_{infinity}$ of 1.1155 at 2000 ppm, and $k_{infinity}$ of 1.3724 at zero ppm.

  – *Base Case Results*. Using XSDRNPM to analyze the base case model, k_{eff} was determined to be 0.9582. Typically, XSDRNPM analyses were performed to add confidence to the values predicted using the KENO code. Generally, the results predicted using the two codes for TMI-2 criticality safety analyses agreed well. However, the first KENO run performed using the base case model predicted k_{eff} to be 0.9663 ± 0.0010. The agreement between the results was not as good as that experienced in previous criticality safety analyses for TMI-2. Because of this difference, the analysis was further investigated. The investigation concluded that the difference was most likely due to the statistical nature of KENO (amplified by the presence of an unborated central region in the model geometry). To confirm this conclusion, nine additional KENO runs were made with a k_{eff} of 0.9599 ± 0.0011 being the mean value of the 10 runs (refer to Tables 4 and 5 of the SER).

    Neither the KENO nor the XSDRNPM results stated above included an analytical uncertainty bias. Applying a 2.5-percent delta-k bias, the KENO result for k_{eff} was 0.9849 and considered to be the base case result. This result met the acceptance criterion for a k_{eff} not exceeding 0.99 for all credible situations during torch usage.

- *Quantification of Conservatisms*. To quantify the effects on k_{eff} of some of the conservatisms inherent in the base case model, as described above, additional analyses were performed and presented in the SER. These analyses were provided to demonstrate that there was a large degree of conservatism in the base case model. They included: (●) extent of a local boron dilution (mixing) resulting from a postulated break in the plasma arc torch's cooling hoses or from a blown torch tip (rather than a local boron displacement as assumed in the criticality evaluation report [315] on limits of foreign materials allowed in the RCS during defueling activities); (●) effects of significant amounts of stainless steel on the neutron multiplication; and (●) reduction in reactivity due to the presence of impurities in the melted fuel.

  – *Hydraulic Mixing*. An evaluation was performed to determine the extent of a local boron dilution (rather than a local boron displacement resulting from a postulated break in the plasma arc torch cooling hoses or from a blown torch tip) as assumed in the licensee's criticality evaluation report for the limits of foreign materials allowed in the RCS. Considerations included the following:

- *Mixing Rate*. The entrainment of the unborated cooling water was calculated using empirical correlations for the mixing of turbulent water jets into large quiescent water systems. [316] No credit was taken for the other mixing mechanisms (e.g., turbulence created by the torch operation, the gas purge, interaction with debris, or the normal vessel convection currents). Based on correlations [317] that showed mixing was essentially a function of the break area, an analysis was performed to determine the average boron concentration of the fluid entrained in the jet at various distances from the postulated cooling line break, as a function of the assumed break area (refer to Figure 6 of the SER). This analysis was performed for two types of breaks, a circular break with both hose ends discharging and a slot break. The analysis results showed that mixing occurred less quickly with larger break areas. Thus, the mixing rates for the maximum break areas were used to incorporate the mixing phenomena into a criticality safety model.

- *Break Size*. To assess the effect of mixing on k_{eff}, the analysis assumed that the boron concentration would be defined using two mixing regions. The limiting boron concentration between the two mixing regions was also arbitrarily selected to be 2000 ppm. Based on the above, a mixing analysis determined the distance and the associated water volume when the average boron concentration increased to a level above 2000 ppm for a full area guillotine break of the coolant hose and a 1.0-square-inch slot break. This was the first step in the process to include mixing in the criticality safety model. The 1.0-square-inch slot break was considered to be the maximum credible size considering the hose used, the planned operating procedures, and fluid conditions existing within the hose. The full area break bounded all other credible breaks including a torch tip blowout. The arbitrary selection of the 2000-ppm boron level was appropriate since the sole purpose of the analysis was to demonstrate that mixing would occur rapidly and that there was a larger degree of conservatism associated with neglecting the effects of mixing in the base case model.

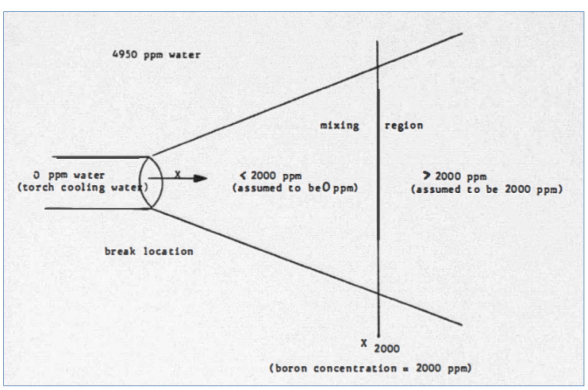
- *Geometrical Model*. In both of the scenarios considered, mixing occurred very rapidly with the volume of water (borated and unborated) entrained in the jet cone being less than 0.25 gallon before the 2000 ppm concentration was reached (refer to Figure 6 of the SER). The criticality safety model to assess reactivity effects of mixing included three concentric spherical zones (refer to Figure 7 of the SER). The innermost fuel zone was assumed to be 0.25 gallon of unborated (0 ppm) water. This water was then optimally mixed with optimally sized Batch 3 fuel particles. Next, the middle fuel zone containing 4.61 gallons of 2000-ppm borated water was optimally mixed with the optimal Batch 3 fuel particles. The 4.61 gallons was used to simulate the mixing of the additional 2.75 gallons of unborated coolant with 1.86 gallons of borated (4950 ppm) vessel water. The outside fuel zone was the remaining balance of the full core fuel inventory, optimally mixed with the burned core average fuel described above. Finally, an infinite borated water reflector was placed external to the fuel regions.

- *Conservatisms*. Conservatisms inherent in this hydraulic mixing model included the following: (●) The effect caused by unborated water being less dense than borated water (tending to rise rather than sink into the fuel) was neglected. (●) All water in the jet with boron concentrations less than 2000 ppm was assumed to be unborated. (●) All water in the jet with boron concentrations greater than 2000 ppm was assumed to have a 2000-ppm concentration. (●) Significant mixing mechanisms were neglected (e.g., turbulence created by torch operation, gas purge, interaction with debris and normal vessel convection currents). (●) Unborated water was placed in the center (highly reactive) location of the model.

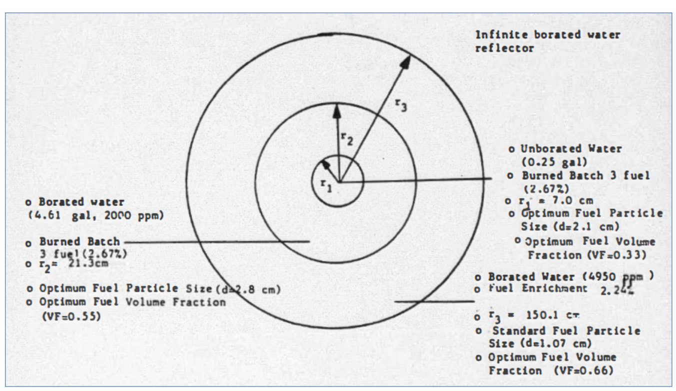
  A more elaborate criticality safety model could have been used that included more fuel regions and a finer boron concentration distribution. This model would result in an even smaller calculated k_{eff} value.

– *Effects of Stainless Steel*. Stainless steel occupied much of the volume within the LCSA region of the reactor vessel. All steel was conservatively neglected in the development of the base case model. The largest piece of steel within the LCSA, the grid forging, was used as the basis for a model developed to assess the reactivity worth of this stainless steel. The grid forging was a 13.5-inch-thick steel plate drilled with about 6.5-inch-diameter holes in a lattice with an edge-to-edge separation of 2.1 inches (refer to Figure 8 of the SER).

  A model of the grid forging was developed to perform the stainless-steel sensitivity analysis; however, each of the holes was assumed to be only 6 inches in diameter with a separation of 2 inches. Additionally, the size of the grid forging was assumed to be infinite in the radial direction and 14 inches high axially. Each hole was assumed to be filled with an optimum mixture of unborated water and fuel. The fuel used in this case was optimum-sized fuel particles, with an enrichment of 2.3 percent. On the top and bottom of the steel, there was an infinite thickness of borated water reflector.



SER Figure 6. Hydraulic geometrical mixing model.



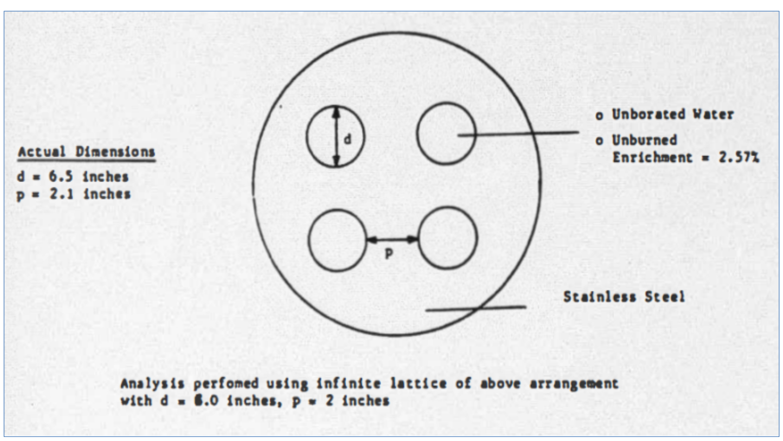
SER Figure 7. Criticality safety model to assess effects of mixing.

Dimensions used in this criticality safety analysis differed slightly from the actual grid forging dimensions.

However, based on the extremely low values of k_{eff} seen for this analysis, the effects of these differences would not affect the overall conclusion that the presence of significant amounts of stainless steel had a negative effect on the neutron multiplication.

– *Effects of Impurities*. To assess the reduction in reactivity due to the presence of impurities in the melted fuel, another series of lattice cell calculations was performed. In these calculations, the average impurities identified in Table 1 of the SER (such as fuel cladding, control rods, and structural materials) were assumed to be mixed with optimally sized, burned Batch 3 fuel. Unborated water was used as the moderating material. The mixture particle size and fuel volume fraction were varied until a maximum ($k_{infinity}$) value was determined. This analysis did not consider the potential depletion of neutron poisons (e.g., boron-10).

– *Results*. Table 4 of the SER gives the results for these additional cases. The main conclusion to be drawn from this table was that the base case model was very conservative. For example, the results of the mixing model showed a nominal k_{eff} of 0.924 ± 0.001. This corresponded to about a 3.6-percent delta-k reduction from the base case analysis. Additionally, by virtue of the extremely low calculated value of k_{eff}, the results of the stainless-steel model indicated that there was a large negative effect on k_{eff} when credit was taken for the significant quantities of stainless steel present in the LCSA. The effects of impurities of the melted fuel could be seen by a comparison of the optimum ($k_{infinity}$) value for pure UO_2 and the optimum value considering the impurities. The value considering the impurities was less than 0.8, while the pure UO_2 ($k_{infinity}$) was 1.37. The negative reactivity effect of the impurities would be decreased if the potential depletion of the neutron poisons was considered. Additionally, the relative worth of the impurities would decrease if borated water was assumed to be the moderating material. Nevertheless, the calculated difference in the ($k_{infinity}$) values demonstrated the conservatism associated with neglecting the presence of impurities in the melted fuel. In



SER Figure 8. Criticality safety model considering presence of stainless steel within the LCSA.

conclusion, the results of these additional analyses demonstrated that there was a large degree of conservatism in the base case model.

- *Conclusion*. Based on the evaluation presented in the SER, the licensee concluded that the plasma arc torch, with a maximum coolant system inventory of 4 gallons of unborated water, could be used to dismantle the LCSA and elliptical flow distributor head without developing a criticality safety concern within the reactor vessel.

- *Operational Limitations*. The licensee's conclusion was based on the following operational limitations stated in its SER: (●) The plasma arc torch would be used only to cut the LCSA. (●) All standing fuel assemblies would be removed from the core region before use of the plasma arc torch in the reactor vessel. (●) A maximum of 4 gallons of unborated water was permitted in the plasma arc torch coolant system with a system configuration such that a maximum of 3 gallons could drain following a line rupture or torch tip blowout with the torch operating in the reactor vessel. (●) Following the loss-of-coolant inventory, the torch would be removed and repaired before refilling the torch cooling system. (●) During in-vessel flushing of the torch, no load handling operations (heavy or light) would be permitted in or above the reactor vessel. (●) Flushing of the plasma arc torch coolant system with the torch within the vessel could occur only under certain restrictions. These restrictions included no known leaks in the coolant system; the torch being in its home position; at least a 1-foot separation between the torch tip and significant debris quantities; and the gas purge operating. Otherwise, the torch had to be removed from the vessel before connection of the flushing tie-in. (●) The maximum inventory of unborated water permitted in the flush system storage tank was 15 gallons.

- ***Evaluation: Criticality (Revision)***. [318] The licensee revised its safety evaluation to remove the operational restriction relating to the removal of all standing fuel assemblies before the use of the plasma arc torch. The licensee's revised safety evaluation stated that all standing fuel assemblies were removed from the reactor vessel with one exception at the R-6 grid location. Because of the unique condition of the R-6 assembly, the licensee believed that the original restriction was satisfied to the best of its ability and would not need to be imposed for the remaining fuel material at R-6.

- *Background*. The reason for the restriction was based on modeling assumptions used in the original criticality safety analysis. In the original analysis, a homogeneous mixture (i.e., core average fuel) of all three fuel batches was assumed to surround the unborated Batch 3 fuel region of the model. Only a limited quantity of segregated Batch 3 fuel was included in the model. The available core debris data and planned plasma arc torch usage were reviewed to evaluate whether a possible fuel configuration could develop that was not bounded by the modeled geometry. The evaluation concluded that the most likely scenario where such a configuration could occur would require the presence of several standing Batch 3 fuel assemblies. If a leak of the unborated cooling water were to occur within the standing Batch 3 assemblies or if these assemblies were to fall into an area where the torch was operating, a large pocket of Batch 3 fuel could potentially develop. Therefore, the requirement to remove all standing fuel assemblies was imposed to ensure that the torch cooling water would not leak into a region containing a substantial quantity of pure Batch 3 fuel.

- *Evaluation*. Considering the rationale for the restriction, the licensee concluded that this restriction would not be imposed for the material remaining at location R-6. This conclusion was based on the following: (●) A video survey of material in the R-6 position indicated the presence of resolidified molten material not resembling original R-6 Batch 3 fuel. Although no samples of the R-6 material were analyzed, analyses of resolidified material throughout the debris bed have yielded average enrichments consistent with the core average. (●) It was unlikely that the material at R-6 was pure UO_2. Analyses in the original evaluation showed that a significant reduction in neutron multiplication could occur if the presence of impurities was considered. (●) Fuel mass at location R-6 was isolated from the majority of the fuel remaining in the vessel and was in a geometry that did not pose a criticality safety concern. (●) The amount of material at position R-6 was limited (measuring about 8.5 inches by 8.5 inches by 18 inches) and was not expected to fall into the areas where the torch would be operating.

- *Conclusion*. Based on the above arguments, potential relocation of the fuel would not be a criticality safety concern. The licensee's safety reevaluation concluded that the above analysis did not challenge the validity of the safety analyses in the original safety evaluation report.

- ***NRC Review***. (319) Editor's Note: NRC reviews of both licensee safety evaluations were addressed under the NRC's SER (320) for the defueling of the LCSA (refer to that SER).

3.5.7.3 Use of Plasma Arc Torch to Cut Baffle Plates and Core Support Shield

- ***Purpose***. To use the plasma arc torch to cut the upper core support assembly (UCSA) baffle plates and the core support shield and to increase the maximum allowable drainable volume for the plasma arc torch coolant system from 3.0 to 3.5 gallons. Cutting the baffle plates was required to gain access to the core debris located behind the baffle plates in the core formers.

- ***Evaluation: Criticality/Boron Dilution***. (321, 322) The licensee's criticality safety evaluation considered the use of an additional 0.5 gallon of cooling water used in the plasma arc torch cooling system over the 3.0 gallons that was considered by the licensee's previous safety evaluation report (SER) (323) for the use of the plasma arc torch to cut the lower core support assembly (LCSA).

- *Introduction*. The licensee's safety evaluation considered criticality safety during use of the plasma arc torch to cut the baffle plates in the UCSA to gain access to the core debris resting behind the baffle plates on the core formers. In this evaluation, the unborated cooling water from the plasma arc torch was assumed to leak into the core debris located behind the baffle plates. When the plasma torch was positioned to cut at or near the top of the baffle plates, the length of the cooling water supply hose immersed in the reactor vessel water would be less than that assumed in the previous SER for LCSA cutting.

Consequently, the amount of cooling water that could drain into the vessel was potentially greater than the 3.0 gallons used in the analyses in the previous SER for LCSA cutting. The volume of unborated water available to leak from the plasma arc coolant system was a key parameter of the criticality safety analysis; increased leakage resulted in an increased effective neutron multiplication (k_{eff}), provided that all other modeling parameters remained unchanged. Therefore, it was important to ensure that the amount of leakage used in this analysis (i.e., 3.5 gallons) bounded all potential operating conditions. To provide an upper bound on the leakage, the analysis conservatively assumed that none of the cooling systems hose inventory would be immersed during baffle plate cutting. The following analysis demonstrates that the maximum drainable volume for the plasma torch system could be increased to 3.5 gallons for baffle plate cutting without posing a criticality safety concern.

The criticality analysis performed to support this SER was very similar to the analysis reported in the previous SER for LCSA cutting. Because of this similarity, the SER presented only a brief descriptive summary of the criticality analysis performed for use of the plasma torch in the UCSA. The previous SER provided a more detailed description of the plasma torch system, along with background information that described the basic logic employed to assess the criticality safety implications of unborated water entering the reactor vessel. The previous SER also provided a more detailed technical discussion of the analytical approach used to perform the criticality safety analysis.

- *Criticality Model*. The model that was used in the criticality analysis for this scenario involved two concentric right circular cylinders and an infinite borated water reflector at the bases of the cylinders (refer to Figure 4 of the SER).

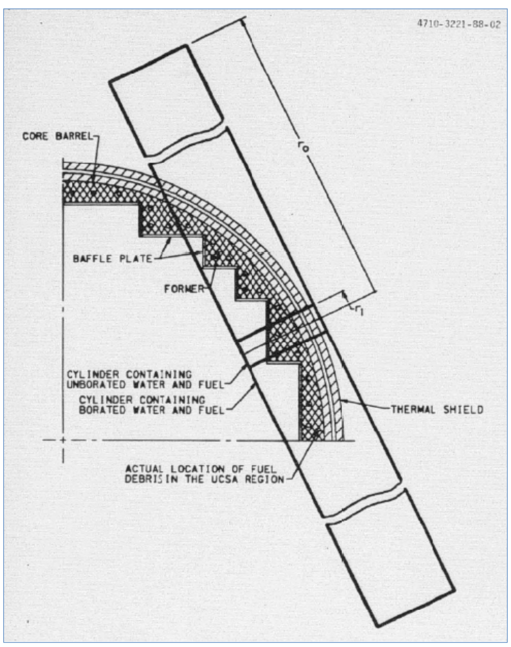
  - *Inner Cylinder*. The drained unborated water was assumed to mix with fuel in the inner cylinder of the model. The volume of this inner cylinder (1122 cubic inches) was determined by combining the 3.5 gallons of unborated water with an optimum quantity of fuel with a fuel volume fraction of 0.28. The height and radius of this cylinder varied, while keeping the volume constant, until a maximum k_{eff} was determined. This approach neglected the physical constraints imposed by the core barrel and baffle plates.

  - *Outer Cylinder*. The inner fuel cylinder, surrounded by another cylinder of the same height, contained a mixture of fuel and 4950 parts per million (ppm) borated water. This region was used to represent the fuel remaining behind the baffle plates that did not become mixed with the unborated water. To avoid imposing limits on the quantity of fuel that could remain behind the baffle plates, the radius of the outer fuel region was set to a large value (150 inches). The top and bottom of these two cylinders were covered with a 4950-ppm borated water reflector. Figure 3 of the SER provided a view of the fuel cylinders superimposed onto the region of the vessel where the unborated water was assumed to leak. The dimensions of the cylinder in this figure corresponded to the dimensions resulting in the maximum value of k_{eff}.

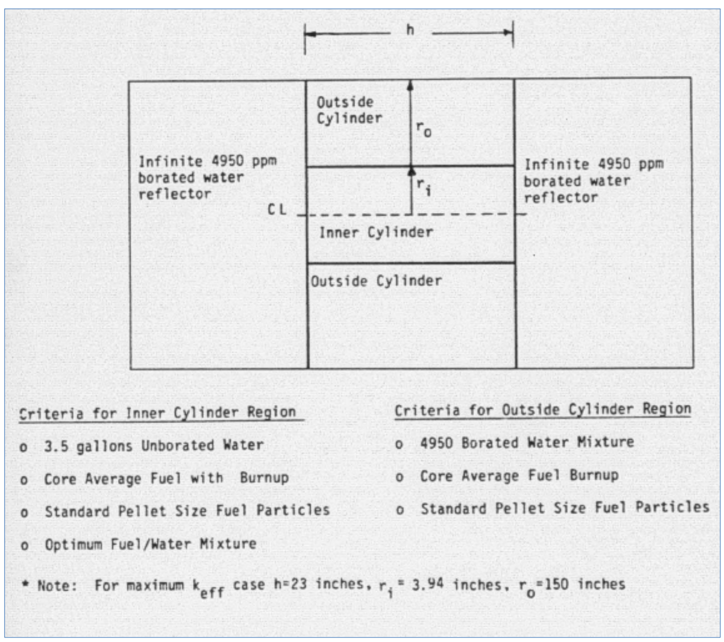
– *Reflectors*. The analysis employed a cylindrical model, which facilitated the use of more realistic reflective boundary conditions. A borated water reflector, used to represent the large reactor vessel water inventory, could be applied on the ends of the fuel cylinders, while a borated water/fuel mixture, representing the remaining fuel behind the baffle plates, could be placed outside the curved surfaces of the unborated cylinder.

– *Fuel Enrichments and Lattice Structure*. As with the previous SER for LCSA cutting, the fuel was represented as a homogeneous medium where the neutronic data corresponded to a dodecahedral lattice structure of spherically shaped fuel pellets. The composition of the fuel corresponded to the TMI-2 core average fuel (i.e., homogeneous mix of the three fuel enrichments). Incorporation of burnup effects resulted in a net uranium-235 enrichment of 2.24 percent. The equivalent of standard size fuel pellets was used as the particle size in all fuel regions. Additionally, no credit was taken for any impurities that existed in the fuel debris (e.g., cladding, control rod, and structural materials).

  This model was appropriate because the region between the baffle plates and core barrel formed an irregular annulus. This annular-like region with its large diameter could be represented by an essentially infinite slab because the sides were neutronically isolated. Further, a large-diameter cylinder in the same plane could be used in lieu of a slab to simplify the model, along with an inner cylinder to represent the unborated water/fuel mixture.



SER Figure 3. Criticality safety model for the core former and baffle plates. (See SER Figure 4 for a simplified version.)



SER Figure 4. Simplified criticality safety model for the core former and baffle plates. (See the centerline cut for SER Figure 3.)

The localized regions of fuel debris behind the baffle plates could have an average enrichment that was greater than the enrichment used in the criticality safety model. However, based on the current defueling data and the mechanisms that could have transported the fuel to a location behind the baffle plates, the analysis concluded that no significant agglomerations of Batch 3 fuel were credible in the annular region behind the baffle plates. Consequently, it was concluded that the use of an average enrichment was appropriate for this evaluation.

- *Conservatisms*. In the development of this criticality safety model, the following conservative assumptions were used: (●) no credit for large amounts of structural or solid poison materials existing in the debris; (●) optimized fuel/moderator ratio in all fuel regions; (●) no credit for mixing of unborated cooling water with borated vessel water; (●) height of the cylindrical fuel model was varied until a maximum neutron multiplication was determined, thus neglecting the physical constraints imposed by the core barrel and the baffle plates; (●) minimum allowable boron concentration of 4950 ppm in borated regions of model; and (●) unborated water region in the center of the fuel model.

  These conservatisms ensured that the geometry model used for the evaluation bounded credible geometries, including the distortion of the baffle plates that could exist during the cutting of the baffle plates with the plasma arc torch. Therefore, the use of this model was considered appropriate for the criticality safety evaluation to assess usage of the plasma arc torch to dismantle the UCSA.

- *Results*. Table 1 of the SER provided the results of the criticality safety analysis completed by Oak Ridge National Laboratory. The maximum calculated neutron multiplication that included an uncertainty bias of 2.54 delta-k was 0.928. This value of k_{eff} occurred with an inner fuel cylinder height of 23.0 inches. A cylinder of this size could not fit (i.e., it was too large) in the region where the unborated water was assumed to leak.

- *Conclusion*. The licensee's safety evaluation concluded that 0.928 was a conservative value for the neutron multiplication as a result of the unborated water in-leakage that could be postulated to occur during the cutting of the baffle plates with the plasma arc torch. This k_{eff} was significantly less than the licensing basis of k_{eff} no greater than 0.99, and the licensee concluded that the plasma arc torch could be used to cut the baffle plates without presenting a criticality safety concern.

- *Operational Limitations*. The above conclusion was based on the following operational limitation and the applicable limitations in previous SERs [324, 325, 326] for LCSA cutting: (●) System configuration was such that a maximum of 3.5 gallons could drain following a line rupture or torch tip blowout while the torch was operating in the reactor vessel. (●) Following the loss-of-coolant inventory, the torch would be removed and repaired before refilling the torch cooling system. (●) During in-vessel flushing of the torch, no load handling operations (heavy or light) were permitted in or above the reactor vessel. (●) Flushing of the plasma arc torch coolant system with the torch within the vessel could occur only if there were no known leaks in the coolant system and the torch was at least 1 foot away from the baffle plates or core formers. Otherwise, the torch must be removed from the vessel before connection of the flushing tie-in. (●) The maximum inventory of unborated water permitted in the flush system storage tank was 15 gallons. (●) The operating procedure "Automated Cutting Equipment System Operation" included a signed verification by the on-duty fuel handling senior reactor operator that the 15-gallon tank was disconnected from the HE-200 unit before system operation and before filling the 15-gallon tank. (●) The plasma arc torch would be positioned more than 1 foot away from fuel-bearing areas, external to the region between the baffle plates and core barrel, which contained greater than or equal to 10 kilograms of fuel. This restriction did not apply to fuel-bearing areas in the LCSA/lower head region (e.g., fuel assembly R-6), which was bounded by the criticality safety assessment in a previous SER for LCSA cutting.

- ***NRC Review: Criticality/Boron Dilution***. [327] The NRC's safety evaluation noted that the request for an allowable potential increase in torch coolant (nonborated water) leakage was due to an increase in the amount of the coolant hose above the water line. The licensee's analysis assumed that the entire hose was out of the water and could drain, because of gravity, into an underwater fuel-bearing area. The geometry where fuel and nonborated water could collect was much more restrictive in the area behind the core baffle plates than the geometry for the previously considered LCSA. The core baffle plates and core barrel allowed fuel to collect in a relatively narrow annular region versus the large lenticular hemisphere considered for the LCSA criticality evaluation.

The NRC noted that the licensee's analysis was well conceived, conservative, and acceptable to the agency. The principal conservatisms were in the geometry, fuel-to-water ratio, and neglect of all of the diluents and poisons and mixing of borated with nonborated water. The NRC had previously determined that defueling activities that could result in core alterations would have a k_{eff} of less than 0.99. The resultant k_{eff} for this case (i.e., the analysis of the cutting of the UCSA components) was less than 0.93, including the uncertainty factor. The licensee's methods to ensure that the potential leakage of nonborated water was limited to less than 3.5 gallons were acceptable.

The NRC concluded that the proposed activities could be accomplished without significant risk, provided that they were conducted in accordance with the limitations stated in the licensee's submittals and the limitations for this safety evaluation.

3.5.7.4 Use of Air as Secondary Gas for Plasma Arc Torch (NA)

3.5.8 Use of Core Bore Machine for Dismantling Lower Core Support Assembly

- ***Purpose***. To use the core bore machine, in conjunction with the automatic cutting equipment system, to dismantle the lower core support assembly and facilitate defueling by providing access to the reactor vessel lower head.

- ***Evaluation: Criticality***. [328] The licensee's safety evaluation stated that safety concerns related to this operation, such as the potential releases of radioactive material, criticality within the reactor vessel, and the potential for a pyrophoric event, were previously addressed in the safety evaluation report [329] for the use of the core bore during core stratification sample acquisition. The consequences of these issues did not differ for the use of the core bore machine on the lower core support assembly.

- ***NRC Review: Criticality***. [330] The NRC's safety evaluation stated that safety issues associated with pyrophoricity, criticality, and the effects of mechanical forces on the reactor vessel and internals would not significantly differ from those reviewed and evaluated for previously approved activities.

3.5.9 Sediment Transfer and Processing Operations

- ***Purpose***. To collect sediment from tanks and sumps in the auxiliary and fuel handling buildings, and also from the containment building basement and sump, in order to transfer the sediment to the spent resin storage tanks and treat or process the sediment (for disposal).

- ***Evaluation: Criticality***. [331] The licensee's safety evaluation concluded that criticality would not occur during sediment processing operations. This conclusion assumed that the fuel estimates for the containment building sump, containment building basement sediment, auxiliary building sump, and sump tank represented a worst case fuel quantity scenario for all sediment sources that would be processed in either the auxiliary and fuel handling building or the containment building.

The licensee's criticality evaluation report [332] for outside the reactor coolant system had previously determined that the critical mass of uranium dioxide with 3 weight percent enrichment and a 0.4-inch maximum pellet diameter in unborated water was 93 kilograms. To ensure that no criticality would occur, a conservative limit was set where the minimum critical mass was determined to be 75 percent of 93 kilograms. Thus, the conservative critical mass was about 70 kilograms. The amount of fuel present in the auxiliary building sump and sump tank was calculated based on the sediment sample analysis reported in the Babcock & Wilcox report, "Characterization of TMI-2 Auxiliary Building Sump and Sump Tank Radwaste," dated December 1984. The mass of uranium dioxide in the sump and sump tank was calculated to be 1.20 kilograms and 0.1325 kilogram, respectively. Consequently, the evaluation concluded that the auxiliary sump and sump tank did not contain sufficient fuel to achieve a critical mass.

Similarly, the licensee's criticality evaluation report [333] for the containment building sump concluded that the amount of fuel present in the basement at the time could vary between 2 kilograms to less than 20 kilograms, which was an insufficient amount of fuel to achieve a critical mass. Therefore, the present evaluation concluded that, based on the above fuel estimates, combining sediment from these sources could not constitute a sufficient quantity of fuel to achieve a critical mass. Additionally, the spent resin storage tank's capacity (3861 gallons) prohibited the combining of all the sediment from these various sources.

- ***NRC Review: Criticality***. [334] A previous NRC review [335] established that the amount of fuel within the various sediments in the basement of the containment building was significantly less than the 70 kilograms required to achieve criticality. The NRC's safety evaluation concluded that the addition of the small amount of fuel that could have been contained in the auxiliary building sump and sump tank would not contribute enough fuel to the total inventory contained in the combined sediments from the containment building and the auxiliary building to create a critical mass.

3.5.10 Pressurizer Spray Line Defueling System

- ***Purpose***. To flush fuel fines and core debris from the pressurizer spray line to the pressurizer vessel and the reactor coolant system (RCS) cold-leg loop 2A. The source of flush water for the pressurizer spray line defueling system (PSLDS) was the defueling water cleanup system. Defueling consisted of flushing the pressurizer spray line in a series of steps to adequately remove fuel fines and debris in each different flowpath from the spray line tie-in.

- ***Evaluation: Criticality***. [336] The licensee's safety evaluation stated that fuel accumulation in the PSLDS would be precluded by using a high-velocity waterflow through the system. This was expected to prevent the settling of fuel fines and debris in the system.

During the TMI-2 recovery operations, the reactor vessel inventory was bled and fed a number of times. This cycling of the vessel water level caused the pressurizer water to combine with the reactor vessel water and create a more uniform boron concentration between the two vessels. Additionally, the operation of the PSLDS promoted circulation of the vessel inventory, which resulted in a more uniform boron concentration. Therefore, boron dilution or concentration was concluded not to be a notable concern.

The PSLDS transported fuel fines into the pressurizer vessel and RCS cold-leg loop 2A. Under optimum conditions, the licensee's technical plan report TMI/TPO-132, "Ex-RCS Criticality Safety," dated November 1985, a criticality study of areas outside the RCS, determined that a minimum of 70 kilograms of fuel was required for criticality. The pressurizer spray line was expected to contain a maximum of 0.1 kilogram of fuel, while the pressurizer vessel and RCS cold-leg loop 2A were expected to contain a maximum of 25 kilograms and 0.1 kilogram, respectively. Therefore, the evaluation determined that the small amount of fuel in the spray line could accumulate in either location and the total amount of fuel would be less than the 70-kilogram limiting value. Further, a possible critical configuration of fuel debris was unlikely. The evaluation concluded that if 70 kilograms of fuel were introduced into the system, the boron concentration present would preclude a criticality.

- ***Evaluation: Boron Dilution***. [337] The licensee's safety evaluation stated that the only credible means of attaining criticality of the fuel contained in the vessel was by deboration of the reactor coolant water. The misconnection of the defueling flexible hose to a nonborated source was identified as a potential cause of a deboration event. To lessen the possibility of a misconnection, all valves, quick-disconnect fittings, and hoses (both ends) would be properly tagged or color coded or both. The control room would monitor all valves used to operate, regulate, and isolate the PSLDS in accordance with procedure. In addition to valve monitoring, all hose connections as well as valve alignment would be visually inspected before system startup to ensure that the flush-water flowpath was as intended by design.

- ***NRC Review: Criticality***. [338] The NRC's safety evaluation stated that the amount of fissile material expected to be contained in the pressurizer spray line was small (less than 0.1 kilogram) when compared to the original amount of fissile material in the pressurizer vessel or the RCS cold leg. The total quantity in either location was much less than the 70 kilograms needed to achieve a critical mass. In addition, the presence of borated water in the pressurizer provided additional margins of safety to ensure subcriticality. Using highly borated water from the defueling water cleanup system (DWCS) to flush out the system was expected to increase the boron concentration in the pressurizer vessel, as well as induce a mixing action. The NRC concluded that, if the RCS chemistry was maintained within the previously approved limits and the DWCS was operated within the constraints specified in the previous NRC safety evaluation report [339] for the DWCS, there would be reasonable assurance that the pressurizer spray line flush program would not cause an inadvertent criticality.

- ***NRC Review: Boron Dilution***. [340] The NRC's safety evaluation stated that dilution of boron concentration in the RCS could result from the process hose being inadvertently connected to a nonborated water source. The NRC concluded that the use of appropriate administrative and procedural controls over the connection process of hoses and the alignment of valves would adequately protect against inadvertent boron dilution in the RCS.

3.5.11 Decontamination Using Ultrahigh Pressure Water Flush

- ***Purpose***. To use ultrahigh pressure (UHP) water flush at 20,000 to 55,000 pounds per square inch to remove surface coatings and surface contamination inside the containment building.

- ***Evaluation: Criticality/Boron Dilution***. [341] The licensee's safety evaluation concluded that activities associated with the use of the UHP water flush would not create the potential for a criticality event in the reactor vessel, given that the water used for the UHP water flush would not be introduced into the reactor coolant system (RCS) during decontamination activities. If the containment building decontamination activities or some other event damages the in-core instrument tubes, recirculation of water from the basement could become necessary. A previous recovery technical specification amendment [342] for boration control addressed recirculation from the basement. The safety evaluation for this amendment demonstrated that capabilities were available to ensure the recirculation of 4350 parts per million (ppm) borated water to the RCS with up to 70,000 gallons of unborated water in the containment building basement. The evaluation considered the effects of foreign materials created by the water flush on the sump recirculation pump and neutron moderation (displacement of boron), boron concentration of water flush, and dilution of the RCS and fuel transfer canal (FTC).

- *Recirculation Pump Clogging*. The intake for the portable pumps that would recirculate water from the containment building basement to the reactor vessel was equipped with a screen with slots of 0.378 inch by 1.5 inches to prevent large debris from entering the pump. The pump was designed to pump water and any entrained debris that could pass through the slots, without damage to the pump. If during operation, the screen became clogged, the screen could be cleared by momentarily shutting the pump off, relocating the pump, or both. Therefore, the evaluation concluded that debris in the containment building basement would not be expected to preclude recirculation.

- *Boron Displacement*. Solid materials that were removed from contaminated surfaces during the decontamination activities, such as concrete and paint chips, could eventually accumulate in the containment building basement. These materials could be introduced into the reactor vessel whenever the recirculation mode was used to maintain the reactor vessel water level. The solid foreign material was not expected to displace borated water as the primary moderator. Therefore, based on the licensee's criticality evaluation report (343) on limits of foreign materials allowed in the RCS, there was essentially no limit on the amount of solid material that could be introduced into the reactor vessel. Consequently, the evaluation concluded that the solid foreign materials, which were removed from contaminated surfaces during the decontamination activities and transported to the reactor vessel during recirculation operations, would not create a criticality safety concern.

- *Flush Water Boration*. The licensee's criticality evaluation report (344) for the containment building sump demonstrated that the water used for containment building decontamination did not require boration to prevent an inadvertent criticality in the containment building sump. Before the NRC's approval of the report, the water used in the UHP water flush was procedurally required to be borated to at least 1700 ppm; however, once the NRC had approved the report, the UHP flush water would not require boration.

- *RCS Dilution*. The use of low boron or unborated water would be procedurally limited to areas where this water could not be intermixed with RCS water, with the exception of the containment building basement reservoir. Additionally, the containment building basement inventory of low boron or unborated water would be maintained below 70,000 gallons to ensure the recirculation of 4350-ppm borated water to the RCS.

  The licensee would maintain records of all low and unborated water uses and inventory in the containment building. The site operations group would act as liaison with the waste management group to match water processing capabilities for decontamination use.

- *FTC Dilution*. The boron concentration in the FTC would be maintained between 4350 and 6000 ppm, according to the recovery technical specifications. Any decontamination activity that could introduce water borated to levels less than 4350 ppm into the FTC would be evaluated to ensure that the operation would not dilute the FTC boron level below this limit. Adequate means, such as FTC water level monitoring or boron sampling, would be available during these decontamination activities to ensure that the boron concentration limit was maintained.

- ***NRC Review: Criticality/Boron Dilution***. (345) The NRC's safety evaluation concluded that the license's submittal provided reasonable assurance that the UHP decontamination program would not present the potential for any inadvertent criticality in the RCS or the containment building sump. Water used for the UHP flush was not to be intentionally introduced into the RCS or FTC during operations. In addition, the UHP flush flow rate would generally be kept low, and, in the unlikely event of an inadvertent introduction of flush water, a rapid boron dilution was not expected to occur. The review also determined that the existing requirements for RCS and FTC level monitoring and boron analysis provided a means to detect the event before a criticality potential would develop. Water usage in the UHP system would be administratively controlled to ensure that the boration requirements were met by the approved evaluations relating to containment building sump criticality control.

3.6 Evaluations for Defueling Operations

3.6.1 Preliminary Defueling

- ***Purpose***. To allow movement of debris within the reactor vessel, to allow installation of defueling equipment, and to identify core debris samples for later removal and analysis by INEL. The movement of debris within the vessel was prepared for early defueling and required some rearrangement of debris in the reactor vessel before the actual removal of fuel. These activities included the loading of small pieces of core debris into debris baskets but did not include actual loading of fuel debris into fuel canisters.

- ***Evaluation: Criticality***. (346) The licensee's proposal letter noted that it had submitted its request for preliminary defueling while the NRC was reviewing the licensee's safety evaluation report (SER) (347) for early defueling. The purpose of this letter was to advise the NRC of the licensee's intention to begin preliminary defueling activities and to define the limit of these activities in the context of the early defueling SER. The preliminary defueling activities would be performed wholly within the reactor vessel and would consist of the segmentation and movement of core debris (i.e., core alterations) necessary to permit full operation of the canister positioning system, as a prerequisite to fuel loading and selection of samples needed for early transfer to INEL. These activities could include the loading of small pieces of core debris into debris baskets but would not include the actual loading of fuel debris into fuel canisters. The core debris would be reconfigured in accordance with the TMI-2 technical specifications and would be bounded by the licensee's criticality evaluation report [348] for the reactor coolant system (RCS).

The operations involving defueling tooling, including hydraulically operated tools, and equipment described in the early defueling SER were pending NRC approval. Operating procedures for these activities were also subject to NRC approval, as appropriate. It was intended that the procedures would be used within the operating limits defined in the licensee request letter, to the extent that these activities were bounded by existing defueling procedures. Therefore, provisional approval was requested for the operating procedure of conducting the preliminary activities specified in the request letter. In addition, before conducting operations using the hydraulically operated tools, the licensee would provide results of the tests of the borated hydraulic fluid to demonstrate: (●) miscibility; (●) compatibility with the defueling canister recombiner catalyst; and (●) stability as a homogeneous borated fluid. The preliminary defueling operations would be supervised by NRC-licensed fuel handling senior reactor operators, most likely from the command center.

- ***NRC Review: Criticality***. [349] The NRC's safety evaluation stated that safety issues related to the movement of material in the reactor vessel were addressed in previous NRC SERs [350, 351, 352, 353] for the (●) reactor vessel head lift; (●) plenum removal preparatory activities; (●) plenum assembly lift and transfer; and (●) heavy load handling over the reactor vessel. The licensee's criticality evaluation report [354] for the RCS concluded that for an RCS boron concentration of 4350 parts per million (ppm), the damaged core would remain subcritical with a shutdown margin of at least 1 percent for any postulated fuel configuration. This conclusion applied to all reactor disassembly and defueling activities, including the proposed preliminary defueling activities. During the proposed activities, the RCS would be maintained at a boron concentration at the administrative limit of 4950 ppm. The NRC concluded that this concentration would provide an additional margin to preclude the potential for criticality due to a boron dilution event or the introduction of foreign material into the RCS.

The effects of foreign materials (e.g., tools, fluids) on RCS reactivity were analyzed. Administrative controls would be implemented to limit the potential for the inadvertent introduction of such materials into the RCS throughout defueling. The NRC reviewed the analysis and concluded that the potential for introduction of foreign materials into the RCS during preliminary defueling activities was very small and that the resulting effects on RCS reactivity would not significantly reduce the existing shutdown margin at the time. In addition, procedures would be in place to prevent the inadvertent grappling of an in-core instrumentation tube that, if damaged, could result in an RCS leak that could not be isolated. In the unlikely event of an RCS leak due to failure of an in-core instrumentation tube, adequate makeup capability existed to ensure that subcriticality was maintained.

- ***NRC Review: Boron Dilution***. [355] A high concentration of boron in the reactor coolant was the primary means used to ensure against the possibility of a criticality event during defueling activities. A revised hazard analysis report [356] for potential dilution of the RCS was submitted to the NRC in support of early defueling. This report evaluated all potential RCS boron dilution pathways and isolation barriers for these paths. Systems with potential boron concentrations below the RCS concentration of 4950 ppm would be isolated by multiple barriers to ensure that they would not be credible dilution sources. The hydraulic fluid used in the operation of defueling tools, which was tested to demonstrate its miscibility and compatibility with RCS water, would be borated to eliminate the potential for RCS dilution. The hazard analysis also described sampling locations, frequencies, and provisions for level monitoring that would provide the capability for early detection and subsequent mitigation of a dilution event. Based on a review of the hazard analysis, the NRC concluded that (●) the potential for a dilution event was small; (●) early detection of a dilution event was likely; and (●) effective remedial action could be taken if dilution occurred.

The RCS would be maintained at a boron concentration of 4950 ppm during the proposed activities, thereby providing an additional margin above the recovery technical specifications limit of 4350 ppm, the value used in the hazard analysis. Thus, the NRC determined that sufficient time would be available to detect and mitigate an unlikely dilution event.

3.6.2 Early Defueling

- ***Purpose***. To remove end fittings, structural materials, and related loose debris that would not involve removal of significant amounts of fuel material, to remove intact segments of fuel rods and other pieces of core debris, and to remove loose fuel fines (particles) by vacuuming operations.

- ***Evaluation: Criticality***. Editor's Note: The licensee's criticality safety evaluation for early defueling was presented in Revision 4 [357] of its safety evaluation report (SER). Revision 4 was identical to the criticality safety evaluation presented in the licensee's subsequent SER for bulk defueling, with the exception of additional considerations unique to bulk defueling (e.g., use of the abrasive/water jet system, removal of the canister head for gasket replacement, and the use of debris containers). This section will not repeat the duplicative criticality evaluation for early defueling. (Refer to the SER for bulk defueling.) However, the safety evaluation for boron dilution during early defueling differed from that for bulk defueling; therefore, separate evaluations are presented in each section.

- ***Evaluation: Boron Dilution (Early Defueling)***. [358] The licensee's SER (Revision 4) concluded that the reactor coolant system (RCS) temperature and chemistry would not be significantly affected during early defueling; therefore, boron solubility would remain essentially unchanged. The evaluation stated that the only scenario in which the RCS boron concentration could be changed in an uncontrolled manner during early defueling was by dilution of the RCS coolant with water that was either unborated or borated below 4950 parts per million (ppm). Potential sources of this water were the various systems connected to the RCS, including the secondary system. Systems with boron concentrations less than 4950 ppm were identified and isolated to ensure that they would not be credible sources of boron dilution, whereas water in the deep end of the fuel transfer canal and in spent fuel pool "A" would be maintained at boron concentrations greater than or equal to 4350 ppm. The licensee's hazard evaluation report [359] for potential boron dilution of the RCS assessed all potential dilution paths and the isolation boundaries for these paths. The hazard report provided dilution detection criteria (e.g., level monitoring, sampling frequency) for static conditions and all modes of water processing during early defueling to ensure that the RCS boron concentration would remain greater than or equal to 4350 ppm.

- ***NRC Review: Criticality***. [(360)] The NRC's safety evaluation concluded that the potential for a criticality event in the reactor vessel or defueling canisters during early defueling activities was acceptably low. The NRC reviewed the potential for a criticality event during early defueling activities that involved fuel in the reactor vessel, as well as fuel transferred and stored in canisters.

  - *Reactor Vessel*. The safety evaluation for preliminary defueling activities referenced the NRC's approval of the licensee's criticality evaluation report [(361)] for the RCS. Based on the NRC's SER [(362)] for preliminary defueling activities and the licensee's criticality evaluation report for the RCS, the NRC determined that, during all reactor disassembly and defueling activities with an RCS boron concentration of 4350 ppm, the damaged core would remain subcritical with a shutdown margin of at least 1 percent for any postulated fuel configuration. The actual RCS boron concentration during defueling activities would be no less than the administrative limit (4950 ppm), and this limit would provide a substantially greater shutdown margin to preclude the potential for criticality of the core. Based on the licensee's analysis and administrative controls, the NRC also concluded that the potential for the introduction of foreign materials into the RCS during early defueling was very small and that the resulting effect on RCS reactivity would not significantly reduce the existing shutdown margin. Therefore, based on the NRC's previous SERs [(363, 364)] for the start of preliminary defueling operations and for RCS criticality, the agency concluded that safety margins were adequate to minimize the potential for criticality of the fuel in the reactor vessel during early defueling activities.

  - *Defueling Canister*. The NRC had independently verified the calculations performed by the licensee to demonstrate that fuel in canisters would remain subcritical during normal and postulated accident conditions. Using conservative assumptions, the analyzed cases included all three types of canisters under the following conditions: (●) single canister, (●) canisters stored in an array, and (●) a canister deformed from a worst case drop accident. Two fuel configurations were analyzed for the knockout and filter canisters, and the effect of the canister transfer shield on the reactivity of undamaged, loaded canisters was also analyzed. In addition, calculations were performed to analyze the case in which the entire contents of a loaded canister were dropped around another filled canister that was stored in the storage racks. All of the analyzed cases yielded effective neutron multiplication values below 0.95. Based on the use of conservative assumptions and the independent verification of the calculations, the NRC concurred with the licensee's criticality analysis for fuel in defueling canisters.

- ***NRC Review: Boron Dilution***. [365] The NRC's safety evaluation stated that criticality of the fuel in the reactor vessel was prevented by maintaining a high boron concentration in the RCS. The licensee's SER [366] for the potential boron dilution of the RCS demonstrated that the potential for a boron dilution event leading to criticality of the core was extremely low during the proposed early defueling activities. That SER evaluated all potential RCS dilution pathways and their isolation barriers and also described the measures for RCS sampling and level monitoring for early detection and mitigation of a potential dilution event. Other precautions would be taken to minimize the potential for dilution, which included the isolation of potential sources of unborated or underborated water using multiple barriers and the boration of hydraulic fluid that was used in the operation of defueling equipment. Based on the review of the licensee's SER for the potential boron dilution of the RCS and other actions taken to prevent a dilution event, the NRC concluded the following: (●) the potential for a boron dilution event during early defueling was small; (●) early detection of a dilution event was likely due to the sampling and monitoring capability and the large margin provided by the operating RCS boron concentration; and (●) effective remedial action could be taken to terminate the dilution and provide borated makeup water to the RCS.

3.6.3 Storage of Upper End Fittings in an Array of 55-Gallon Drums

- ***Purpose***. To use shielded 55-gallon drums to store end fittings and other structural material removed from the reactor vessel.

- ***Evaluation: Criticality***. [367] The licensee's safety evaluation considered critical conditions for a single storage container, multiple storage containers, and storage areas.

  - *Single Container*. Each end fitting was estimated to contain about 2 or 3 kilograms of fuel if the fittings were packed solidly within the flow spaces in the end fitting casting. End fittings would be examined to ensure that fuel debris was not fixed or clinging to the surface so that the fuel content did not substantially exceed the estimated quantities. Thus, a maximum of 10 end fittings in a container represented a potential fuel loading of 20 to 30 kilograms. This fuel loading was significantly less than the minimum 70 kilograms of 3-percent enriched fuel required for a possible criticality under any condition. The end fittings would be submerged in water with a minimum of 4350 parts per million (ppm) boron. Both the 55-gallon drum and NC-90 containers would be closed with a lid to prevent inadvertent dilution of the borated water. Therefore, criticality in a single drum would be precluded under any condition.

  - *Multiple Containers*. The potential for criticality of about 10 containers was evaluated. As stated previously, the fuel would be submerged in water with a minimum of 4350 ppm boron. The licensee's safety evaluation report [368] for bulk defueling showed this condition to be critically safe for any conceivable configuration or array. [bb] The licensee's safety evaluation

[bb] Editor's Note: The referenced report itself does not specifically evaluate 55-gallon drums containing these materials. It does, however, include an evaluation for debris containers (refer to Section 4.2.4 of that report). Although the dimensions of the canisters, which are similar to defueling canisters, differ from those of a 55-gallon drum, the evaluation for those containers is relevant and applicable to the 55-gallon drums. The applicability of the

report [369] for seismic design criteria examined the possibility for criticality of the reactor vessel and defueling canisters under draindown conditions. The report concluded that the core material would not become critical due to the lack of moderator. Therefore, if one of the double-walled containers were to leak and expose the end fittings, a criticality still would not occur.

- *Storage Areas*. Other storage areas that were allowed to contain more than 70 kilograms of fuel were not permitted to be within 12 feet of the storage area for the 55-gallon drums. The containers would be treated as special nuclear material and handled in accordance with approved plant procedures. Since 10 such containers stored in one place constituted a potential accumulation of about 4200 to 6300 grams of fissile material (based on a 3-percent enrichment), suitable criticality monitors would be installed in the vicinity of the storage area in accordance with 10 CFR 70.24, “Criticality accident requirements.” A change request for the recovery operations plan [cc] was submitted to support installation of these monitors. Monitoring the boron content of the containers was not planned after the initial filling of the containers. This decision was based on the following: (●) Water was used with a minimum of 4350 ppm boron. (●) Double-walled containers used included covers. (●) Containers were stored in a restricted access area. (●) Demineralized water sources were limited in the immediate area of storage.

- ***Evaluation: Criticality (Load Drop)***. [370] The licensee’s safety evaluation noted that each storage container weighed about 3000 pounds when loaded and would be positioned in a lifting rig on the defueling platform. The lifting rig would be load tested to 200 percent of the loaded weight. Restrictions on lifting this weight would be observed. When the container was filled, it would be lifted out of the protective lead shield of the platform, moved to the 347-foot elevation of the containment building, and positioned near the reactor head storage area. Once the container was properly located, the crane would be remotely decoupled from the container and moved back to the defueling platform for the next container. Two postulated accident conditions were evaluated: a loaded storage container dropped onto the containment floor and a heavy load dropped onto the containers in storage.

- *Loaded Container Dropped*. In the unlikely event that a loaded storage container was dropped onto the containment floor, the evaluation assumed that the container would be damaged and its contents spilled onto the floor. Since each storage container had a maximum of 30 kilograms of fuel, criticality was not a concern as the fuel available for spillage was less than the 70 kilograms required for a criticality event. Pyrophoricity concerns and releases due to a dropped storage container were bounded by the analysis performed for a dropped fuel canister in the safety evaluation report [371] for bulk defueling.

---

evaluation is evident from a comparison of the contents and the soluble boron concentrations for the containers with the 55-gallon drums, as well as from the assumptions used in the analysis for the canisters.

[cc] Editor’s Note: Section 4 of the recovery technical specifications was called the “recovery operations plan”; this plan required NRC approval but not an operating license amendment.

- *Load Drop onto Containers*. While the containers were being stored, there was the potential that a heavy load could drop on them. In the event of a heavy load dropped into the storage area, the evaluation assumed that all the storage containers in the area at the time of the drop were destroyed. As mentioned previously, 70 end fittings, each containing 2 to 3 kilograms of fuel, would be stored in the containers. Thus, on a worst case basis, 210 kilograms of fuel could be spilled if all containers were destroyed. The 210 kilograms of fuel would pose a criticality potential. However, this potential was determined to be small enough to be considered nonexistent based on the following: (●) fuel on the floor with 4350 ppm borated water would be subcritical; (●) fuel on a dry floor (i.e., no moderator) would be subcritical; (●) fuel mixed with unborated water could become critical, although unborated water at a sufficient depth of the 347-foot elevation of the containment building would be highly unlikely; and (●) fuel in the end fittings was assumed to be within the flow space of the end fittings. Attempts would be made to remove this fuel from the flow space before its transfer to the storage container; therefore, it was unlikely that fuel remaining in the flow space would be dislodged by a heavy load drop. The fuel in the end fittings flow space would be subcritical in any type of water.

- *Conclusion*. The licensee concluded that a criticality due to the release of fuel from all containers as a result of a heavy load drop was unlikely. Releases because of a heavy load drop onto the storage containers were bounded by the analysis performed for a dropped fuel canister in the licensee's safety evaluation reports [372, 373] for bulk defueling.

- ***Evaluation: Criticality (Revised)***. [374] In response to the NRC's comments, the licensee stated in supplementary correspondence that Oak Ridge National Laboratory had performed computer analyses for two potential configurations in the storage containers.

- *Criticality Models*. The parameters used in the criticality models for the two cases included the following:

  – *Case 1*: The modeling parameters for Case 1 included the following: (●) A single 55-gallon drum was contained within an NC-90 overpack storage container. (●) The 55-gallon drum was wrapped within 1.375 inches of lead. (●) The overpack storage container was a polyethylene/fiberglass 90-gallon drum with a 1-1/32 inch wall thickness where about 3/16 inch was polyethylene. For the analysis, the 3/16-inch polyethylene was conservatively assumed to be equivalent to 11/32 inch of unborated water. (●) The 55-gallon drum volume was filled with optimally moderated Batch 3 fuel (2.96 weight percent enrichment with fuel burnup) that used borated water with a 4950-ppm concentration. (●) The annulus between the liner and the drum was filled with unborated water. (●) The liner rested on the concrete floor. (●) For additional conservatism, a 2-inch-thick lead shield was assumed to be placed on top of containers.

– *Case 2*: The modeling parameters for Case 2 included an infinite planar array of storage containers as described in Case 1, with no clearance between adjacent storage containers. [dd]

- *Results*. The criterion used to establish a safe configuration in the storage containers was at least a 1-percent shutdown margin (i.e., effective neutron multiplication (k_{eff}) no greater than 0.99). The computer analysis results demonstrated that the storage containers would be subcritical (specifically, a k_{eff} of 0.820 for Case 1 and a k_{eff} of 0.918 for Case 2).

- ***NRC Review***. [375, 376] The NRC's safety evaluation [ee] considered both a single drum and an array of drums.

- *Single Container*. The NRC stated that the licensee's results showed a more than adequate shutdown margin for a single drum. The calculation case for a single drum was supported by the licensee's safety evaluation report [377] on limits of foreign materials allowed in the reactor coolant system, which the NRC had previously reviewed [378] in its safety evaluation.

- *Multiple Containers*. The NRC stated that the licensee's criticality calculations showed a shutdown margin significantly greater than 5 percent (k_{eff} of 0.918) for the worst case planar array. The NRC indicated that the licensee's model was very conservative for the following reasons: (●) neglected neutron absorption by the stainless-steel end fitting; (●) assumed drums full of fuel (actual drums would be partially full); and (●) assumed entirely Batch 3 fuel (highest enrichment) and optimum fuel-to-water ratio.

- *Conclusion*. The NRC's calculations using similar conservative models independently confirmed the licensee's results. The NRC concluded that the transfer and storage of end fittings in shielded 55-gallon drums did not present a significant criticality risk, provided that 4950 ppm boron was maintained in any water contained in the drums.

- *Restrictions*. Since soluble boron provided the shutdown margin, the NRC stated that the licensee's operating procedures should limit those activities (hydrolasing) that could introduce nonborated water in the storage area. These procedures should also provide for periodic reverification of the boron concentration if the storage interval was protracted.

3.6.4 Defueling (Also Known as "Bulk" Defueling)

- ***Purpose***. To develop defueling tools and to conduct activities necessary to remove the remaining fuel and structural debris located in the original core volume. An additional activity

---

[dd] Editor's Note: The source document does not clearly indicate whether the planar array was a square-pitch or a triangular-pitch array. For an optimized array, a triangular-pitch array should be used. Therefore, the editor believes that the analysis described here used a triangular-pitch array.

[ee] Editor's Note: The first NRC safety evaluation report approved the loading and storage of a single drum in the 347-foot elevation storage area. A follow-on review approved loading and storage of an array of 55-gallon drums.

was to address vacuuming of debris from the lower head of the vessel, a primary activity associated with lower head defueling.

- ***Evaluation: Criticality/Boron Dilution***. Editor's Note: The licensee's criticality safety evaluation for bulk defueling was presented in Revision 10 [379, 380] of its safety evaluation report (SER). Revision 10 was identical to the criticality safety evaluation presented in Revision 4 [381] of the SER for early defueling, with the exception of additional considerations unique to bulk defueling (e.g., use of the abrasive/water jet system, removal of the canister head for gasket replacement, and the use of debris containers). Criticality discussions that applied specifically to Revision 10 are presented below inside **[**brackets**]**.

Safety evaluations for boron dilution during early defueling differed from those for bulk defueling; therefore, discussions in this section specific to bulk defueling are not presented inside brackets.

- ***Evaluation: Criticality***. [382, 383] The licensee's safety evaluation considered the potential for criticality in the reactor vessel, defueling canister handling operations, and canister storage racks.

- o *Reactor Vessel*. The licensee's safety evaluation considered the effect of movement of reactor components and fuel in the reactor vessel on reactivity and the introduction of foreign materials that could increase neutron moderation.

  - – *Fuel Movement*. Criticality calculations were previously performed to determine the minimum boron concentration required in the reactor coolant system (RCS) to maintain a 1-percent shutdown margin. A conservative and bounding fuel model was used to determine the minimum boron concentration. The results of this design-basis model also accounted for computer code uncertainty. The results from these previous calculations showed that a boron concentration of 4350 parts per million (ppm) would ensure that the reactor core would be maintained subcritical with effective neutron multiplication (k_{eff}) no greater than 0.99 during all reactor disassembly and defueling operations. This included the movement of any reactor component, including fuel, within the vessel, whether planned or due to an accident such as a heavy load drop. The licensee's criticality evaluation report [384] for the RCS provided the basis and models used in the selection of a subcritical boron concentration for defueling.

  - – *Foreign Materials*. The potential existed for reactivity to increase with the introduction of additional materials to the RCS or reactor vessel. This increase could occur when the introduced materials acted as neutron moderators or reflectors or diluted the boron concentration to below 4950 ppm. To avoid this situation, materials that could be located on the defueling work platform or handled within the reactor vessel were reviewed. In determining the effect of these materials on the shutdown margin, assuming they were brought into contact with the fuel, the materials were considered as moderators and reflectors. For the purpose of this evaluation, the RCS boron concentration was assumed to be the lower administrative limit for operations of 4950 ppm. A quantity of the various materials was determined such that the resultant k_{eff} did not exceed 0.99 for all credible situations, as described in the licensee's SER [385] on limits of foreign materials allowed in the RCS. Procedural controls would be implemented to ensure that limitations on material type and quantity were not violated.

[Additionally, the abrasive grit to be used in the abrasive/water jet system was evaluated in Revision 10 of the SER for bulk defueling to ensure that the addition of this material to the reactor vessel would not result in a criticality safety concern. Controls were implemented to ensure that limitations on material type and quantity were not violated (e.g., only abrasive grit materials that were evaluated for their effect on k_{eff} would be used; hydraulic system working fluid was borated to at least 4350 ppm).**]**

- *Canister Handling Operations*. The licensee's safety evaluation stated that canister handling operations presented **[**three**]** areas of concern for criticality. The first dealt with the transport of the canisters in the canister transfer shield, and the second dealt with dewatering of the canisters. **[**The third was added in Revision 10 of the SER for bulk defueling and related to the removal of the canister head for gasket replacement.**]**

  – *Canister Transport*. Lead and steel in the canister transfer shield and shield collar were anticipated to act as a reflector for neutrons when a canister filled with core debris was placed inside the shield. Criticality calculations were performed to verify that an adequate shutdown margin (i.e., k_{eff} no greater than 0.95) would be maintained during operations involving the canister transfer shield. The criticality analyses for the various configurations using the canister transfer shield were analyzed using the KENO Version IV computer code.

    ▪ *Modeling Assumptions*. The calculational models for the canister in the transfer shield assumed the following conservative conditions: (●) Batch 3 unirradiated fresh fuel only; (●) fuel enrichment of 2.98 weight percent enrichment (2.96 weight percent + 2 sigma); (●) no cladding or core structural material; (●) no soluble poison or control rod material from the reactor core; and (●) optimal fuel lump size and volume fraction and optimal water moderator density (except in parametric cases for the optimization study).

    ▪ *Conclusion*. Insertion studies concluded that the 100-percent canister insertion configuration (i.e., a canister completely inside the canister transfer shield) was the most reactive. The results of the calculation indicated that no poison material was required in the design of the transfer shield because k_{eff} was calculated to remain below 0.95. These results were valid for standard, unruptured canisters and for canisters with ruptured internals. The licensee's technical evaluation report [386] (TER) for the defueling canisters presented further details.

  – *Canister Dewatering*. The evaluation concluded that the criticality concerns for this process could be considered bounded by the results in the TER; the analyses performed for the TER were completed with optimal fuel/moderator ratios.

- [*Canister Gasket Replacement*. The licensee's SER (Revision 10) considered criticality concerns for the replacement of canister gaskets with canisters containing fuel debris. The licensee's TER [387] for defueling canisters demonstrated that the maximum k_{eff} for a single, loaded fuel canister moderated with unborated water was 0.857. The removal of the canister head was not expected to appreciably affect this value. Additionally, the canister would remain in spent fuel pool "A" (SFP-A) during the gasket replacement. As the recovery technical specifications required the water in SFP-A to be borated to at least 4350 ppm, the water within any open fuel canister would also be borated. Taking credit for the borated water within the canister would reduce k_{eff} to a value well below 0.857. Consequently, the evaluation concluded that the planned activities associated with the head gasket replacement would not result in a canister k_{eff} exceeding 0.95. Additionally, if any of the open canister's contents were spilled into SFP-A, subcriticality would be ensured by the 4350-ppm boron concentration in the pool.]

- *Canister Storage Racks*. Criticality calculations were previously performed to demonstrate that the defueling canister array in the storage racks would maintain a k_{eff} no greater than 0.95. The licensee's TER [388] for defueling canisters presented further details.

- [*Debris Containers*. The licensee's SER for bulk defueling (Revision 10) considered criticality concerns for debris containers. Debris containers were used for removing fuel assembly upper end fittings, control component spiders, or other structural material from the reactor vessel. The contents of the debris container would then be emptied into a fuel canister for long-term storage.]

  [The criticality evaluation was performed independently of the fuel inventory in the debris containers; therefore, no restrictions were required on the amount of fuel loaded into the containers. However, efforts were made to limit the amount of fuel entering the containers. These efforts included limiting fuel rod end stubs to about 2 inches and limiting debris to structural materials with no significant quantities of unidentifiable material (i.e., chunks or agglomerations) attached. Thus, the containers were not expected to contain significant quantities of fuel. The criticality analysis was typically bounded by the licensee's criticality SER [389] for the RCS (referred to below as SER-RCS). The evaluation included loaded containers in the reactor vessel or SFP-A, containers inside the canister transfer shield, and container storage.]

  - [*Loaded Containers*. When being loaded in the reactor vessel or temporarily stored in either the fuel transfer canal or SFP-A, the debris containers would be submerged in water with a boron concentration of at least 4350 ppm. Since the containers were vented, the evaluation assumed that the boron concentration of the water within the containers was equal to that of the surrounding water. Previous SER-RCS analyses demonstrated that the core would remain shut down, with a k_{eff} no greater than 0.99, when the RCS water was borated to a concentration of at least 4350 ppm. Although differences existed between the assumptions used in the SER-RCS analyses and those that would actually have been used for an explicit analysis of submerged containers, direct application of the SER-RCS results to this evaluation was conservative.]

- **[***Model Assumptions*. The evaluation stated two ways in which the model was conservative. First, the SER-RCS model included the entire core. With the containers being significantly smaller than the core, the containers would experience significantly more neutron leakage; thus, k_{eff} was reduced. Second, the SER-RCS model did not consider structural materials, whereas the majority of the containers' inventory would comprise structural materials. Previous analyses showed that structural material, such as the stainless steel in end fittings, tended to act as a neutron poison; thus, structural material would reduce k_{eff}.**]**

- **[***Conclusion*. Based on the results of the SER-RCS analyses and associated conservatisms, the evaluation concluded that the containers would be subcritical (i.e., k_{eff} no greater than 0.99) under all of the following conditions: (●) one or more containers were submerged in water; (●) containers were filled with water; and (●) containers have a boron concentration of at least 4350 ppm.**]**

– **[***Canister Transfer Shield (CTS)*. For the container inside the CTS, the lead and steel walls of the CTS were expected to act as an additional neutron reflector, thus tending to increase k_{eff}. The analyses of the SER-RCS and the licensee's criticality evaluation report [390] on the limit of foreign materials allowed in the RCS during defueling activities were used to demonstrate that the containers were subcritical when inside the CTS. In the foreign materials report, a lead shell 65 centimeters (25.6 inches) thick was applied to the outside of the core region. The resulting net increase in k_{eff} was 0.03 percent. Though this analysis was completed for a boron concentration of 4950 ppm, the increase in k_{eff} was expected to be similar for a 4350-ppm boron concentration. When this increase was added to the k_{eff} of 0.9896, which was calculated for the core with a 4350-ppm boron concentration, the resultant k_{eff} was still below 0.99. Consequently, the evaluation demonstrated that the entire core would remain critically safe after the addition of a 65-centimeter-thick lead reflector.**]**

  - **[***Modeling Assumptions*. Aspects of the criticality models used in these previous analyses (i.e., the analyses for the SER-RCS and foreign materials allowed in the RCS) were identified as conservative versus the configuration and operation conditions for debris containers in the CTS. These conservatisms included the following: (●) Both previous analyses included the entire core. Additionally, the CTS walls were significantly thinner (about 6.5 inches) than the 25.6-inch-thick shell. With the containers being significantly smaller than the core and the CTS walls thinner than the modeled shell, the containers within the CTS would experience significantly more neutron leakage, thereby reducing k_{eff}. (●) Both previous analyses did not consider structural material, whereas most of the containers' inventory would be made up of structural material. Previous analyses showed that structural material, such as the stainless steel in end fittings, tended to act as a neutron poison; thus, structural material would reduce k_{eff}. (●) The actual boron concentration of the RCS would be administratively maintained at greater than or equal to 4950 ppm. The recovery technical specifications required the water in SFP-A to have a minimum concentration of 4350-ppm boron; however, the fuel transfer canal and SFP-A were expected to be operated at about 4500-ppm boron. Any concentrations greater than 4350 ppm would cause a reduction in k_{eff} below the value calculated in the SER-RCS. (●) The SER-RCS value of k_{eff} assumed the presence of an 8-inch stainless-steel reflector on the outside of the core; thus, the k_{eff} value of 0.9896 already took credit for some reduced neutron leakage.**]**

- **[***Conclusion*. Previous analyses demonstrated that the entire core would remain critically safe after the addition of a 65-centimeter-thick lead reflector. Accounting for the conservative nature of these previous analyses versus the configurations and operations of the debris containers inside the CTS, k_{eff} would be reduced even further. Therefore, the debris containers would remain subcritical inside the CTS.**]**

– **[***Container Storage*. The criticality effect of storing the debris next to defueling canisters that were co-located in the storage racks was evaluated. Positioning of debris containers near defueling canisters was not allowed if the k_{eff} of the defueling canisters exceeded their licensing criteria. These criteria would be met by separating the debris containers from any defueling canister by at least one empty storage cell. An individual debris container did not need to be separated from another debris container by an empty cell because the 4350-ppm boron concentration ensured that the containers would remain below a k_{eff} of 0.99.**]**

- ***Evaluation: Boron Dilution (Bulk Defueling)***. [391] The licensee's SER (Revision 10) stated that defueling would not significantly affect the RCS temperature and chemistry; therefore, boron solubility would remain essentially unchanged. The only way the RCS boron concentration could be changed in an uncontrolled manner during defueling was by dilution of the RCS coolant with water that was either unborated or borated below 4950 ppm.

o *Dilution Sources*. Potential sources of deborated water were the various systems connected to the RCS, including the secondary system. Systems with boron concentrations less than 4950 ppm were identified and isolated to ensure that they would not be credible sources of boron dilution. The water in the deep end of the fuel transfer canal and in SFP-A would be maintained at boron concentrations greater than or equal to 4350 ppm. The licensee's criticality evaluation report [392] for the containment building sump assessed potential dilution paths and the isolation boundaries for three paths. This reference provided dilution detection criteria (e.g., level monitoring, sampling frequency) for static conditions and all modes of water processing during defueling to ensure that the RCS boron concentration would remain greater than or equal to 4350 ppm.

o *Abrasive/Water Jet Cutting System*. The use of the ultrahigh pressure pump of the abrasive/water jet cutting system for containment building decontamination activities, as well as in-vessel cutting operations, presented a boron dilution safety concern that was not specifically addressed in the sump criticality report.

– *Water Supply*. The supply water to the pump during cutting operations was borated to greater than or equal to 4950 ppm. During containment building decontamination operations, the pump was supplied with water from a low boron (or unborated) water source. Consequently, when the decontamination water source was connected to the pump, the pump could not be used for in-vessel operations to avoid introducing low-boron water into the fuel regions of the reactor vessel. In addition, when changing from the decontamination mode of operation to the in-vessel cutting mode, the residual low-boron water in the pump would be purged before the insertion of the cutting tool into the fuel region.

– *Preventive Measures*. To eliminate the above-mentioned safety concerns, the low-borated water supply hose was coupled permanently to the decontamination nozzle hose and the greater than or equal to 4350-ppm borated water supply hose was permanently coupled to the abrasive/water jet cutting tool hose. A common mating fixture was attached to the pump supply and discharge lines. To further prevent the use of a low-borated water source for in-vessel cutting, the set of hoses for decontamination activities would have a permanent tag attached to the connecting fixture to warn plant personnel not to use them for in-vessel operations. The ultrahigh pressure pump for either in-vessel or decontamination activities would be operated only after verification that the proper hoses were connected to the pump.

– *Residual Low Borated Water*. The ultrahigh pressure pump would contain about 5 gallons (according to the pump manufacturer) of low-borated water at the cessation of the decontamination mode of operation. Upon starting the pump for in-vessel cutting operations (after disconnecting the decontamination hoses and connecting the abrasive/water jet cutting hoses), this residual water could be injected into the reactor vessel. The criticality safety report for the sump summarized evaluations that addressed the issue of mixing the residual water with the RCS volume. The report stated that unborated water entering the reactor vessel at the elevation of the hot- and cold-leg piping was likely to rise directly to the internal indexing fixture rather than flow down to the core region.

– *Administrative Controls*. Administrative controls would require that the abrasive water jet cutting nozzle be discharged in a radial direction above the reactor vessel flange and away from suction lines to other systems within the reactor vessel, for a minimum of 5 minutes (pump flow rate was about 2.8 gallons per minute and residual water in pump was about 5 gallons) after the pump was used for decontamination activities. This would minimize the potential for the low-borated water to enter the debris bed or RCS processing systems before mixing with the RCS water.

The water in the RCS was maintained at a boron concentration of greater than or equal to 4950 ppm. In addition, this concentration would be verified before the use of the ultrahigh pressure pump for in-vessel cutting and following the use of the pump for decontamination activities. The introduction of 5 gallons of unborated water to the upper region of the reactor vessel during purging of the residual pump water would have a negligible effect on overall RCS boron concentration.

- ***NRC Review: Criticality***. [(393)] The NRC's safety evaluation stated that the potential for a criticality event during defueling activities was effectively minimized by maintaining a high boron concentration in the RCS. In the NRC's SER [(394)] for early defueling, the report referenced an earlier conclusion that, at an RCS boron concentration of 4350 ppm, the damaged core would remain subcritical with a shutdown margin of at least 1 percent for any postulated fuel configuration. Additional margin existed since the licensee would administratively maintain the actual RCS boron concentration at 4950 ppm. Experience in defueling efforts to date demonstrated the effectiveness of this approach in maintaining subcriticality.

In its early defueling review, the NRC also concluded that sufficient margin existed to maintain subcriticality in case of the inadvertent introduction of foreign materials into the RCS. The tools and equipment used for bulk defueling were analyzed to ensure that a 1-percent shutdown margin would exist for all credible events. The NRC concluded that approximate measures were being used to ensure that adequate margins existed to minimize the potential for criticality of the remaining fuel in the reactor vessel.

Canister handling and storage procedures would not differ significantly from those currently in use; therefore, the conclusions of the early defueling review also applied to core region bulk defueling activities (i.e., the potential for a criticality event in the handling and storage of defueling canisters would be acceptably low).

- ***NRC Review: Boron Dilution***. [(395)] In the NRC's SER [(396)] for early defueling, the agency had previously concluded that the licensee implemented acceptable controls to minimize the potential for a boron dilution event and to effectively mitigate the consequences of such an event during early defueling. These controls, which would remain in effect during bulk defueling, included the use of multiple barriers to isolate potential dilution sources and the boration of the hydraulic fluid in defueling equipment.

  - *Dilution Sources*. Additional potential sources of boron dilution during core region defueling that were not evaluated for early defueling activities were the hydraulic systems for the core equipment and the ultrahigh pressure decontamination water. In the NRC's SER [(397)] for the core stratification sample acquisition (core bore), the agency concluded that the two unborated hydraulic fluids that were used with the core bore equipment did not present a credible source for a dilution event that would result in advertent criticality of the core. This conclusion was based on the following: (●) sources were of small volume (1.4 to 27 gallons), compared to the large volume of borated RCS water above the core region (20,000 gallons); (●) fluids would tend to mix well with the borated RCS water; and (●) fluids would be introduced near the surface of the RCS water, away from the core. This conclusion also applied to the use of the core bore equipment for defueling purposes.

  - *Administrative Controls*. The licensee's SER [(398, 399)] for bulk defueling described the physical and administrative controls that would prevent a boron dilution event from an improper alignment of the ultrahigh pressure pump. The pump was used with a borated water supply during in-vessel abrasive/water jet cutting operations. When used for containment building decontamination, the pump was supplied with unborated (or underborated) water. Consequently, the licensee adopted procedures to prevent the introduction of this water into the RCS. The two separate supply hoses were permanently coupled to their respective nozzles, with a common mating fixture attached to the pump. The set of hoses used for decontamination would be permanently tagged with a warning to personnel not to use those hoses for in-vessel operations. Procedures would require verification of proper hose alignment before use of the ultrahigh pressure pump.

- *Residual Unborated Water*. Following the use of this pump in the in-vessel decontamination mode, about 5 gallons of unborated water could remain in the system and could be injected into the RCS when the pump was used in the in-vessel cutting mode. Administrative control would require that the discharge from the in-vessel cutting nozzle be directed upward, away from the core and other suction lines, for a minimum of 5 minutes to allow the removal of the unborated water in the pump and lines. This precaution was required only following the use of the pump in the in-vessel decontamination mode. Additionally, the RCS boron concentration would be verified to be at or above 4950 ppm before use of the pump.

- *Conclusion*. The NRC concluded that, with these precautions, the potential for boron dilution due to the introduction of unborated water used with the ultrahigh pressure pump was acceptably low. Therefore, the potential for a boron dilution event that resulted in core criticality during core region defueling activity was acceptably low.

3.6.5 Use of Core Bore Machine for Bulk Defueling

- ***Purpose***. To use the core stratification sample acquisition (core bore) tooling as a defueling tool so that other defueling tools could more effectively break up and remove the remaining core debris. The core bore tool used a solid-faced bit to perforate the hard crust region of the core, down to the lower grid support structure, at multiple locations. The defueling work platform orientation system was used to position the drill mechanism with restrictions.

- ***Evaluation: Criticality***. (400) The licensee's safety evaluation stated that the use of the core bore tool for defueling operations was bounded by the licensee's safety evaluation report (401) for core stratification sample acquisition, except that drilling sites would be located using the defueling work platform locating system, as opposed to the theodolite system described in the licensee's safety evaluation report (402) for bulk defueling. The licensee concluded that the criticality discussions in the safety evaluation report for core stratification sample acquisition remained valid.

- ***NRC Review***. (403) Editor's Note: The NRC documented its review of the licensee's proposal in the agency's safety evaluation report (404) for bulk defueling.

3.6.6 Lower Core Support Assembly Defueling

- ***Purpose***. To dismantle and defuel the lower core support assembly (LCSA) and to partially defuel the lower reactor vessel head. Structural material removed from the LCSA included the (●) lower grid rib assembly; (●) lower grid forging; (●) distribution plate; and (●) in-core guide support plate. The lowest elliptical flow distributor and the gusseted in-core guide tubes would not be removed under this proposed activity. The use of the core bore machine, plasma arc cutting equipment, cavitating water jet, robot manipulators, automatic cutting equipment system, and other previously approved tools and equipment was included in this activity and associated safety evaluations.

- ***Evaluation: Criticality***. [405] The licensee's safety evaluation considered criticality concerns inside the reactor vessel during defueling operations and in the containment building basement during a postulated leak due to an instrument penetration.

  - *Reactor Vessel*. The licensee's previous safety evaluation report [406, 407] for bulk defueling and its criticality evaluation report [408] for the reactor coolant system (RCS) generally bounded criticality concerns during LCSA defueling. The plasma arc torch cooling system contained a design maximum of unborated coolant inventory of less than 4 gallons. However, the maximum amount of unborated water that could drain into the reactor vessel from the torch's coolant system during the operating position was determined to be no more than 3 gallons. This quantity of unborated water exceeded the limit of 2 gallons, which was established in the licensee's safety evaluation report [409] for potential boron dilution in the RCS.

    A criticality analysis was performed to demonstrate that the use of unborated coolant for the plasma arc torch would not pose a criticality safety concern. The criticality evaluation report [410] for using the plasma arc torch to cut the LCSA provided the basis, assumptions, and bounding fuel models that were used in the plasma arc torch criticality analysis. This analysis concluded that the plasma arc torch with a maximum drainable coolant system inventory of 3 gallons of unborated water [ff] could be used to dismantle the LCSA without developing a criticality safety concern within the reactor vessel. The above conclusion was based on the operational limitations listed in the criticality safety assessment report for using the plasma arc torch.

  - *Containment Building Basement*. The evaluation concluded that dropped objects would not cause reactor vessel leak; however, the potential effect of reactor vessel leak was considered for completeness. The evaluated scenario assumed the failure of one nozzle, resulting in a leak of 125 gallons per minute from the reactor vessel. The NRC's previous safety evaluation report [411] for the use of the core bore machine for dismantling the LCSA

[ff] Editor's Note: Although the torch's coolant system capacity was less than 4 gallons, the licensee determined that no more than 3 gallons of inventory could drain from the torch into the reactor vessel during LCSA cutting operations. Refer to Section 5.7.2 of this chapter that discusses use of the plasma arc torch to cut the LCSA for more details.

stated that such an event would be promptly detected and capabilities existed to maintain the RCS level. However, previous defueling tasks resulted in the deposit of a large accumulation of small fuel-bearing particles into the reactor vessel lower head. Consequently, the evaluation assumed that a portion of the fuel debris in the lower head would migrate into the basement cavity below the reactor vessel if a reactor vessel leak occurred.

– *Fuel Relocation*. An analysis was performed to evaluate the criticality safety concerns associated with the relocated fuel within the basement cavity under the reactor vessel.[gg] The results of this analysis, which was performed using optimum geometry and moderation, were provided in Appendix A to the licensee's safety evaluation report [412] for LCSA defueling. This analysis concluded that the maximum allowable fuel mass within the basement cavity under the reactor vessel, assuming a boron concentration of 2950 parts per million (ppm), was about 40,000 pounds. The analysis considered relocation of this much fuel to the basement cavity under the reactor vessel through the failure of a degraded in-core nozzle to be incredible. Consequently, the analysis concluded that maintaining the boron concentration of any water within the basement cavity under the reactor vessel above 2950 ppm would eliminate a criticality safety concern associated with relocating fuel to this cavity.

– *Administrative Controls*. To ensure that the boron concentration of the water in the basement cavity under the reactor vessel would always remain above 2950 ppm, a sample pump would be installed in this cavity. The following general approach would be used: (●) Enough borated water would be added to the basement cavity to increase the boron concentration within this cavity region to at least 3500 ppm. (●) The water in the basement cavity would be sampled weekly or whenever the containment building water level increased by more than 3 inches since the previous sample. This increase in level was assumed to result from other activities within the containment building (e.g., decontamination). (●) After sampling, if the boron concentration was below 3500 ppm, enough borated water would be added to increase the boron concentration to at least 3500 ppm. (●) The above steps would be repeated until the water level in the containment building basement reached the 283.40-foot elevation; at this point, the containment building's water level would be reduced and the basement cavity under the reactor vessel reborated to at least 3500 ppm.

  If boron concentration in the basement cavity decreased below 2950 ppm, core alterations would be suspended until the concentration was restored. Because this evaluation concluded that reactor vessel leakage would not occur, the controls discussed above were not required. However, for conservatism, before removal of the lower grid forging, these controls would be implemented.

gg Editor's Note: The LCSA defueling safety evaluation uses the term "reactor cavity." This section uses the term "basement cavity under the reactor vessel" in place of "reactor cavity." For the purposes of this summary, both terms refer to the same location, which is the containment building basement area below the reactor vessel.

- ***Evaluation: Boron Dilution***. [413] Boron dilution concerns during LCSA defueling were bounded by the evaluations provided in the licensee's previous evaluations. [414, 415, 416] To preclude the possibility of a hydraulic fluid leak leading to a possible critical configuration of fuel and moderator, all hydraulic fluid used with LCSA defueling tools, with the exception of the core bore machine, would be borated to at least 4350 ppm. The licensee's safety evaluation report [417] for the extended core stratification sample acquisition in the lower reactor vessel head region concluded that hydraulic fluid in the core bore machine did not need to be borated, given that there was no potential for the fluid to mix with the fuel.

- ***NRC Review: Criticality***. [418] The NRC's safety evaluation considered the consequences of the unlikely potential of a complete failure of an in-core instrument penetration due to a load drop event.

  - *Containment Building Basement*. The NRC stated that any fuel leakage into the cavity below the reactor vessel would be bounded by the 40,000-pound load used in the analysis. The NRC found that maintaining a boron concentration of 2950 ppm in the cavity water would provide adequate protection against potential criticality. Initial boration of the water (before removal of the lower grid forging) in the cavity below the reactor vessel to 3500 ppm and periodic sampling and analysis ensured that a minimum concentration of 2950 ppm would be maintained.

  - *Operational Restrictions*. Additional restrictions were applied while the plasma arc equipment was installed in the reactor vessel. Those restrictions associated with criticality included the following: (●) before removal of the lower grid forging, the boron concentration within the cavity beneath the reactor vessel would be increased to at least 3500 ppm and maintained at a minimum of 2950 ppm; (●) if the boron concentration fell below 2950 ppm, all activities that result in core alterations would be suspended; and (●) neutron monitoring instrumentation would be maintained as required by technical specifications.

    The NRC examined [419] the potential for criticality due to cooling water leakage during plasma arc cutting. The following additional restrictions would be required while the plasma arc cutting equipment was installed in the reactor vessel: (●) independent verification that the 15-gallon flush water tank was disconnected from the HE-200 cooling system when filling the 15-gallon tank; (●) independent verification that the 15-gallon flush water tank was disconnected from the HE-200 unit before moving the torch to within 1 foot of fuel-bearing areas, except for small isolated areas containing less than 22 pounds (10 kilograms) of fuel; and (●) active secondary purge gas flow when the 15-gallon flush tank was connected to the HE-200 unit.

    Removal of any gusseted in-core guide tubes and the elliptical flow distributor was not included in the scope of the NRC's safety evaluation. These two structures formed part of the protection for heavy load drops inside and over the reactor vessel during LCSA defueling.

3.6.7 Completion of Lower Core Support Assembly and Lower Head Defueling

- ***Purpose***. To remove the elliptical flow distributor and gusseted in-core guide tubes of the lower core support assembly (LCSA) and to defuel the reactor vessel lower head (RVLH).

- ***Evaluation: Criticality***. [420] The licensee's safety evaluation stated that its previous safety evaluation reports generally bounded criticality concerns during LCSA/RVLH defueling. These safety evaluations [421, 422, 423, 424] included: (●) bulk defueling; (●) criticality of the reactor coolant system; (●) limits of foreign materials allowed in the TMI-2 reactor coolant system during defueling activities; and (●) modifications to the plasma arc torch coolant system. Based on the results of these analyses, the evaluation concluded that the plasma arc torch, with a maximum drainable coolant system inventory of 3 gallons of unborated water, [hh] could be used to dismantle the LCSA, including the elliptical flow distributor head, without developing a criticality safety concern within the reactor vessel. This conclusion was based on operational limitations listed in the licensee's safety evaluation reports [425, 426, 427, 428] for the lower LCSA defueling, the criticality assessment for using the plasma arc torch to cut the lower LCSA, and the modification to the plasma arc torch coolant system.

- ***Evaluation: Boron Dilution***. [429] Boron dilution concerns during LCSA/RVLH defueling were bounded by the evaluations provided in the licensee's previous safety evaluation reports [430, 431] for bulk defueling and the potential for boron dilution of the reactor coolant system. To preclude the possibility of a hydraulic fluid leak that could lead to a possible critical configuration of fuel and moderator, all hydraulic fluid used with LCSA/RVLH defueling tools, with the exception of the core bore machine, would be borated to at least 4350 parts per million. The hydraulic fluid in the core bore machine did not need to be borated since there was no potential for the fluid to mix with the fuel. [432]

- ***NRC Review: Criticality***. [433] The NRC's safety evaluation considered: (●) existing damage of the RVLH and penetrations due to the core melt; (●) possible leak rates of damaged penetrations caused by defueling activities; and (●) potential for criticality in the reactor vessel cavity [ii] and containment building sump.

  - *Lower Head Damage*. Previous observation showed little damage to the in-core instrument penetrations and no damage to the RVLH. Since many of the penetrations and much of the lower head were hidden under core debris, the potential for damage could not be precluded. Therefore, the evaluation concluded that the potential area of interaction could either be intact or partially degraded. In addition, the evaluation determined that adequate forces could be generated from defueling equipment or a dropped load to shear an intact

[hh] Editor's Note: Refer to Section 5.7.2 of this chapter that discusses use of the plasma arc torch to cut the LCSA for more details about how the volume of drainable coolant was determined.

[ii] This area is the same area referred to as the "basement cavity under the reactor vessel" in the preceding section of this summary.

penetration, if the forces were applied horizontally or obliquely. The potential for damage (thinning) to the lower head due to jet impingement and ablation by molten material during the accident was limited to the area beneath fuel assemblies R6 and R7 and the area outside the core baffle plates.

- *Lower Head Penetration Leakage*. The NRC's evaluation discussed two potential, though unlikely, events; the consequences of those events; and the methods to compensate for the effects of those events. In the unlikely event of a complete shear of a penetration, an annular gap would exist between the in-core instrument string and the lower head. The maximum leakage through this annular gap would be 0.4 gallon per minute per sheared penetration. This leakage was well within the licensee's capability to make up water to the reactor coolant system using gravity feed or pumping. If an unspecified mechanism provided adequate force to push the instrument string through the lower head, a 1-inch-diameter hole and a leak of 120 gallons per minute could result. Active pumping of borated water would be required to maintain the reactor vessel level. Maintaining the reactor vessel level would not be required to maintain subcriticality or to protect the health and safety of the public. However, radiation and airborne activity could limit access to the containment building, and fuel debris could be flushed to the reactor vessel cavity.

- *Cavity and Sump Criticality*. The NRC evaluated the potential for criticality in the reactor vessel cavity and sump under these conditions. The licensee's analysis indicated that 2950 parts per million of boron in the water in this cavity would maintain subcriticality with an effective neutron multiplication less than 0.99. The NRC found this analysis to be conservative, with effective neutron multiplication likely to be significantly less than 0.99. The licensee's method of initial boration and weekly sampling of the water in the reactor vessel cavity was acceptable to the NRC. Fuel particle size and total mass assumptions were kept within the bounds analyzed by the licensee and the NRC by restricting activities near the area of potential ablation of the lower head. This precluded the creation of a potential leakage path larger than 1 inch. Significant damage to the lower head during the TMI-2 accident was unlikely. After this could be confirmed visually, these restrictions would not be needed.

3.6.8 Upper Core Support Assembly Defueling

- ***Purpose***. To cut and move the baffle plates and to defuel the upper core support assembly (UCSA). This evaluation addressed the following activities: (●) cutting the baffle plates for later removal; (●) removing bolts from the baffle plates; (●) removing the baffle plates; and (●) removing core debris from the baffle plates and core former plates.

- ***Evaluation: Criticality***. [434] The licensee's safety evaluation considered the potential for criticality in the reactor vessel and containment building basement.

- *Reactor Vessel*. The licensee's safety evaluation stated that its previous safety evaluation reports generally bounded criticality concerns during UCSA defueling. These safety evaluations [435, 436, 437, 438, 439] included: (●) bulk defueling; (●) lower core support assembly

(LCSA) defueling; (●) criticality of the reactor coolant system; (●) criticality safety assessment for use of the plasma arc torch to cut the UCSA baffle plates and the core support shield; and (●) the potential for boron dilution of the reactor coolant system.

Based on the results of these analyses, the evaluation concluded that the plasma arc torch, with a maximum drainable coolant system inventory of 3 gallons of unborated water, [jj] could be used to dismantle the LCSA and the elliptical flow distributor head without developing a criticality safety concern within the reactor vessel. This conclusion was based on operational limitations listed in the licensee's safety evaluation reports (440, 441, 442, 443, 444) for the lower LCSA defueling; the criticality assessment for using the plasma arc torch to cut the lower LCSA; modifications to the plasma arc torch coolant system; and criticality safety assessment for use of the plasma arc torch to cut the UCSA baffle plates and the core support shield.

Subcriticality was ensured by the following measures: (●) establishment of the reactor coolant system boron concentration at greater than 4350 parts per million (ppm); (●) monitoring of boron concentration and inventory levels; and (●) isolation of potential deboration pathways. To prevent overheating and potential criticality, systems were in place to add borated cooling water to the core in the event of a nonisolatable leak from the reactor vessel. Additional borated water was added to the cavity beneath the reactor vessel to increase the boron concentration above 3500 ppm, as specified in the safety evaluations associated with the LCSA and lower head defueling. The evaluation determined that this action would ensure that a criticality event external to the vessel was not credible. The evaluation also concluded that the introduction of unborated water from the torch cooling system would not create the potential for a criticality. This conclusion was based on no more than 3.5 gallons of unborated water being able to inadvertently drain into the reactor vessel, as described in the licensee's criticality safety assessment (445) for using the plasma arc torch to cut the upper core support structure.

- *Containment Building Basement*. The potential for a criticality event in the containment building basement was previously addressed in the licensee's safety evaluations (446, 447) for LCSA defueling and containment building sump criticality. The controls discussed in those safety evaluations that ensured subcriticality for potential leakage into the cavity of the reactor vessel [kk] were also maintained during UCSA defueling. Therefore, the evaluation concluded that criticality would be precluded. Monitoring of the containment building basement would continue during UCSA defueling.

- ***Evaluation: Boron Dilution***. (448) The licensee's safety evaluation concluded that boron dilution concerns during UCSA defueling were bounded by the evaluations in the licensee's safety evaluation reports for bulk defueling (449) and potential for boron dilution of the reactor

---

[jj] Editor's Note: Refer to Section 5.7.2 of this chapter that discusses use of the plasma arc torch to cut the LCSA for more details about how the volume of drainable coolant was determined.

[kk] Editor's Note: This section refers to "the cavity of the reactor vessel" and the "containment building cavity." These terms refer to the same area, which is the same as the area referred to as the "basement cavity under the reactor vessel" in the preceding section on LCSA defueling.

coolant system. [450] To preclude the possibility of a hydraulic fluid leak that could lead to a possible critical configuration of fuel and moderator, all hydraulic fluid used with UCSA defueling tools would be borated to at least 4350 ppm.

- ***NRC Review: Criticality/ Boron Dilution***. [451] The NRC determined that the remaining fuel was well characterized, including its location. Criticality controls and boron dilution controls remained as requirements that had been imposed in previous safety evaluations. The known safety margin to criticality was increased due to defueling progress and the knowledge that the fuel was in a significantly less than optimal geometry.

3.7 Evaluations for Waste Management

3.7.1 EPICOR II (NA)

3.7.2 Submerged Demineralizer System

3.7.2.1 Submerged Demineralizer System Operations

- ***Purpose***. To decontaminate the containment building sump water and reactor coolant system (RCS) water using the submerged demineralizer system (SDS), followed by effluent polishing with the EPICOR II system.

- ***Evaluation: Criticality***. [452, 453, 454] The licensee evaluated the potential to accumulate a critical mass in the SDS ion exchange vessels during the processing of accident-generated water from the containment building sump and RCS (for the first time). Further, the licensee included criticality prevention measures in the design and operations of the SDS.

  - *Source Term*. In August 1979, the licensee obtained a sample of the water in the containment building sump. Analysis of that sample revealed that it contained 4.55×10^{-3} microgram per milliliter of solid uranium, 2.8×10^{-2} microgram per milliliter of dissolved uranium, and 3.3×10^{-5} microgram per milliliter of plutonium. Based on processing 600,000 gallons of sump water, assuming uniform distribution of both radionuclides, the total inventory of uranium in the sump would be about 73 grams, and the total inventory of plutonium in the sump would be about 0.0749 grams.

  - *Sump Water Processing*. The evaluation assumed that all uranium and plutonium were deposited in one SDS ion exchange vessel, which the evaluation considered highly unlikely. During SDS sump water processing, one vessel would contain 2.96 weight percent uranium. A critical mass of this weight percent uranium would require about 2.2 kilograms of uranium-235. Since only 73 grams of the material were present, the evaluation concluded that criticality was not possible. Furthermore, for criticality to occur, 450 grams of plutonium-239 would have to be present concurrently (nonuniform slurry) with 700 grams of uranium-235 and 520 grams of uranium-233. Since only 73 grams of uranium were available, criticality was not possible. Therefore, the licensee concluded that criticality from the presence of the radioisotopes alone or mixed was not possible.

- *RCS Water Processing*. The RCS contained a concentration of 100 parts per billion of uranium and 0.24 parts per billion of plutonium. These concentrations yielded a total of 37 grams of mixed uranium and plutonium. Therefore, based on the previous discussions concerning criticality requirements, criticality was not possible.

- *Criticality Prevention*. The licensee concluded that, based on the small quantities of fissile material in the sump and RCS, criticality in the spent SDS vessel was not possible. Nevertheless, the licensee instituted the following additional measures to prevent criticality.

- *Prefilters*. Two sand filters were used to filter out solids in the untreated contaminated water before the water was processed by ion exchangers. Mixed uniformly with the sand was about 6 pounds of borosilicate glass, which was at least 22 weight percent boron. The purpose of the borosilicate was to prevent the possibility of criticality should any fuel fines be transported in the SDS letdown piping.

- *Fuel Debris Monitoring*. To prevent a criticality accident during RCS water processing, transuranic monitoring was provided at the reactor coolant bleed tanks to detect the presence of fuel debris. This monitoring consisted of a two-channel analyzer to detect the presence of cerium-144/praseodymium-144. These isotopes were extremely insoluble and provided a good indication of the presence of fuel debris.

- *RCS Boron Concentration*. During the processing of the RCS, the boron concentration would be maintained to greater than 3500 parts per million to maintain the core in a noncritical safe condition. This requirement was adopted as a recovery technical specification.

- ***NRC Review***. The original NRC safety evaluation report (NUREG-0796) [455] did not specifically address this topic. However, the NRC asked the licensee during a meeting to evaluate the potential issue of criticality in the SDS ion exchange vessels. No other correspondence concerning criticality was located.

3.7.2.2 Submerged Demineralizer System Liner Recombiner and Vacuum Outgassing System (NA)

3.8 Endnotes

Reference citations are exact filenames of documents on the Digital Versatile Discs (DVDs) to NUREG/KM-0001, Supplement 1, [456] "Three Mile Island Accident of 1979 Knowledge Management Digest: Recovery and Cleanup," issued June 2016, DVD document filenames start with a full date (YYY-MM-DD) or end with a partial date (YYYY-DD). Citations for references that are not on a DVD begin with the author organization or name(s), with the document title in quotation marks. Contents of the DVDs are available at the Idaho National Laboratory Web site (https://tmi2kml.inl.gov).

---

[1] (1984-11-08) GPU, Criticality Report, Reactor Coolant System

[2] (1979-05-14) NRC Review, Recriticality Potential of TMI-2 Core

[3] ORNL-CSD-TM-106, Criticality Analyses of Disrupted Core Models of Three Mile Island Unit 2 (1979-12)

[4] (1982-06-01) BAW, Criticality Calculations to Support Recovery Activities Through Head Removal

[5] (1982-10-01) BAW, Verification of Criticality Calculations for TMI-2 Recovery Operations Through Head Removal, Addend 1

[6] (1983-12-01) BAW, Criticality Analysis for Heavy Load Drop Accident in Support of Recovery Through RV Head Removal

[7] GEND-026, ADDITION OF SOLUBLE AND INSOLUBLE NEUTRON ABSORBERS TO THE RCS OF TMI-2 (1984-07)

[8] (1990-02-22) GPU, Defueling Completion Report, Final Submittal (see Appendix B)

[9] Clark, H.K., "Critical and Safe Masses and Dimensions of Lattices of U and UO2 Rods in Water," DP-1014, Savannah River Laboratory, February 1966, Washington [Available at osti.gov]

[10] (1984-11-08) GPU, Criticality Report, Reactor Coolant System

[11] GPU Nuclear Corporation, "Report on Limits of Foreign Materials Allowed in the TMI-2 Reactor Coolant System During Defueling Activities," Rev. 1, 15737-2-N09-002, October 1985 [ADAMS Accession No. ML19302G716]

[12] (1987-11-30) GPU Criticality Assessment for Using Plasma Arc Torch to Cut Lower Core Support Assembly, Rev. 1

[13] Oak Ridge National Labratory, "Nuclear Safety Guide," TID-7016, Revision 2, June 1978 [Available at osti.gov; see page 27]

[14] GEND-026, ADDITION OF SOLUBLE AND INSOLUBLE NEUTRON ABSORBERS TO THE RCS OF TMI-2 (1984-07)

[15] (1983-02-18) GPU Safety Evaluation, Polar Crane Load Test

[16] (1980-02-13) NRC Order (2-11-1980), Provides TMI-2 Recovery Technical Specifications

[17] USNRC, "Safety Evaluation and Environmental Assessment, Three Mile Island Nuclear Station, Unit No. 2," NUREG-0647, February 11, 1980

[18] (1982-06-01) BAW, Criticality Calculations to Support Recovery Activities Through Head Removal

[19] USNRC, "Control of Heavy Loads at Nuclear Power Plants," NUREG-0612, July 1980

[20] USNRC, "Control of Heavy Loads," Generic Letter 80-113, December 22, 1980

[21] USNRC, "Control of Heavy Loads," Generic Letter 81-07, February 3, 1981

[22] (1983-02-18) GPU Safety Evaluation, Polar Crane Load Test

[23] GEND-026, ADDITION OF SOLUBLE AND INSOLUBLE NEUTRON ABSORBERS TO THE RCS OF TMI-2 (1984-07)

[24] (1985-05-24) BAW, TMI-2 Defueling Canisters Final Design Technical Report

[25] (1985-09-10) GPU Technical Evaluation, Defueling Canister TER (re 08-15-1985)

[26] EGG-TMI-6156-R1, TMI-2 Fuel Canister Interface Requirements for INEL (1984-06)

[27] (1984-11-08) GPU, Criticality Report, Reactor Coolant System

[28] GPU Nuclear Corporation, "Report on Limits of Foreign Materials Allowed in the TMI-2 Reactor Coolant System During Defueling Activities," Rev. 1, 15737-2-N09-002, October 1985 [ADAMS Accession No. ML19302G716]

[29] (1986-01-17) GPU, Plan to Use Modified Debris Canisters for removing End Fittings, etc
[30] (1979-05-14) NRC Review, Recriticality Potential of TMI-2 Core
[31] ORNL-CSD-TM-106, Criticality Analyses of Disrupted Core Models of Three Mile Island Unit 2 (1979-12)
[32] (1980-04-04) NRC Review, A Further Evaluation of Risk of Recriticality at TMI-2 (re 03-06-1980)
[33] NUREG-0683, Vol. 1, PEIS-Decontamination and Disposal of Radioactive Wastes Resulting from TMI-2 (1981-03)
[34] (1979-05-14) NRC Review, Recriticality Potential of TMI-2 Core
[35] ORNL-CSD-TM-106, Criticality Analyses of Disrupted Core Models of Three Mile Island Unit 2 (1979-12)
[36] ANL-NRC-RAS-81-1, Potential for and Consequences of Recriticality During Cleanup and Defueling at TMI-2 (1981-06)
[37] (1982-06-01) BAW, Criticality Calculations to Support Recovery Activities Through Head Removal
[38] (1982-10-01) BAW, Verification of Criticality Calculations for TMI-2 Recovery Operations Through Head Removal, Addend 1
[39] (1982-06-01) BAW, Criticality Calculations to Support Recovery Activities Through Head Removal
[40] (1983-12-01) BAW, Criticality Analysis for Heavy Load Drop Accident in Support of Recovery Through RV Head Removal
[41] EGG-TMI-6156-R1, TMI-2 Fuel Canister Interface Requirements for INEL (1984-06)
[42] GEND-026, ADDITION OF SOLUBLE AND INSOLUBLE NEUTRON ABSORBERS TO THE RCS OF TMI-2 (1984-07)
[43] (1984-11-08) GPU, Criticality Report, Reactor Coolant System
[44] (1985-03-15) NRC Safety Evaluation, Reactor Coolant System Criticality Report (re 11-08-1984)
[45] (1985-09-10) GPU Technical Evaluation, Defueling Canister TER (re 08-15-1985)
[46] (1987-05-07) GPU Technical Evaluation, Defueling Canisters, Rev. 3
[47] (1985-09-27) GPU Hazards Analysis, Potential for Boron Dilution of Reactor Coolant System, Rev. 2
[48] GPU Nuclear Corporation, "Report on Limits of Foreign Materials Allowed in the TMI-2 Reactor Coolant System During Defueling Activities," Rev. 1, 15737-2-N09-002, October 1985 [ADAMS Accession No. ML19302G716]
[49] (1984-11-08) GPU, Criticality Report, Reactor Coolant System
[50] USNRC, "Review of TMI-2 Foreign Materials Criticality Report," Internal NRC Memorandum, December 2, 1985 [ADAMS Accession No. ML19337B690]
[51] GPU Nuclear Corporation, "Technical Plan: Ex-RCS Criticality Safety," GPU/TMI-132, Rev. 1, November 1985
[52] ORNL-CSD-TM-222, TMI-2 Criticality Studies, Lower-Vessel Rubble and Analytical Benchmarking (1985-12)
[53] DOE-NCT-01, Review of the State of Criticality of the TMI-2 Core and Reactor Vessel (1987-04)
[54] (1989-02-13) GPU Criticality Safety Assessment for Increasing the TMI-2 Safety Fuel Mass Limit, Rev. 0
[55] DOE-SNF-REP-084, TMI Fuel Characteristics for Disposal Criticality Analysis (2003-09)
[56] (1982-06-15) GPU Safety Evaluation, Axial Power Shaping Rod Insertion Test
[57] (1982-06-15) GPU Safety Evaluation, Axial Power Shaping Rod Insertion Test
[58] (1982-06-15) GPU Safety Evaluation, Axial Power Shaping Rod Insertion Test

[59] (1980-04-04) NRC Review, A Further Evaluation of Risk of Recriticality at TMI-2 (re 03-06-1980)
[60] (1979-05-14) NRC Review, Recriticality Potential of TMI-2 Core
[61] ORNL-CSD-TM-106, Criticality Analyses of Disrupted Core Models of Three Mile Island Unit 2 (1979-12)
[62] (1982-05-17) Order Amendment
[63] (1980-04-04) NRC Review, A Further Evaluation of Risk of Recriticality at TMI-2 (re 03-06-1980)
[64] (1982-07-06) GPU Safety Evaluation, Insertion Camera Through Reactor Vessel Leadscrew Opening, Rev. 2
[65] (1982-06-01) BAW, Criticality Calculations to Support Recovery Activities Through Head Removal
[66] (1982-07-06) GPU Safety Evaluation, Insertion Camera Through Reactor Vessel Leadscrew Opening, Rev. 2
[67] (1982-07-13) NRC Review, Control Rod Drive Mechanism Quick Look Camera Inspection (re 07-06-1982) (2)
[68] (1982-07-13) NRC Review, Control Rod Drive Mechanism Quick Look Camera Inspection (re 07-06-1982) (2)
[69] (1983-05-19) GPU Safety Analysis, Radiation Characterization Under Reactor Vessel Head (No Polar Crane), Rev. 0
[70] (1982-07-06) GPU Safety Evaluation, Insertion Camera Through Reactor Vessel Leadscrew Opening, Rev. 2
[71] (1982-06-01) BAW, Criticality Calculations to Support Recovery Activities Through Head Removal
[72] (1983-05-19) GPU Safety Analysis, Radiation Characterization Under Reactor Vessel Head (No Polar Crane), Rev. 0
[73] (1982-07-06) GPU Safety Evaluation, Insertion Camera Through Reactor Vessel Leadscrew Opening, Rev. 2
[74] (1983-07-13) NRC Review, Reactor Vessel Underhead Characterization (re Various Letters)
[75] (1983-07-20) GPU Safety Evaluation, Underhead Characterization, Core Sampling Addendum
[76] USNRC, “Three Mile Island Accident of 1979 Knowledge Management Digest: Recovery and Cleanup,” NUREG/KM-0001, Supplement 1, June 2016 [Available at nrc.gov]
[77] (1983-07-20) GPU Safety Evaluation, Underhead Characterization, Core Sampling Addendum
[78] (1982-07-06) GPU Safety Evaluation, Insertion Camera Through Reactor Vessel Leadscrew Opening, Rev. 2
[79] (1982-06-01) BAW, Criticality Calculations to Support Recovery Activities Through Head Removal
[80] Savannah River Laboratory, “Critical and Safe Masses and Dimensions of Lattices of U and UO_2 Rods in Water,” TID-4500, February 1966 [Available at ntrl.ntis.gov]
[81] (1983-07-20) GPU Safety Evaluation, Underhead Characterization, Core Sampling Addendum
[82] (1983-08-19) NRC Safety Evaluation, Addendum to the Underhead Characterization Study (re 07-30-1983)
[83] (1986-06-11) GPU Safety Evaluation, Core Stratification Sample Acquisition, Rev. 4
[84] (1985-09-27) GPU Hazards Analysis, Potential for Boron Dilution of Reactor Coolant System, Rev. 2

[85] (1985-09-27) GPU Hazards Analysis, Potential for Boron Dilution of Reactor Coolant System, Rev. 2

[86] (1984-11-08) GPU, Criticality Report, Reactor Coolant System

[87] (1985-10-10) GPU Safety Evaluation, Early Defueling of the TMI-2 Reactor Vessel, Rev. 4

[88] GPU Nuclear Corporation, "Report on Limits of Foreign Materials Allowed in the TMI-2 Reactor Coolant System During Defueling Activities," Rev. 1, 15737-2-N09-002, October 1985 [ADAMS Accession No. ML19302G716]

[89] (1986-05-05) NRC Safety Evaluation, Core Stratification Sample Acquisition SER, Rev. 1 (re 08-30-1985, 12-31-1985)

[90] GPU Nuclear Corporation, "Report on Limits of Foreign Materials Allowed in the TMI-2 Reactor Coolant System During Defueling Activities," Rev. 1, 15737-2-N09-002, October 1985 [ADAMS Accession No. ML19302G716]

[91] (1986-05-05) NRC Safety Evaluation, Core Stratification Sample Acquisition SER, Rev. 1 (re 08-30-1985, 12-31-1985)

[92] (1986-05-05) NRC Safety Evaluation, Core Stratification Sample Acquisition SER, Rev. 1 (re 08-30-1985, 12-31-1985)

[93] (1989-08-18) GPU Safety Evaluation, Remove Metallurgical Samples from Reactor Vessel SER

[94] (1989-10-20) GPU Safety Evaluation, Remove Metallurgical Samples from Vessel, Rev. 1 (re 08-18-1989) (effective pages)

[95] (1986-05-15) GPU Safety Evaluation, Defueling SER, Rev. 10

[96] (1988-01-18) GPU Safety Evaluation, Lower Core Support Assembly Defueling, Rev. 2

[97] (1989-08-18) GPU Safety Evaluation, Remove Metallurgical Samples from Reactor Vessel SER

[98] (1989-10-20) GPU Safety Evaluation, Remove Metallurgical Samples from Vessel, Rev. 1 (re 08-18-1989) (effective pages)

[99] GPU Nuclear Corporation, "Report on Limits of Foreign Materials Allowed in the TMI-2 Reactor Coolant System During Defueling Activities," Rev. 1, 15737-2-N09-002, October 1985 [ADAMS Accession No. ML19302G716]

[100] (1989-11-28) NRC Safety Evaluation, Reactor Vessel Lower Head Metallurgical Sampling (re 08-18-1989)

[101] (1988-06-27) GPU Safety Evaluation, Completion of Lower Core Support and Lower Head Defueling, Rev. 1

[102] (1988-12-01) NRC Safety Evaluation, Lower Core Support and Lower Head Defueling (re 06-06-1988)

[103] (1989-11-28) NRC Safety Evaluation, Reactor Vessel Lower Head Metallurgical Sampling (re 08-18-1989)

[104] USNRC, "Review of TMI-2 Foreign Materials Criticality Report," Internal NRC Memorandum, December 2, 1985 [ADAMS Accession No. ML19337B690]

[105] (1987-11-30) GPU Criticality Assessment for Using Plasma Arc Torch to Cut Lower Core Support Assembly, Rev. 1

[106] (1986-03-28) GPU Safety Evaluation, Reactor Building Decontamination and Dose Reduction Activities for 1986, Rev. 0

[107] (1986-09-05) GPU Safety Evaluation, Reactor Building Sump Criticality Safety

[108] (1985-08-08) Order Amendment (also exemption from 10CFR50 AppA)

[109] GPU Nuclear Corporation, "Report on Limits of Foreign Materials Allowed in the TMI-2 Reactor Coolant System During Defueling Activities," Rev. 1, 15737-2-N09-002, October 1985 [ADAMS Accession No. ML19302G716]

[110] (1984-02-03) NRC Safety Evaluation, Containment Decontamination and Dose Reduction (re 09-29, 01-23-1983)
[111] (1986-09-11) NRC Review, Reactor Building Sump Criticality Safety Evaluation Report (re various letters)
[112] GPU Nuclear Corporation, "Report on Limits of Foreign Materials Allowed in the TMI-2 Reactor Coolant System During Defueling Activities," Rev. 1, 15737-2-N09-002, October 1985 [ADAMS Accession No. ML19302G716]
[113] (1985-08-08) Order Amendment (also exemption from 10CFR50 AppA)
[114] (1982-11-22) GPU, Inform Plans to Refill RCS
[115] (1982-12-14) GPU, Changes in Plan to Vent Pressurizer During RCS Refill (re 11-22-1982)
[116] (1983-02-18) GPU Safety Evaluation, Polar Crane Load Test
[117] (1982-06-01) BAW, Criticality Calculations to Support Recovery Activities Through Head Removal
[118] NUREG-0557, Evaluation of Long-Term Post-Accident Core Cooling of TMI-2 (1979-05)
[119] (1983-02-18) GPU Safety Evaluation, Polar Crane Load Test
[120] (1983-11-18) NRC Safety Evaluation, Reactor Building Polar Crane Load Test
[121] (1982-05-17) Order Amendment
[122] (1983-09-29) GPU, Plans for First Pass Stud Detensioning for Head Removal
[123] (1984-01-30) GPU Response to NRC, First Pass Stud Detensioning SER (re 09-29, 12-06-1983)
[124] (1983-12-06) NRC Review, First Pass Stud Detensioning for Head Removal (re 09-29-1983)
[125] (1984-02-17) NRC Safety Evaluation, First Pass Vessel Stud Detensioning (re 09-29, 12-06-1983, 01-30-1984)
[126] (1984-03-09) GPU Safety Evaluation, Head Removal, Rev. 5
[127] (1982-06-01) BAW, Criticality Calculations to Support Recovery Activities Through Head Removal
[128] (1982-10-01) BAW, Verification of Criticality Calculations for TMI-2 Recovery Operations Through Head Removal, Addend 1
[129] (1984-03-09) GPU Safety Evaluation, Head Removal, Rev. 5
[130] (1983-05-19) GPU Safety Analysis, Radiation Characterization Under Reactor Vessel Head (No Polar Crane), Rev. 0
[131] (1984-07-17) NRC Safety Evaluation, Reactor Pressure Vessel Head Lift (re -3-09-1984)
[132] (1984-07-17) NRC Safety Evaluation, Reactor Pressure Vessel Head Lift (re -3-09-1984)
[133] USNRC, "Control of Heavy Loads at Nuclear Power Plants," NUREG-0612, July 1980 [Available at nrc.gov]
[134] (1986-06-02) GPU Safety Evaluation, Heavy Load Handling Inside Containment, Rev. 3
[135] (1984-03-09) GPU Safety Evaluation, Head Removal, Rev. 5
[136] (1983-02-18) GPU Safety Evaluation, Polar Crane Load Test
[137] (1985-09-10) GPU Technical Evaluation, Defueling Canister TER (re 08-15-1985)
[138] (1985-04-19) GPU Safety Evaluation, Heavy Load Handling Over Reactor Vessel SER
[139] (1985-11-15) NRC Safety Evaluation, Heavy Load Handling Inside Containment (re 09-11-1985)
[140] (1987-07-23) NRC Review, Heavy Load Handling Inside Containment (re various letters)
[141] (1985-10-30) NRC (Internal), Criticality Evaluation of Loaded Canister Dropping its Load in Another Canister.pdf
[142] (1985-04-19) GPU Safety Evaluation, Heavy Load Handling Over Reactor Vessel SER
[143] (1984-11-08) GPU, Criticality Report, Reactor Coolant System
[144] (1985-09-27) GPU Hazards Analysis, Potential for Boron Dilution of Reactor Coolant System, Rev. 2

[145] (1985-05-02) NRC Safety Evaluation, Heavy Load Handling Over the TMI-2 Reactor Vessel (re 04-19-1985)
[146] (1985-08-08) Order Amendment (also exemption from 10CFR50 AppA)
[147] (1984-06-18) GPU Safety Evaluation, Preparatory Activities for Plenum Assembly Removal
[148] (1984-03-09) GPU Safety Evaluation, Head Removal, Rev. 5
[149] (1984-06-18) GPU Safety Evaluation, Preparatory Activities for Plenum Assembly Removal
[150] (1984-03-09) GPU Safety Evaluation, Head Removal, Rev. 5
[151] (1984-05-31) GPU Safety Evaluation, Internals Indexing Fixture Processing System, Rev. 1
[152] (1984-07-31) NRC Review, Preparatory Activities for Plenum Assembly Removal SER (re 06-18-1984)
[153] (1984-09-14) NRC Safety Evaluation, Plenum Assembly Removal (re various letters)
[154] (1982-07-13) NRC Review, Control Rod Drive Mechanism Quick Look Camera Inspection (re 07-06-1982)
[155] (1983-07-13) NRC Review, Reactor Vessel Underhead Characterization (re Various Letters)
[156] (1984-09-14) NRC Safety Evaluation, Plenum Assembly Removal (re various letters)
[157] (1984-07-17) NRC Safety Evaluation, Reactor Pressure Vessel Head Lift (re -3-09-1984)
[158] (1984-09-14) NRC Safety Evaluation, Plenum Assembly Removal (re various letters)
[159] (1984-07-24) NRC Safety Evaluation, Internals Index Fixture Water Processing System (re 05-31, 07-16-1984)
[160] (1985-05-01) GPU Safety Evaluation, Plenum Removal and Transfer, Rev. 3
[161] (1984-11-08) GPU, Criticality Report, Reactor Coolant System
[162] (1985-09-27) GPU Hazards Analysis, Potential for Boron Dilution of Reactor Coolant System, Rev. 2
[163] (1985-05-01) GPU Safety Evaluation, Plenum Removal and Transfer, Rev. 3
[164] (1985-05-07) NRC Safety Evaluation, Lifting, Transfer, and Storage of Plenum Assembly (re various letters)
[165] (1984-07-17) NRC Safety Evaluation, Reactor Pressure Vessel Head Lift (re -3-09-1984)
[166] (1985-03-15) NRC Safety Evaluation, Reactor Coolant System Criticality Report (re 11-08-1984)
[167] (1984-09-14) NRC Safety Evaluation, Plenum Assembly Removal (re various letters)
[168] GEND-039, TMI-2 Technical Information and Examination Program Annual Report 1983 (1984-04)
[169] HEDL-SA-2834, Neutron dosimetry measurements for fuel debris assessment of TMI-2 demineralizer-A (1982-12)
[170] (1983-02-25) GPU Safety Evaluation, Makeup and Purification Demineralizer Resin Sampling
[171] (1984-07-19) GPU Safety Evaluation, Makeup and Purification Cesium Elution SER
[172] USDOE, "TMI-2 Purfication Deminerlizer Resin Study," GEND-INK-013, May 1984 [Available at osit.gov] (Refer the the attached Appendix E report, Hanford Energy Development Laboratory, "Resin and Debris Removal System Conceptual Design," HEDL-7335, March 1983)
[173] (1984-09-19) NRC Safety Evaluation, Makeup and Purification Demineralizer Cesium Elution (re 07-19, 08-29-1984)
[174] (1984-05-31) GPU Safety Evaluation, Internals Indexing Fixture Processing System, Rev. 1
[175] (1982-07-06) GPU Safety Evaluation, Insertion Camera Through Reactor Vessel Leadscrew Opening, Rev. 2
[176] (1984-03-09) GPU Safety Evaluation, Head Removal, Rev. 5
[177] (1984-07-24) NRC Safety Evaluation, Internals Index Fixture Water Processing System (re 05-31, 07-16-1984)

[178] (1984-07-24) NRC Safety Evaluation, Internals Index Fixture Water Processing System (re 05-31, 07-16-1984)

[179] (1989-05-12) GPU Technical Evaluation, Defueling Water Cleanup System, Rev. 12

[180] (1987-05-07) GPU Technical Evaluation, Defueling Canisters, Rev. 3

[181] (1989-05-12) GPU Technical Evaluation, Defueling Water Cleanup System, Rev. 12

[182] (1989-05-12) GPU Technical Evaluation, Defueling Water Cleanup System, Rev. 12

[183] (1984-03-09) GPU Safety Evaluation, Head Removal, Rev. 5

[184] (1985-05-17) GPU Safety Evaluation, Operation of Internal Indexing Fixture Processing System, Rev. 3

[185] (1985-09-27) GPU Hazards Analysis, Potential for Boron Dilution of Reactor Coolant System, Rev. 2

[186] (1984-03-09) GPU Safety Evaluation, Head Removal, Rev. 5

[187] (1985-08-06) NRC Safety Evaluation, Defueling Water Cleanup System (re various letters)

[188] (1984-11-08) GPU, Criticality Report, Reactor Coolant System

[189] (1987-05-21) GPU Safety Analysis, Defueling Water System Cross-Connect to Reactor Vessel Cleanup System, Rev. 0

[190] (1986-12-31) GPU, Use of Different Coagulants (re 12-15-1986)

[191] (1987-02-20) GPU Safety Evaluation, Criticality Safety Evaluation for Coagulants (re 12-15, 12-31-1986)

[192] (1987-05-21) GPU Safety Analysis, Defueling Water System Cross-Connect to Reactor Vessel Cleanup System, Rev. 0

[193] (1985-04-23) Order Amendment (see ROP-27)

[194] (1987-06-11) NRC Review, DWCS Cross-Connect to Reactor Vessel Cleanup System (re 05-21-1987)

[195] (1986-12-01) GPU Safety Evaluation, TMI-2 Temporary Reactor Vessel Filtration System, Rev. 3 (Missing pages)

[196] (1984-11-08) GPU, Criticality Report, Reactor Coolant System

[197] Pacific Nuclear Systems, Inc., “Safety Analysis Report for the NuPac 125-B Fuel Shipping Cask,” Rev. 0, June 1985 [ADAMS Accession No. ML20132E700]

[198] (1986-12-01) GPU Safety Evaluation, TMI-2 Temporary Reactor Vessel Filtration System, Rev. 3 (Missing pages)

[199] (1986-02-07) NRC Safety Evaluation, Temporary Reactor Vessel Water Filtration System

[200] (1986-03-18) NRC Review, Restricted Operations of Temporary Vessel Filtration System (re various letters)

[201] (1986-03-06) GPU Safety Evaluation, Temporary Reactor Vessel Filtration System, Rev. 1

[202] (1984-11-08) GPU, Criticality Report, Reactor Coolant System

[203] (1985-03-15) NRC Safety Evaluation, Reactor Coolant System Criticality Report (re 11-08-1984)

[204] (1986-04-14) NRC Review, Temporary Reactor Vessel Filtration System, Rev. 2 (re various letters)

[205] (1986-04-14) GPU Safety Evaluation, Temporary Reactor Vessel Filtration System, Rev. 2

[206] (1984-11-08) GPU, Criticality Report, Reactor Coolant System

[207] (1986-02-21) NRC Review, Modification of Fines and Debris Vacuum System (re various letters)

[208] (1986-11-17) GPU Safety Evaluation, TMI-2 Temporary Reactor Vessel Filtration System, Rev. 3

[209] (1986-06-30) GPU Safety Evaluation, Operation of Filter-Aid Feed System

[210] (1986-06-30) GPU Safety Evaluation, Operation of Filter-Aid Feed System

[211] (1986-07-09) NRC Review, Operation of Filter-Aid Feed System, Use of Diatomaceous Earth (re various letter)
[212] (1987-02-20) GPU Safety Evaluation, Criticality Safety Evaluation for Coagulants (re 12-15, 12-31-1986)
[213] (1984-11-08) GPU, Criticality Report, Reactor Coolant System
[214] (1986-12-15) GPU Safety Evaluation, Addition of Coagulants to the Reactor Coolant System, Rev. 0
[215] (1986-12-31) GPU, Use of Different Coagulants (re 12-15-1986)
[216] Brookhaven National Laboratory, "Neutron Cross Section," BNL-325, Vol. 2, Third Edition, January 1976
[217] Thomas, J.T., "The Effect of Boron and Gadolinium Concentrations on Calculated Neutron Multiplication Factor of U(3)02 Fuel Pins in Optimum Geometries," Oak Ridge National Laboratory, ORNL/CSD/TM-218, October 1984 [Available at osti.gov]
[218] (1987-03-27) NRC Review, Criticality Safety Evaluation for Coagulants (re 12-11-1986, 02-20, 01-05-1987)
[219] (1986-12-22) NRC Review, Addition of Coagulants to the Reactor Coolant System, Rev. 0 (re various letters)
[220] (1987-01-05) NRC Review, Use of a Chemical Coagulant in the Reactor Vessel Filtration System (re various letters)
[221] (1987-02-20) GPU Safety Evaluation, Criticality Safety Evaluation for Coagulants (re 12-15, 12-31-1986)
[222] (1987-06-10) GPU, Filter Canister Media Modification
[223] (1986-05-05) GPU Safety Evaluation, Addition of a Biocide to the TMI-2 Reactor Coolant System, Rev. 0
[224] (1986-05-06) NRC Review, Addition of a Biocide to the TMI-2 Reactor Coolant System (re 05-05-1986)
[225] GEND-049, TMI-2 Technical Information and Examination Program Annual Report 1984 (1985-04)
[226] GEND-INF-073, TMI-2 DEFUELING TOOLS ENGINEERING REPORT (1986-02)
[227] GEND-055, USDOE TMI Research and Development Program Annual Report 1985 (1986-04)
[228] GEND-058, DROP TESTS OF THE THREE MILE ISLAND KNOCKOUT CANISTER (1987-01)
[229] (1986-06-23) NRC (Internal), Potential Impact of TMI-2 Fuel Movements on Workers in TMI-1 (re 11-15, 06-20-1985)
[230] INL-CON-12-26246, Lessons from TMI Packaging, Transport and Disposition that Apply to Fukushima (2012-07)
[231] (1985-09-10) GPU Technical Evaluation, Defueling Canister TER (re 08-15-1985)
[232] (1985-08-15) GPU Response to NRC, Defueling Canister TER (re 06-10-1985)
[233] (1985-06-10) NRC Review, Defueling Canisters TER (re 04-09-1985)
[234] (1985-04-09) GPU Technical Evaluation, Defueling Canisters, Rev. 0
[235] (1985-11-05) NRC Safety Evaluation, Defueling Canisters (re various letters)
[236] (1986-03-06) GPU Technical Evaluation, Defueling Canister TER, Rev. 2
[237] (1986-08-12) NRC Review, Defueling Canister TER, Rev. 2 (re various letters)
[238] (1986-02-21) NRC Review, Modification of Fines and Debris Vacuum System (re various letters)
[239] (1987-05-07) GPU Technical Evaluation, Defueling Canisters, Rev. 3
[240] (1987-12-04) NRC Review, Defueling Canister TER (re various letters)
[241] (1987-09-22) GPU Technical Evaluation, Defueling Canister TER, Rev. 4

[242] (1987-12-04) NRC Review, Defueling Canister TER (re various letters)
[243] (1987-09-22) GPU Technical Evaluation, Defueling Canister TER, Rev. 4
[244] (1987-02-20) GPU Safety Evaluation, Criticality Safety Evaluation for Coagulants (re 12-15, 12-31-1986)
[245] (1985-05-24) BAW, TMI-2 Defueling Canisters Final Design Technical Report
[246] (1986-05-29) GPU Technical Evaluation, Fuel Canister Storage Racks, Rev. 1
[247] (1987-06-10) GPU, Filter Canister Media Modification
[248] (1987-05-07) GPU Technical Evaluation, Defueling Canisters, Rev. 3
[249] (1987-05-07) GPU Technical Evaluation, Defueling Canisters, Rev. 3
[250] (1987-09-22) GPU Technical Evaluation, Defueling Canister TER, Rev. 4
[251] (1985-11-05) NRC Safety Evaluation, Defueling Canisters (re various letters)
[252] (1985-11-05) NRC Safety Evaluation, Defueling Canisters (re various letters)
[253] (1985-11-05) NRC Safety Evaluation, Defueling Canisters (re various letters)
[254] (1985-11-05) NRC Safety Evaluation, Defueling Canisters (re various letters)
[255] (1985-11-05) NRC Safety Evaluation, Defueling Canisters (re various letters)
[256] (1985-11-05) NRC Safety Evaluation, Defueling Canisters (re various letters)
[257] (1985-11-05) NRC Safety Evaluation, Defueling Canisters (re various letters)
[258] (1985-11-05) NRC Safety Evaluation, Defueling Canisters (re various letters)
[259] (1986-04-11) GPU Safety Evaluation, Replacement of Loaded Fuel Canister Head Gaskets
[260] (1986-03-06) GPU Technical Evaluation, Defueling Canister TER, Rev. 2
[261] (1984-11-08) GPU, Criticality Report, Reactor Coolant System
[262] (1986-04-29) NRC Review, Replacement of Loaded Fuel Canister Head Gaskets (re 04-11-1986)
[263] (1986-01-17) GPU, Plan to Use Modified Debris Canisters for removing End Fittings, etc
[264] (1984-11-08) GPU, Criticality Report, Reactor Coolant System
[265] GPU Nuclear Corporation, “Report on Limits of Foreign Materials Allowed in the TMI-2 Reactor Coolant System During Defueling Activities,” Rev. 1, 15737-2-N09-002, October 1985 [ADAMS Accession No. ML19302G716]
[266] (1986-01-21) NRC Review, Plan to Use Modified Fuel Canisters to Remove End Fittings, etc (re 01-17-1986)
[267] GPU Nuclear Corporation, “Use of Debris Canisters,” Letter to NRC, January 17, 1986 [ADAMS Accession No. ML20140A477]
[268] (1986-05-29) GPU Technical Evaluation, Fuel Canister Storage Racks, Rev. 1
[269] (1985-11-05) NRC Safety Evaluation, Defueling Canisters (re various letters)
[270] (1985-05-03) NRC Review, Fuel Canister Storage Racks TER (re 02-27-1985)
[271] (1985-11-05) NRC Safety Evaluation, Defueling Canisters (re various letters)
[272] (1988-03-29) GPU Safety Evaluation, Canister Handling and Preparation for Shipment, Rev. 6, (Effective Pages Only)
[273] (1984-11-08) GPU, Criticality Report, Reactor Coolant System
[274] (1985-09-27) GPU Hazards Analysis, Potential for Boron Dilution of Reactor Coolant System, Rev. 2
[275] (1987-12-02) GPU Safety Evaluation, Use of Non-Borated Water in Canister Decontamination System, Rev. 1
[276] (1987-12-04) NRC Review, Use of Non-Borated Water in Canister Loading Decon System SER (re 12-02-1987, 08-12-1986)
[277] (1986-03-06) GPU Technical Evaluation, Defueling Canister TER, Rev. 2
[278] (1986-06-20) NRC Safety Evaluation, Canister Handling and Preparation for Shipment Program (re 02-17, 06-11-1986)
[279] (1986-04-11) NRC, Certificate of Compliance 9200, Rev. 0, for Model 125-B Shipping Cask

[280] (1987-04-02) GPU, Safety Evaluation, Canister Handling and Preparation for Shipment SER, Rev. 4 (Effective Pages Only)
[281] (1987-05-26) GPU Safety Evaluation, Canister Handling and Preparation for Shipment, Rev. 5 (Effective Pages Only)
[282] (1986-06-20) NRC Safety Evaluation, Canister Handling and Preparation for Shipment Program (re 02-17, 06-11-1986)
[283] (1988-03-03) NRC Review, Canister Handling and Preparation for Shipment Program TER (re various letters)
[284] (1987-12-02) GPU Safety Evaluation, Use of Non-Borated Water in Canister Decontamination System, Rev. 1
[285] (1987-12-02) GPU Safety Evaluation, Use of Non-Borated Water in Canister Decontamination System, Rev. 1
[286] (1988-03-29) GPU Safety Evaluation, Canister Handling and Preparation for Shipment, Rev. 6, (Effective Pages Only)
[287] (1987-12-04) NRC Review, Use of Non-Borated Water in Canister Loading Decon System SER (re 12-02-1987, 08-12-1986)
[288] (1987-09-22) GPU Technical Evaluation, Defueling Canister TER, Rev. 4
[289] (1986-06-02) GPU, Testing of Core Region Defueling Techniques
[290] (1985-10-10) GPU Safety Evaluation, Early Defueling of the TMI-2 Reactor Vessel, Rev. 4
[291] (1986-01-22) GPU, Plan to Use Hydraulic Impact Chisel to Separate Fused Material
[292] (1986-06-04) NRC Safety Evaluation, Heavy Duty Tong Tool, Spade Bucket, Hydraulic Impact Chisel (re various letters)
[293] (1985-11-12) NRC Safety Evaluation, Early Defueling of the TMI-2 Reactor Vessel (10-10-1985)
[294] (1985-10-10) GPU Safety Evaluation, Early Defueling of the TMI-2 Reactor Vessel, Rev. 4
[295] (1985-11-12) NRC Safety Evaluation, Early Defueling of the TMI-2 Reactor Vessel (10-10-1985)
[296] (1986-01-28) GPU, Fines-Debris Vacuum System Modifications (re various letters)
[297] (1985-09-10) GPU Technical Evaluation, Defueling Canister TER (re 08-15-1985)
[298] (1986-01-28) GPU, Fines-Debris Vacuum System Modifications (re various letters)
[299] (1986-02-21) NRC Review, Modification of Fines and Debris Vacuum System (re various letters)
[300] (1985-11-05) NRC Safety Evaluation, Defueling Canisters (re various letters)
[301] (1986-03-26) GPU, Plan to Use Hydraulic Shredder to Reduce Core Debris
[302] GPU Nuclear Corporation, “Report on Limits of Foreign Materials Allowed in the TMI-2 Reactor Coolant System During Defueling Activities,” Rev. 1, 15737-2-N09-002, October 1985 [ADAMS Accession No. ML19302G716]
[303] (1986-04-16) NRC Review, Use of Hydraulic Shredder
[304] (1986-08-27) GPU, Plan to Use Plasma Arc Torch to Cut Upper End Fittings (re 08-18-1986)
[305] (1987-08-20) NRC Review, Use of Plasma Arc Torch to Cut Upper End Fittings (re various letters)
[306] (1987-11-30) GPU Criticality Assessment for Using Plasma Arc Torch to Cut Lower Core Support Assembly, Rev. 1
[307] GPU Nuclear Corporation, “Report on Limits of Foreign Materials Allowed in the TMI-2 Reactor Coolant System During Defueling Activities,” Rev. 1, 15737-2-N09-002, October 1985 [ADAMS Accession No. ML19302G716]
[308] GPU Nuclear Corporation, “Report on Limits of Foreign Materials Allowed in the TMI-2 Reactor Coolant System During Defueling Activities,” Rev. 1, 15737-2-N09-002, October 1985 [ADAMS Accession No. ML19302G716]

[309] (1984-11-08) GPU, Criticality Report, Reactor Coolant System

[310] (1985-09-27) GPU Hazards Analysis, Potential for Boron Dilution of Reactor Coolant System, Rev. 2

[311] GPU Nuclear Corporation, “Report on Limits of Foreign Materials Allowed in the TMI-2 Reactor Coolant System During Defueling Activities,” Rev. 1, 15737-2-N09-002, October 1985 [ADAMS Accession No. ML19302G716]

[312] (1984-11-08) GPU, Criticality Report, Reactor Coolant System

[313] USNRC, “SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluations,” Vols. 1–3, NUREG/CR-0200, 1982, Revised June 1983, December 1984

[314] GEND-071, TMI CRITICALITY STUDIES LOWER VESSEL RUBBLE AND ANALYTICAL BENCHMARKING (1986-05)

[315] (1987-11-30) GPU Criticality Assessment for Using Plasma Arc Torch to Cut Lower Core Support Assembly, Rev. 1

[316] *Perry’s Chemical Engineers’ Handbook*, 6th Edition, 1984

[317] *Perry’s Chemical Engineers’ Handbook*, 6th Edition, 1984

[318] (1988-03-16) GPU, Criticality Safety Assessment for Use of Plasma Arc Torch (re 11-30-1987)

[319] (1988-04-01) NRC Safety Evaluation, Lower Core Support Assembly Defueling (re 01-18-1988)

[320] (1988-04-01) NRC Safety Evaluation, Lower Core Support Assembly Defueling (re 01-18-1988)

[321] (1988-08-11) GPU, Criticality Assessment, Use of Plasma Arc Torch to Cut Baffle Plates and Core Support Shield

[322] (1988-12-27) GPU Criticality Assessment for Using Plasma Arc Torch to Cut Upper Core Support Assembly, Rev. 1

[323] (1987-11-30) GPU Criticality Assessment for Using Plasma Arc Torch to Cut Lower Core Support Assembly, Rev. 1

[324] (1987-11-30) GPU Criticality Assessment for Using Plasma Arc Torch to Cut Lower Core Support Assembly, Rev. 1

[325] (1988-04-01) NRC Safety Evaluation, Lower Core Support Assembly Defueling (re 01-18-1988)

[326] GPU Nuclear Corporation, “Response to NRC comments on Criticality Safety Assessment for Use of the Plasma Arc Torch to Cut the Lower Core Support Assembly,” Letter to NRC, February 26, 1988 [ADAMS Accession No. ML20147D727]

[327] (1989-01-31) NRC Safety Evaluation, Criticality Assessment for Plasma Arc Torch to Cut Vessel Internals (08-11, 12-27-1988)

[328] (1987-12-28) GPU, Use of Core Bore Machine for Dismantling the Lower Core Support Assembly

[329] (1986-06-11) GPU Safety Evaluation, Core Stratification Sample Acquisition, Rev. 4

[330] (1988-01-08) NRC Review, Use of Core Bore Machine for Dismantling Lower Core Support Assembly (re 12-03, 12-28-1987)

[331] (1989-01-10) GPU Safety Evaluation, Sediment Transfer and Processing Operations, Rev. 4

[332] GPU Nuclear Corporation, “Ex-RCS Criticality Safety,” Technical Plan, TMI/TPO-132, November 1985

[333] (1986-09-05) GPU Safety Evaluation, Reactor Building Sump Criticality Safety

[334] (1986-09-25) NRC Review, Sediment Transfer and Processing Operations (re 03-18-1985)

[335] (1986-09-11) NRC Review, Reactor Building Sump Criticality Safety Evaluation Report (re various letters)

[336] (1987-01-15) GPU Safety Analysis, Pressurizer Spray Line Defueling System, Rev. 2
[337] (1987-01-15) GPU Safety Analysis, Pressurizer Spray Line Defueling System, Rev. 2
[338] (1986-12-22) NRC Review, Proposal for Defueling the Pressurizer Spray Line (re 08-06-1985, 11-26-1986)
[339] (1985-08-06) NRC Review, Defueling Water Cleanup System (re various)
[340] (1986-12-22) NRC Review, Proposal for Defueling the Pressurizer Spray Line (re 08 06 1985, 11-26-1986)
[341] (1986-03-14) GPU Safety Evaluation, Decontamination Using Ultrahigh Pressure Water Flush, Rev. 0
[342] (1985-08-08) Order Amendment (also exemption from 10CFR50 AppA)
[343] GPU Nuclear Corporation, "Report on Limits of Foreign Materials Allowed in the TMI-2 Reactor Coolant System During Defueling Activities," Rev. 1, 15737-2-N09-002, October 1985 [ADAMS Accession No. ML19302G716]
[344] (1986-09-05) GPU Safety Evaluation, Reactor Building Sump Criticality Safety
[345] (1986-07-16) NRC Safety Evaluation, Decontamination using Ultrahigh Pressure Water Flush (re 03-14-1986)
[346] (1985-10-24) GPU, Commencement of Preliminary Defueling Operations
[347] (1985-10-10) GPU Safety Evaluation, Early Defueling of the TMI-2 Reactor Vessel, Rev. 4
[348] (1984-11-08) GPU, Criticality Report, Reactor Coolant System
[349] (1985-10-29) NRC Safety Evaluation, Commencement of TMI-2 Preliminary Defueling Operations (re 10-24-1985)
[350] (1984-07-17) NRC Safety Evaluation, Reactor Pressure Vessel Head Lift (re -3-09-1984)
[351] (1984-09-14) NRC Safety Evaluation, Plenum Assembly Removal (re various letters)
[352] (1985-05-07) NRC Safety Evaluation, Lifting, Transfer, and Storage of Plenum Assembly (re various letters)
[353] (1985-05-02) NRC Safety Evaluation, Heavy Load Handling Over the TMI-2 Reactor Vessel (re 04-19-1985)
[354] (1984-11-08) GPU, Criticality Report, Reactor Coolant System
[355] (1985-10-29) NRC Safety Evaluation, Commencement of TMI-2 Preliminary Defueling Operations (re 10-24-1985)
[356] (1985-09-27) GPU Hazards Analysis, Potential for Boron Dilution of Reactor Coolant System, Rev. 2
[357] (1985-10-10) GPU Safety Evaluation, Early Defueling of the TMI-2 Reactor Vessel, Rev. 4
[358] (1985-10-10) GPU Safety Evaluation, Early Defueling of the TMI-2 Reactor Vessel, Rev. 4
[359] (1985-09-27) GPU Hazards Analysis, Potential for Boron Dilution of Reactor Coolant System, Rev. 2
[360] (1985-11-12) NRC Safety Evaluation, Early Defueling of the TMI-2 Reactor Vessel (10-10-1985)
[361] (1984-11-08) GPU, Criticality Report, Reactor Coolant System
[362] (1985-10-29) NRC Safety Evaluation, Commencement of TMI-2 Preliminary Defueling Operations (re 10-24-1985)
[363] (1985-10-29) NRC Safety Evaluation, Commencement of TMI-2 Preliminary Defueling Operations (re 10-24-1985)
[364] (1985-03-15) NRC Safety Evaluation, Reactor Coolant System Criticality Report (re 11-08-1984)
[365] (1985-11-12) NRC Safety Evaluation, Early Defueling of the TMI-2 Reactor Vessel (10-10-1985)
[366] (1985-09-27) GPU Hazards Analysis, Potential for Boron Dilution of Reactor Coolant System, Rev. 2

[367] (1986-08-16) GPU, Storage of Upper End Fittings in Shielded 55 Gal Drums
[368] (1986-05-15) GPU Safety Evaluation, Defueling SER, Rev. 10
[369] (1985-04-16) GPU Safety Evaluation, Justifying Non-Seismic Design of TMI-2 Post-Accident Systems, Rev. 0
[370] (1986-08-16) GPU, Storage of Upper End Fittings in Shielded 55 Gal Drums
[371] (1986-05-15) GPU Safety Evaluation, Defueling SER, Rev. 10
[372] (1986-05-15) GPU Safety Evaluation, Defueling SER, Rev. 10
[373] (1986-05-20) GPU Safety Evaluation, Defueling SER, Rev. 10 (Correction)
[374] (1986-09-09) GPU Response to NRC (Discussion), Storage of Upper End Fittings (re 08-16-1986)
[375] (1986-09-10) NRC Review, Plans to Store Upper End Fittings (re various letters)
[376] (1986-09-25) NRC Safety Evaluation, Storage of Upper End Fittings in an Array of 55 Gallon Drums (re various letters)
[377] GPU Nuclear Corporation, "Report on Limits of Foreign Materials Allowed in the TMI-2 Reactor Coolant System During Defueling Activities," Rev. 1, 15737-2-N09-002, October 1985 [ADAMS Accession No. ML19302G716]
[378] USNRC, "Review of TMI-2 Foreign Materials Criticality Report," Internal NRC Memorandum, December 2, 1985 [ADAMS Accession No. ML19337B690]
[379] (1986-05-15) GPU Safety Evaluation, Defueling SER, Rev. 10
[380] (1986-05-20) GPU Safety Evaluation, Defueling SER, Rev. 10 (Correction)
[381] (1985-10-10) GPU Safety Evaluation, Early Defueling of the TMI-2 Reactor Vessel, Rev. 4
[382] (1986-05-15) GPU Safety Evaluation, Defueling SER, Rev. 10
[383] (1986-05-20) GPU Safety Evaluation, Defueling SER, Rev. 10 (Correction)
[384] (1984-11-08) GPU, Criticality Report, Reactor Coolant System
[385] GPU Nuclear Corporation, "Report on Limits of Foreign Materials Allowed in the TMI-2 Reactor Coolant System During Defueling Activities," Rev. 1, 15737-2-N09-002, October 1985 [ADAMS Accession No. ML19302G716]
[386] (1985-04-09) GPU Technical Evaluation, Defueling Canisters, Rev. 0
[387] (1986-03-06) GPU Technical Evaluation, Defueling Canister TER, Rev. 2
[388] (1985-04-09) GPU Technical Evaluation, Defueling Canisters, Rev. 0
[389] (1984-11-08) GPU, Criticality Report, Reactor Coolant System
[390] (1987-11-30) GPU Criticality Assessment for Using Plasma Arc Torch to Cut Lower Core Support Assembly, Rev. 1
[391] (1985-10-10) GPU Safety Evaluation, Early Defueling of the TMI-2 Reactor Vessel, Rev. 4
[392] (1986-01-23) GPU Safety Evaluation, Reactor Building Criticality Evaluation, Rev. 2
[393] (1986-07-24) NRC Review, Core Region Bulk Defueling Operations Limited to the Core Region (re 05-15, 07-23-1986)
[394] (1985-11-12) NRC Safety Evaluation, Early Defueling of the TMI-2 Reactor Vessel (10-10-1985)
[395] (1986-07-24) NRC Review, Core Region Bulk Defueling Operations Limited to the Core Region (re 05-15, 07-23-1986)
[396] (1985-11-12) NRC Safety Evaluation, Early Defueling of the TMI-2 Reactor Vessel (10-10-1985)
[397] (1986-05-05) NRC Safety Evaluation, Core Stratification Sample Acquisition SER, Rev. 1 (re 08-30-1985, 12-31-1985)
[398] (1986-05-15) GPU Safety Evaluation, Defueling SER, Rev. 10
[399] (1986-05-20) GPU Safety Evaluation, Defueling SER, Rev. 10 (Correction)
[400] (1986-07-23) GPU, Plan to Use Core Bore Tooling as Defueling Tool
[401] (1986-06-11) GPU Safety Evaluation, Core Stratification Sample Acquisition, Rev. 4

[402] (1986-05-15) GPU Safety Evaluation, Defueling SER, Rev. 10

[403] (1986-07-24) NRC Review, Core Region Bulk Defueling Operations Limited to the Core Region (re 05-15, 07-23-1986)

[404] (1986-07-24) NRC Review, Core Region Bulk Defueling Operations Limited to the Core Region (re 05-15, 07-23-1986)

[405] (1988-01-18) GPU Safety Evaluation, Lower Core Support Assembly Defueling, Rev. 2

[406] (1986-05-15) GPU Safety Evaluation, Defueling SER, Rev. 10

[407] (1986-05-20) GPU Safety Evaluation, Defueling SER, Rev. 10 (Correction)

[408] (1984-11-08) GPU, Criticality Report, Reactor Coolant System

[409] (1985-09-27) GPU Hazards Analysis, Potential for Boron Dilution of Reactor Coolant System, Rev. 2

[410] (1987-11-30) GPU Criticality Assessment for Using Plasma Arc Torch to Cut Lower Core Support Assembly, Rev. 1

[411] (1988-01-08) NRC Review, Use of Core Bore Machine for Dismantling Lower Core Support Assembly (re 12-03, 12-28-1987)

[412] (1988-01-18) GPU Safety Evaluation, Lower Core Support Assembly Defueling, Rev. 2

[413] (1988-01-18) GPU Safety Evaluation, Lower Core Support Assembly Defueling, Rev. 2

[414] (1986-05-15) GPU Safety Evaluation, Defueling SER, Rev. 10

[415] (1986-05-20) GPU Safety Evaluation, Defueling SER, Rev. 10 (Correction)

[416] (1985-09-27) GPU Hazards Analysis, Potential for Boron Dilution of Reactor Coolant System, Rev. 2

[417] (1986-07-11) GPU, Extended Core Stratification Sample Acquisition Activity (re 06-23-1986)

[418] (1988-04-01) NRC Safety Evaluation, Lower Core Support Assembly Defueling (re 01-18-1988)

[419] USNRC Memorandum, L.H. Thonus to J.F. Stolz and W.D. Travers, "Review of Criticality Potential from Plasma Arc Torch at TMI-2," March 17, 1988

[420] (1988-06-06) GPU Safety Evaluation, Completion of Lower Core Support and Lower Head Defueling, Rev. 0

[421] (1986-05-15) GPU Safety Evaluation, Defueling SER, Rev. 10

[422] (1984-11-08) GPU, Criticality Report, Reactor Coolant System

[423] GPU Nuclear Corporation, "Report on Limits of Foreign Materials Allowed in the TMI-2 Reactor Coolant System During Defueling Activities," Rev. 1, 15737-2-N09-002, October 1985 [ADAMS Accession No. ML19302G716]

[424] (1988-04-29) GPU, Modification to Plasma Arc Torch Coolant System (re 11-30-1987, 04-15-1988)

[425] (1988-04-01) NRC Safety Evaluation, Lower Core Support Assembly Defueling (re 01-18-1988)

[426] (1987-11-30) GPU Criticality Assessment for Using Plasma Arc Torch to Cut Lower Core Support Assembly, Rev. 1 [ADAMS Accession No. ML19316B289]

[427] GPU Nuclear Corporation, "Response to NRC Comments on the Criticality Safety Assessment for Using Plasma Arc Torch to Cut the Lower LCSA," Letter to NRC, February 26, 1988 [ADAMS Accession No. ML20147D727]

[428] (1988-04-29) GPU, Modification to Plasma Arc Torch Coolant System (re 11-30-1987, 04-15-1988)

[429] (1988-06-06) GPU Safety Evaluation, Completion of Lower Core Support and Lower Head Defueling, Rev. 0

[430] (1986-05-15) GPU Safety Evaluation, Defueling SER, Rev. 10

[431] (1985-09-27) GPU Hazards Analysis, Potential for Boron Dilution of Reactor Coolant System, Rev. 2

[432] (1986-06-11) GPU Safety Evaluation, Core Stratification Sample Acquisition, Rev. 4
[433] (1988-12-01) NRC Safety Evaluation, Lower Core Support and Lower Head Defueling (re 06-06-1988)
[434] (1988-09-15) GPU Safety Evaluation, Completion of Upper Support Core Assembly Defueling, Rev. 0
[435] (1986-05-15) GPU Safety Evaluation, Defueling SER, Rev. 10
[436] (1988-04-01) NRC Safety Evaluation, Lower Core Support Assembly Defueling (re 01-18-1988)
[437] (1984-11-08) GPU, Criticality Report, Reactor Coolant System
[438] (1988-08-11) GPU, Criticality Assessment, Use of Plasma Arc Torch to Cut Baffle Plates and Core Support Shield
[439] (1985-09-27) GPU Hazards Analysis, Potential for Boron Dilution of Reactor Coolant System, Rev. 2
[440] (1988-04-01) NRC Safety Evaluation, Lower Core Support Assembly Defueling (re 01-18-1988)
[441] (1987-11-30) GPU Criticality Assessment for Using Plasma Arc Torch to Cut Lower Core Support Assembly, Rev. 1
[442] GPU Nuclear Corporation, "Response to NRC Comments on the Criticality Safety Assessment for Using Plasma Arc Torch to Cut the Lower LCSA," Letter to NRC, February 26, 1988 [ADAMS Accession No. ML20147D727]
[443] (1988-04-29) GPU, Modification to Plasma Arc Torch Coolant System (re 11-30-1987, 04-15-1988)
[444] (1988-08-11) GPU, Criticality Assessment, Use of Plasma Arc Torch to Cut Baffle Plates and Core Support Shield
[445] (1988-08-11) GPU, Criticality Assessment, Use of Plasma Arc Torch to Cut Baffle Plates and Core Support Shield
[446] (1988-01-18) GPU Safety Evaluation, Lower Core Support Assembly Defueling, Rev. 2
[447] (1986-01-23) GPU Safety Evaluation, Reactor Building Criticality Evaluation, Rev. 2
[448] (1988-09-15) GPU Safety Evaluation, Completion of Upper Support Core Assembly Defueling, Rev. 0
[449] (1986-05-15) GPU Safety Evaluation, Defueling SER, Rev. 10
[450] (1985-09-27) GPU Hazards Analysis, Potential for Boron Dilution of Reactor Coolant System, Rev. 2
[451] (1989-04-04) NRC Safety Evaluation, Completion of Upper Core Support Assembly Defueling (re 9-15-1988)
[452] (1981-04-29) GPU Response to NRC (Discussions), Potential for Criticality in Spent SDS Vessels
[453] (1982-03-23) GPU Technical Evaluation, Addendum to SDS, RCS Processing Plan
[454] (1987-09-28) GPU Technical Evaluation, SDS, Rev. 5
[455] NUREG-0796, Operation of the Submerged Demineralizer System at TMI-2 (1981-06)
[456] USNRC, "Three Mile Island Accident of 1979 Knowledge Management Digest: Recovery and Cleanup," NUREG/KM-0001, Supplement 1, June 2016 [Available at nrc.gov]

4 HYDROGEN SAFETY EVALUATIONS

TABLE OF CONTENTS

Note: "NA" (not applicable) means the licensee and the NRC determined that the safety topic was not important for the activity.

4.1 Introduction

4.1.1 Background

The broad area of hydrogen safety covered essentially the entire fuel cycle, ending with storage or disposal. The NRC had a presence in each of these areas; however, this report focuses on hydrogen safety as it applied to TMI-2 postaccident conditions and defueling operations. The issue of hydrogen safety at TMI-2 was a subject of intense interest with many concerned parties providing analytical support based on observations, examinations, and experiments.

Residual hydrogen generated during the accident was the initial concern at locations throughout the plant, which included the reactor coolant inside the containment building and radioactive waste storage tanks located in the auxiliary building. While the hydrogen burn event inside the containment building during the accident had removed most of the hydrogen and the hydrogen recombiner had removed residual hydrogen gas, there was a safety concern for hydrogen dissolved in the reactor coolant and process systems early during plant recovery. Hydrogen generation from radiolysis of water in reactor coolant was determined not to be a concern as long as the concentration of dissolved hydrogen in the coolant was maintained above a certain threshold (a practice used in pressurized-water reactors during normal operations). However, radiolysis became an unexpected and urgent concern during the early storage of spent resin liners generated by the EPICOR II wastewater cleanup system.

Research and studies sponsored by the DOE, NRC, licensee, and others determined options to mitigate and control the effects of radiolysis inside highly radioactive EPICOR liners. Experience with EPICOR II liners provided inputs to subsequent designs of the submerged demineralizer system vessels (also known as liners) and defueling canisters. Hydrogen trapped in the damaged reactor core was a safety concern during defueling. Ventilation systems inside the defueling work platform, positioned over the reactor vessel and inside the containment building, prevented flammable concentrations of hydrogen by dispersal and ultimate purging from the containment building.

4.1.2 Chapter Contents

This chapter presents hydrogen safety evaluations of the postaccident TMI-2 and defueling operations. It describes the many studies performed, thereby enabling the reader to understand the thinking of the analysts, the expectations versus the reality, the uncertainties in the data, the measurement and mitigation methods, and the high-level timeline of cleanup activities at that time.

The evaluations presented in this chapter ensured that all activities that could generate or release hydrogen and other combustive gases in radioactive wastes and fuel debris were: (●) addressed and consequences evaluated; (●) controlled in accordance with the requirements of the plant's license, technical specifications, procedures, and applicable regulatory requirements; and (●) covered by adequate contingencies for normal operations and accident conditions.

Key activities of concern with hydrogen generation included: (●) early activities that involved opening the reactor coolant system and processing systems such as Quick Look video inspection of the core, reactor vessel underhead characterization, and makeup and purification demineralizer resin sampling; (●) systems that involved the processing, storage, and shipment of high-level radioactive waste and fuel debris such as defueling canisters, EPICOR II water processing system, and submerged demineralizer system; and (●) others.

Section 2 summarizes the key studies used to support safety evaluations, and the subsequent sections present the safety evaluation for each applicable cleanup activity or system. Section 8 lists the endnotes of references cited throughout this chapter.

4.2 Key Studies

Safety concerns about hydrogen were focused on the residual hydrogen gas from the accident and hydrogen generated in waste processing liners and debris canisters. Residual accident-generated hydrogen included hydrogen dissolved in the reactor coolant, pockets of hydrogen gas initially in the reactor coolant system, and gases trapped in the core debris. Hydrogen gas was generated from radiolysis of highly radioactive fuel debris and fission products in EPICOR II resin liners, submerged demineralizer system liners, and defueling debris canisters.

This section describes a chronological progression of the hydrogen studies during the cleanup. Each subsection below provides a high-level summary of the document.

4.2.1 Introduction

The massive stabilization and cleanup operations subsequent to the TMI-2 accident highlighted the importance of flammable gas control. During the first day of the accident, the metal-water reaction in the reactor core caused hydrogen to build up in the 5700-cubic-meter volume of the containment building. The hydrogen increased to the point where it ignited and burned. [(1)] Even though the containment building gas pressure increased to about 30 pounds per square inch gauge (psig) pressure, and the average gas temperature increased to about 1200 degrees Fahrenheit for a short period, the containment building with its design pressure of 60 psig easily maintained its integrity. Hydrogen-oxygen recombiners were used to remove most (122 kilograms) of the remaining hydrogen from the building. [(2)]

The production of flammable gases by the radiolysis of water and organic materials was fairly well understood by the time defueling of the damaged core began in late 1985. Examinations of highly loaded EPICOR II prefilter liners showed that gas production was proportional to the amount of ionizing radiation absorbed by the water or the organic material. Therefore, the buildup of flammable mixtures of gases could be expected and was observed in enclosed systems that contained organic or wet radioactive materials. Since the presence of flammable gases posed a number of potential hazards, particularly where concentrations might detonate, careful evaluation and control of the handling, shipping, storage, and disposal of these materials were essential to ensure an acceptable degree of safety. [(3)]

4.2.2 EPICOR II Liners

During the planning and early operation of the EPICOR II system, safety concerns focused on the integrity of the spent liners, not on the hazards of hydrogen production. The NRC's environmental impact assessment in August 1979 [(4)] provided the earliest assessment of the EPICOR II system for the decontamination of wastewater in the auxiliary building. This assessment concluded that the prefilter media and ion exchange resins would be changed well before any resin degradation could occur because of radiation levels. The NRC approval of the EPICOR II design, construction, and operating procedures in October 1979 [(5)] and the agency's order [(6)] that directed the licensee to use the EPICOR II system to decontaminate intermediate water in the auxiliary building a few days later provided no insights into the hazards of hydrogen production inside spent resin liners. The NRC order included a requirement to solidify the spent resins before shipment; however, the solidification requirement was later rescinded. The 50 EPICOR II prefilter liners [(a)] contained the most radioactive loading contamination from the EPICOR II system.

4.2.2.1 Radiation Effects on Ion Exchange Materials
(Brookhaven National Laboratory, BNL-50781, November 1977)

This preaccident study [(7)] was often cited in early reports associated with radiation effects on EPICOR II prefilter liner resins. This report documented an extensive literature review and data compilation on the radiation damage of ion exchange resins used in nuclear fuel and waste processing areas. The report noted that the understanding of the behavior of resins under ionizing radiation was too limited to justify quantitative predictive modeling. A number of generalizations were stated, several of which were believed to apply to EPICOR II resins. The study noted that the gaseous products resulting from the process of ionizing radiation on ion exchange resins primarily consisted of hydrogen.

4.2.2.2 Leachability, Structural Integrity, and Radiation Stability of Resins in Cement
(Brookhaven National Laboratory, May 1980)

This report [(8)] presented results of a literature review and recent experiments on the radiation stability of resins. Based on the 1977 literature review, [(9)] the 1980 report concluded that the fundamental processes causing radiation damage in resins were not understood, and preliminary calculations indicated a moderate level of gas generation as a result of radiolysis of water with the resin. During the irradiation of ion exchange resins, the main gaseous products formed were hydrogen and carbon dioxide. In cation exchange resins, the evolution of hydrogen was reported as a linear function of absorbed dose. The hydrogen evaluation also showed an increase with water content (swelling) of the resin. Following the review of the 1980 Brookhaven report, the NRC requested [(10)] design information on the liners and detailed resin content of

[a] Editor's Note: The initial 50 EPICOR II prefilter liners were highly loaded with radioactivity. The DOE accepted these liners for research and disposal. One prefilter was shipped to the DOE's Battelle Columbus Laboratory for study of the resin and liner degradations. The other 49 liners were shipped to INEL for disposal studies. Four of these liners were shipped to the DOE's Pacific Northwest Laboratory for resin vilification tests. The remaining 45 liners were disposed of using first-of-its-kind high-integrity containers at the commercial burial site in the State of Washington.

each liner from the licensee in order to assess the performance capabilities of the EPICOR II liners/resin system.

4.2.2.3 EPICOR II Radwaste System Resin Irradiation Evaluation
(Metropolitan Edison Company, July 1980)

This licensee report [11] was a response to the NRC's request for information. This report indicated that the primary effect of gas generation in EPICOR II wastes was the increase of liner pressure to 2 pounds per square inch (psi) over the infinite life of a liner under worst-case conditions (design pressure was 19 psi). The secondary effect was the production of an acid that might lower the pH of the liner internals, which would result in more aggressive chemical attack on the paint, the liner surface, or both.

4.2.2.4 Summary Studies on Stability of Ion Exchange Resins in Radiation Environments
(INEL, October 1980)

This internal report [12] summarized the state-of-the-art research on resin stability as applicable to the known technical nature of the EPICOR II resins. Previous studies (by Brookhaven National Laboratory, Pennsylvania State University, and Georgia Institute of Technology) and information reviewed in the INEL report focused on resin degradation and liner integrity. The report concluded that actual sampling and analysis of the EPICOR II resins would be desirable to positively evaluate the liner and resin status. The report suggested that hydrogen would be present due to radiolysis and that combustible or explosive mixtures could result during handling, so care should be exercised.

4.2.2.5 Evaluation of the Liner Integrity of the TMI-2 EPICOR II Radwaste Systems
(Metropolitan Edison Company, December 1980)

This licensee report [13] evaluated the integrity of the EPICOR II spent resin liners affected by various internal environmental conditions that were believed to exist at this time. This evaluation was based on effluent data from 65 spent liners. The report recommended that gas samples should be taken from the EPICOR II liners to determine if hydrogen was being produced or if other gases were detected that could indicate degradation of the resin beads. Tests conducted on site had also showed that the liners could not be pressurized above 2 pounds per square inch gauge pressure, thus eliminating concerns about overpressurizing the liners because of the evolution of gases from the resins.

4.2.2.6 Hydrogen Release Incident at TMI-2 (May 16, 1981)

Before a heavily loaded EPICOR II prefilter liner was shipped to the DOE's Battelle Columbus Laboratories for characterization, the liner was placed in a shielded pit in the EPICOR II building and manually vented. The liner vent plug was partially removed (unscrewed by four turns) during the time when a combustible mixture of gas was detected, after which personnel were evacuated from the area around the liner. The detection of a combustible gas mixture continued for about 5 hours. [14] Shortly after the liner arrived at Battelle, a gas sample was drawn that confirmed the presence of hydrogen gas and indicated a depletion of oxygen. The liner contained 12 percent hydrogen and 0.2 percent oxygen. Based on this information, the licensee decided that the EPICOR II prefilter liners stored at TMI should be sampled for gas, vented, and, if necessary, purged with inert gas before shipment from TMI-2. [15]

4.2.3 Submerged Demineralizer System

Planning began in 1980 to replace the EPICOR II system that would decontaminate highly radioactive accident-generated water located in the containment building basement, reactor coolant system, and reactor bleed tanks in the auxiliary building. Neither the EPICOR II system design and early operations nor the initial technical evaluation report submitted in April 1980 for the submerged demineralizer system (SDS) design and its revision [b] 1 year later mentioned the hydrogen generation or associated hazards. Nevertheless, the SDS was equipped with a vent system to allow for venting of temporary stored SDS liners (also known as vessels) in the spent fuel pool. Also, the SDS liners were designed and fabricated as certified pressure vessels in accordance with the requirements of the American Society of Mechanical Engineers *Boiler and Pressure Vessel Code*. [16, 17] In September 1980, the NRC indicated in a letter [18] to the licensee that it would not approve any method for decontamination of the contaminated water in the TMI-2 containment building sump until the completion of the agency's environmental review in the TMI-2 PEIS. The PEIS, [19] which the NRC issued in March 1981, did not mention hydrogen generation as a safety concern.

4.2.3.1 Potential for Explosive Mixtures in Submerged Demineralizer System Liners
(Metropolitan Edison Company, April 1981)

At the request of the NRC, this licensee letter [20]reviewed the potential for the generation of explosive gas mixture [c] in the spent submerged demineralizer system (SDS) ion exchange liners. An explosive gas mixture could result from the dissociation of water under a radiation field. Calculations showed that the SDS zeolite liners would remain structurally sound even in the event of a hydrogen detonation. The liners, designed and tested for 24.8 atmospheres service pressure, would be continuously vented to 1 atmosphere to the SDS off-gas system, since the maximum pressure pulse for any hydrogen explosion (initiated at 1 atmosphere) was 18.05 atmospheres. [d] Furthermore, it was improbable that an explosive hydrogen mixture would develop. Available water radiolysis data for irradiated zeolite systems indicated that

---

[b] The original technical evaluation plan was revised in March 1981 to include the EPICOR II system as a polisher to refine the decontamination of the SDS discharge.

[c] Editor's Note: Hydrogen combustion is usually classified either as "deflagrations" or "detonations." The term "explosion" usually refers to a detonation but is somewhat ambiguous and should be avoided. Deflagrations are combustion waves in which unburned gases are heated by thermal conduction to temperatures high enough for a chemical reaction to occur. Deflagrations normally travel subsonic and result in quasi-static (nearly steady-state) loads on containment. Detonations are combustion waves in which heating of the unburned gases is due to compression from shock waves. Detonation waves travel supersonic and produce dynamic or impulsive loads on containment in addition to quasi-static loads.

[d] Editor's Note: It is unclear whether this is the peak detonation wave Chapman-Jouget pressure. Higher pressures are possible, but it seems the peak pressure value is irrelevant if an explosive mixture is improbable.

recombination would occur, and subsequent hydrogen/oxygen concentrations would be nonexplosive.

The licensee's follow-on letter [(21)] to the NRC stated that scaling the results of the Pennsylvania State University experiments (reported in "Radiation Effects on Ion Exchangers Used in Radioactive Waste Management," dated October 1980, under contract with Brookhaven National Laboratory) to exposure dose rates corresponding to 60,000 curies per liner would not yield explosive hydrogen concentrations (0.74 to 5.6 volume percent) with corresponding scaled oxygen concentrations. The MAXSIMA CHEMIST computer code calculated equilibrium hydrogen concentrations in the range of 5 to 14 volume percent and oxygen concentrations in the range of 18 to 20 volume percent, with equilibrium gas pressures less than 1.18 atmospheres. The licensee's supplementary safety evaluation stated that hydrogen was flammable in the range of 4 to 72.2 volume percent and generally explosive [(e)] in the range of 14 to 68 volume percent. Explosion of the 14-volume-percent mixture would result in an overpressure of 5.15 atmospheres and could be safely contained. The licensee concluded that an explosion would not occur, no credible ignition source would exist (the liners were underwater), and the zeolite would be continuously cooled by vapor condensation.

4.2.3.2 Evaluation of Increased Cesium Loading on Submerged Demineralizer System Zeolite Beds

(DOE, DOE-NE-0012, May 1981)

This report [(22)] from the DOE's submerged demineralizer system (SDS) task force evaluated the relative technical and financial benefits in storing, shipping, treating, and disposing of SDS zeolite liners, assuming that the liners would be loaded to a level higher than that originally planned (10,000 curies per liner) by the licensee. The task force concluded that it was technically feasible to load the zeolite liners used in the SDS to levels up to 60,000 curies of cesium per liner without additional preoperational testing. Findings from the report concerning hydrogen generation included the following: (●) Loaded SDS zeolite liners would contain considerable amounts of water. (●) Radiolysis of the water during storage would generate hydrogen and oxygen. (●) Evolved hydrogen and oxygen gases could be dispersed by venting the stored SDS liners to an appropriate off-gas system, as planned in the SDS design. (●) Wet SDS liners loaded with strontium and cesium should not be stored for prolonged periods in sealed containers or casks because of the potential for overpressurization from radiolytic decomposition of all the residual water to hydrogen and oxygen.

---

[e] Editor's Note: Experiments have shown that, if the hydrogen concentration in dry air is 15 percent or more, a deflagration can accelerate into a detonation, which is the phenomenon identified as "deflagration to detonation transition." If there are other gases, the acceleration of the burn is likely to be slowed with increasing concentrations of other gases. It has been observed that detonation limits are functions of geometry and scale and not universal values at given mixture concentrations, temperatures, and pressures.

4.2.4 Evaluation of Long-Term Postaccident Core Cooling of TMI-2
(NRC, NUREG-0557, May 1979)

This early report [23] addressed the acceptability of a proposed method for long-term cooling of the damaged reactor. The licensee proposed to adopt the Babcock & Wilcox recommendation to use natural circulation core cooling. The method of cooling with one reactor coolant pump running was preferred; however, forced flow cooling had uncertainties related to eventual pump and instrument degradation by the harsh environment inside the containment building. The proposed method would place both the steam generator secondary cooling systems and the reactor coolant system (RCS) in a water-solid condition for a closed-cycle cooling mode. The report evaluated the short- and long-term potential for evolution of noncondensable gases, including hydrogen. Specifically, the study evaluated the minimum primary pressure required to prevent noncondensable gas development and the effects of a coolant void on natural circulation.

The report concluded that no bubbles would form at high points in the RCS by development of noncondensable gases in solution or residing in RCS pockets, as long as the RCS pressure remained in excess of 300 pounds per square inch gauge (psig). The procedure for achieving natural circulation conditions would require a series of incremental pressure reductions from 1000 psig down to 300 psig.

The report considered the potential effects of radiolysis of reactor coolant. In gamma and neutron fields typical of the RCS, a hydrogen concentration of 17 cubic centimeters per kilogram (cc/kg) water was needed to suppress radiolysis of the primary coolant (the report referenced U.S. Patent 2937981, “Suppression of Water Decomposition,” dated May 24, 1960). In operating plants, the usual hydrogen concentration was maintained a little higher, and the saturation concentrations in the TMI-2 reactor coolant at the time of the report were significantly higher. Therefore, the report concluded that no significant radiolysis would take place at the operating conditions defined by the proposed procedure for achieving natural circulation.

The report then evaluated the effects of noncondensable gas on natural circulation that might unexpectedly come out of solution under certain core exit and hot-leg temperatures and pressure. The released gas would collect at high points in the upper portion of the reactor vessel, hot-leg “candy canes” near the top of the steam generators, and the reactor coolant pumps. The gas could be driven back into the solution by increasing pressure to maintain sufficient subcooling. The time required for plant operators to accomplish this reverse process was on the order of hours. This time would be within the available timeframe for operator actions following loss of natural recirculation.

4.2.5 Programmatic Environmental Impact Statement
(NRC, NUREG-0683, March 1981)

The TMI-2 PEIS related to decontamination and disposal of radioactive wastes resulting from the accident at TMI-2 [24] addressed the options and associated environmental impacts of cleanup activities, as well as the potential radiological health and safety impacts. At the time the report was issued, the issues of hydrogen generation in the spent EPICOR II resin liners were just becoming known, so the PEIS does not mention hydrogen as a safety issue.

4.2.6 Status of EPICOR Prefilter Liners

(GPU Nuclear, August 4, 1981)

This licensee's letter report [25] to the NRC addressed the evaluation related to gas generation in the EPICOR II prefilter liners, hydrogen combustion, pressure buildups, and long-term storage of the liners. The evaluation, based on a total organic mixture, was considered to be the bounding case, so the results would be conservative for liners that contained a mixture of both inorganic and organic ion exchange media.

- ***Conclusion***. The report concluded that hydrogen generation and the storage of the liners, for the shorter term, would not threaten the health and safety of the public. For the longer term, the liner depressurization and high-integrity container would provide containment for the prefilter liners for a minimum of 300 years. After this time, activity levels would be comparable to those of typical low-level radioactive waste. Specific conclusions from the letter report are summarized below.

  - *Liner Pressure*. Experimental data indicated that pressures in the liners were not expected to exceed 19 psig and liners began gas leakage around 2 psig. At or below 19 psig, the lids and seals would deform to the point that the seals would no longer hold the pressure, and the pressure would decay to an equilibrium point.

  - *Liner Oxygen Concentration*. Calculations indicated that hydrogen concentrations were expected to be in the range of 40 to 65 percent by volume, and oxygen was expected to be below 0.2 percent by volume. Oxygen concentrations must be at least 5 percent by volume to make a flammable mixture.

  - *Requirements for Combustion*. The status report evaluated the four elements needed for combustion inside EPICOR II storage modules: (●) fuel (hydrogen); (●) oxidizer (oxygen); (●) ignition source; and (●) chain carrier ions.

    The evaluation concluded the following: (●) Hydrogen would not build up in the storage modules because of the combination of low hydrogen generation rates, low hydrogen leak rates out of the liners, and hydrogen leakage from the modules due to diffusion and buoyancy. (●) Oxygen depletion mechanisms at work in the liners (corrosion, carbon dioxide production, carbon monoxide production, and sulfur dioxide production) would keep oxygen well below the flammable mixture threshold. The presence of carbon monoxide was an indication of oxygen depletion. (●) No ignition sources, such as sparks from impacts, friction, static electricity, improper grounding, and open flame sources (matches and cigarettes), were known to be in the closed storage modules, and efforts to reduce the potential for ignition sources were being made for open modules. Special handling procedures would be developed for handling prefilter liners. (●) Radiation from the liners would somewhat increase the presence of chain carrier ions (necessary to propagate combustion) but in amounts insufficient to affect the flashpoint or combustion rate of hydrogen.

- *Postulated Combustion (Storage Module)*. Calculations indicated that if the entire content of hydrogen in a liner were released to the module instantaneously, the equilibrium mixture would be slightly above the lower limit of flammability but well below the limit for detonation. The combustion for this amount of hydrogen would result in the slight lifting of the 16-ton concrete module lid. The resins would see a small temperature increase that would not be expected to exceed normal resin operating temperatures.

- *Postulated Combustion (Liner)*. Analysis showed that a combustible mixture would not occur in the liner. Any postulated combustion was considered highly unlikely to cause EPICOR liner damage extensive enough to release resins.

- *Liner Depressurization*. The planned liner depressurization would minimize any long-term concern about hydrogen generation by ensuring the mechanical integrity of the liners. Depressurization would reduce the inventory of hydrogen in the liners, thus reducing the possibility of combustion, as well as relieving pressure-induced stress in the liners.

4.2.7 Submerged Demineralizer System Processing of TMI-2 Wastewater
(INEL, GEND-031, February 1983)

This report [(26)] covered the history, development, and operations of the submerged demineralizer system through August 1982, including the processing of the first six batches of reactor coolant system water, reactor coolant bleed tank, and containment building basement water. There were two main advantages of using zeolites rather than organic resins for ion exchange beds at TMI-2. First, zeolites showed good stability to doses of 10^{11} rad and higher (DOE-NE-0012 [(27)]), while organic resins were limited to integrated radiation doses of less than 10^8 rad before they became significantly degraded. [(f)] The stability of zeolite at high doses was an important criterion in designing a system to absorb high concentrations of radioisotopes.

4.2.8 Preparations To Ship EPICOR Liners
(GPU Nuclear, GEND-029, June 1983)

This report [(28)] discussed the prototype gas sampling tool for breaching the containment of the liners, the support equipment for sampling and inerting the liners, and the characterization program for determining the radiolytic hydrogen generation rates in the liners. The sampling and

[f] Editor's Note: The reference was a Pennsylvania State University report, "Radiation Effects on Ion Exchangers Used in Radioactive Waste Management," issued October 1980. This work was done under contract with the DOE at Brookhaven National Laboratory. The report cannot be found.

analysis of the hydrogen-rich atmosphere of the 49 EPICOR II prefilter liners [g] provided data to ensure safe storage and shipment of highly loaded ion exchange media.

- ***Results***. Hydrogen generation studies were performed on 10 of the 18 prefilter liners. Generation rates on the other eight liners were determined by interpolating a plot of hydrogen generation rate versus curie loading for the 10 studied liners. After successful inerting and shipment of the 18 prefilters, the following observations were made: (●) The majority of the liners were not airtight, and consequently, very few liners were found to contain any pressure. (●) In addition to hydrogen, oxygen, and nitrogen, all liners contained carbon dioxide and trace amounts of carbon monoxide and methane. (●) All liners showed significant oxygen deficiency, generally less than 0.2-percent oxygen upon opening. (●) Average hydrogen production rate for the prefilter liners was 5.94×10^{-6} liter per curie-hour. With this rate and assuming the liner did not leak, a 2000-curie liner would generate a 4 percent mixture of hydrogen in a nitrogen and carbon dioxide atmosphere in about 100 days. (●) Hydrogen generation rate versus curie loading was approximately linear for the range of liners studied. (●) Two liners were found at a significant positive pressure indicating a tightly sealed vessel. The volume of hydrogen in these vessels compared to within 24 percent of that predicted by the experimentally determined generation rate (5.94×10^{-6} liter per curie-hour). (●) During shipment of the 18 inerted prefilter liners to INEL, all liners remained well below 4 percent by volume hydrogen and below 5 percent by volume oxygen.

- ***Conclusion***. The report concluded that the magnitude of this generation rate predicted safe shipping windows in excess of the necessary 16-day shipping window required by the DOE. Further, it was not expected that the 31 remaining prefilter liners would vary significantly from the first 10 studied, with respect to hydrogen generation. Consequently, all remaining liners were predicted to meet the safe shipping criterion of less than 4-percent hydrogen during transport. The prototype gas sampler and its auxiliary facilities provided a safe, remote method of sampling, purging, and inerting the EPICOR II prefilters. The remote nature of this equipment saved significant dose exposure, avoided the personnel hazards associated with critical quantities of hydrogen gas, and provided data to aid in the processing of highly loaded ion exchange media.

Editor's note: Several previous reports documented the results of examinations of EPICOR II prefilter liners. [29, 30, 31, 32, 33] The NRC issued subsequent reports under its Low-Level Waste Database Development Project to study the chemical and physical conditions of the synthetic ion exchange resins found in several EPICOR II prefilter liners. [34, 35, 36]

[g] Editor's Note: The initial 50 EPICOR II prefilter liners were highly loaded with radioactivity. The DOE accepted these liners for research and disposal. One prefilter was shipped to the DOE's Battelle Columbus Laboratory for study of resin and liner degradations. The other 49 liners were shipped to INEL for disposal studies. Four of these liners were shipped to the DOE's Pacific Northwest Laboratory for resin vilification tests. The remaining 45 liners were disposed of using first-of-its-kind high-integrity containers at the commercial burial site in the State of Washington.

4.2.9 Submerged Demineralizer System Vessel Shipment Report
(INEL, GEND-035, June 1984)

This report [37] described the successful approach used to ensure safe shipment of the submerged demineralizer system (SDS) liners, in compliance with government regulations. Specifically, a catalyst was inserted into each sealed liner to recombine the radiolytic gases back into water. The liners were vacuum dried to enhance catalyst system performance. This report described: (●) testing methods that the licensee used to determine gas generation rates; (●) DOE laboratory testing of catalyst performance; (●) TMI demonstration of catalytic recombination in a radioactive vessel; (●) demonstration of the vacuum drying system and catalyst insertion tool; (●) preparations for subsequent shipments; and (●) DOE research and development programs.

Liners containing zeolites and absorbed fission products from processing accident-generated water at TMI through the SDS were found to generate radiolytic hydrogen and oxygen gases. In some liners with high curie contents, gas generation during shipment could have resulted in flammable gas concentrations exceeding Federal limits for radioactive material shipments. Tests of a catalyst bed in the liner demonstrated that recombination of the gases back into water would permit safe shipment of the sealed liners. A catalyst was loaded into a screen assembly of each liner. Liner pressure monitoring ensured that net gas generation had stopped and that hydrogen and oxygen concentrations were kept below flammable limits. All shipments complied with Federal regulations.

Radiolytic gas generation from the concentrated fission products on the zeolites was recognized before system startup and characterized after the first liners were removed from service. Catalyst testing at more than twice the observed maximum gas generation rate established the performance of recombiners in SDS liners. Implementation of the vacuum drying and the addition of the catalyst process at TMI-2 successfully demonstrated radiolytic gas control by recombination for the liner with the highest amount of radioactivity. Hydrogen and oxygen concentrations in each liner were maintained within acceptable limits. All shipments complied with Federal regulations and were conducted without incident.

4.2.10 Commercial Disposal of High-Integrity Containers with EPICOR II Prefilters
(INEL, GEND-048, September 1985)

This report [38] briefly described the processes, equipment, and facilities used in the disposal of 45 EPICOR II prefilters, [h] each sealed in a high-integrity container (HIC). The significance of this effort involved the first-of-a-kind production use of a reinforced concrete HIC at the U.S. Ecology, Inc., land burial facility in the State of Washington as part of the commercial disposal campaign. This allowed for safe disposal of high specific activity ion exchange

[h] The initial 50 EPICOR II prefilter liners were highly loaded with radioactivity. The DOE accepted these liners for research and disposal. One prefilter was shipped to the DOE's Battelle Columbus Laboratory for study of the resin and liner degradations. The other 49 liners were shipped to INEL for disposal studies. Four of these liners were shipped to the DOE's Pacific Northwest Laboratory for resin vilification tests. The remaining 45 liners were disposed of using first-of-its-kind HICs at the commercial burial site in the State of Washington.

materials in EPICOR II prefilters. [i] Successful completion of the HIC disposal demonstration was the result of a 4-year effort. The design analysis of the HIC for disposal of EPICOR II prefilter liners was presented in EGG-TMI-6304. [39]

The HIC was a reinforced concrete cylindrical container designed specifically for disposal of an EPICOR II prefilter at a commercial disposal facility for low-level radioactive waste. The container was designed for a minimum of 300 years (about 10 half-lives of the predominant isotopes) so radioactive isotopes contained in the resins could decay to a nonhazardous level.

The HIC was equipped with a vent system to release the gas produced by radiolysis in the EPICOR II prefilter. The EPICOR II prefilter was purged with argon gas before being loaded into the HIC. The vent was cast into, and protected by, the reinforced concrete lid assembly. Without venting, the HIC had sufficient burst strength to contain the gas that could be generated, based on a 300-year life. The vent system could accommodate 0.15 mole of hydrogen generation per day, a flow rate nearly 3 times greater than the design basis of 0.052 mole per day. The vent system consisted of the following components: (●) stainless-steel filter element with a 40-micron pore size, which ensured that solid (resin) particles would not escape from the container to the external environment; (●) PVC (thermoplastic) water trap, which was self-purged by gases generated within the container; and (●) polyethylene external filter (70 microns) with large surface area, which was located in a recessed PVC pocket at the lid edge to prevent mud and debris from entering the vent.

4.2.11 Evaluation of Special Safety Issues Associated with Handling TMI-2 Core Debris (INEL, GEND-051, June 1985)

This report [40] addressed the evaluation and resolution of safety concerns relating to radiolytically generated hydrogen and oxygen, as well as pyrophoricity. The potential for steam generation in debris canisters during an accidental fire was also addressed. This document described the methods, techniques, configurations, and conditions that maximized safety and minimized cost and scheduling needs when resolving these safety issues.

- ***Hydrogen Generation***. Radiolytic hydrogen and oxygen gas generation of wet core debris differed from undamaged irradiated fuel assemblies because of the close contact that the debris had with water. When debris was in direct contact with water, a high fraction of radiolytic hydrogen and oxygen was generated from beta radiation in the TMI fuel canisters. In addition, fine core debris particles trapped a considerable amount of water, producing a higher rate of hydrogen and oxygen. An evaluation of these qualitative conditions indicated that the two factors combined could increase the radiolytic production rate of hydrogen and oxygen gases in

---

[i] INEL had the administrative and technical responsibility for developing and testing the HIC, acquiring regulatory approvals, and demonstrating commercial disposal of the HIC loaded with an EPICOR II prefilter. Under INEL subcontracts, Sandia National Laboratories developed specifications for the HIC; Nuclear Packaging, Inc. (NuPac) designed the HIC, presented a design review, built two prototype HICs, and tested one unit. The State of Washington, with support from the NRC, commented on the HIC design and reviewed the disposal program. The State issued the Certification of Compliance to US Ecology for disposal of the 50 EPICOR II prefilters using the HIC designed by NuPac.

canisters. The production rates from moist core debris would be several orders of magnitude higher than undamaged fuel assemblies with the same irradiation and cooling histories.

The calculations in the report indicated a probable-maximum hydrogen plus oxygen generation rate of 0.11 liter per hour (0.076 liter per hour of hydrogen and 0.038 liter per hour of oxygen) per 800 kilograms of TMI-2 core debris (refer to Appendix A to the report). This probable-maximum gas generation rate could be reduced by drying the core debris in the canister or by inserting an oxygen scavenger, such as carbon steel wool or hydrazine, to chemically remove some or all of the oxygen. However, the degree of dryness necessary to keep hydrogen and oxygen concentrations below flammable limits was reported as extreme, and the effects that oxygen scavenging would have on hydrogen overpressure required further research.

The pressure and gas concentrations for various canister loadings after 88 days (twice the planned shipping time) were based on calculations presented in the report. The report noted that if the hydrogen or oxygen gas concentrations were kept below their lower limits of flammability (4 percent for hydrogen or 5 percent for oxygen) for the 88-day period, the ratio of the volume of wet core debris to the canister empty volume would be limited to less than 15 percent. To avoid exceeding the 3-percent oxygen limit, as the report suggested for pyrophoricity control, the volume of wet core debris in a canister would be limited to 10 percent. To make better use of the canister volume, other means of gas control, such as drying, adding oxygen scavengers, and adding hydrogen-oxygen gas recombiners, were further evaluated in Sections 7 and 8 of the report).

- ***Hydrogen and Oxygen Control***. The report examined three types of oxygen scavengers for the defueling canisters: hydrazine, carbon steel wool, and catalytic hydrogen-oxygen recombiners.

  - *Hydrazine*. The disadvantage of hydrazine was that it replaced oxygen with nitrogen. The calculated net gas generation would require an increase in the design pressure of the canister/cask, unless it could be demonstrated that hydrogen overpressure would prevent the net generation of hydrogen and oxygen before reaching the design pressure. Another disadvantage was that hydrazine would be consumed as oxygen was scavenged. About 40 grams of hydrazine would be consumed per month if 0.038 liter per hour of oxygen were reacted. In addition, hydrazine would need to be extensively tested in a stagnant system to ensure that diffusion would not limit its effectiveness. As a result of these disadvantages, the use of hydrazine was not further considered.

  - *Carbon Steel Wool*. The use of carbon steel wool in this application was tested. Carbon steel wool would be consumed as oxygen was scavenged. About 90 grams of iron would be consumed per month if 0.038 liter per hour of oxygen was reacted. Iron oxidation would allow hydrogen buildup in the canister, but unlike hydrazine, no secondary gas would be generated. Because of the increased uncertainties and lack of any advantage over the use of catalysts, the addition of carbon steel wool was not further considered.

- *Catalytic Hydrogen-Oxygen Recombiners*. Catalytic recombiners had a long history of satisfactory use in controlling gas buildup. The DOE's Hanford Operations conducted a series of tests evaluating catalysts and catalyst bed parameters by using special wet-proof and proven "industry standard" catalysts in simulated shipping canisters. Section 7 of the report presented the results of this effort.

  Results common to all types of catalysts tested included the following: (•) When catalysts were totally submerged in water, essentially no recombination occurred. (•) Waterborne contaminants (synthetic cooling water with dissolved salts and particulates) had only slight, if any, effect on the catalysts. (•) Catalyst beds that were drained after submersion underwater at 2 atmospheres for about 24 hours started recombining hydrogen and oxygen, even in a 100-percent relative humidity atmosphere. Recombination rates increased with bed drying as a result of the exothermic reaction. (•) Thin beds with a larger surface area exposed to the canister interior performed distinctly better than compact beds. (•) Catalyst tests starting under frozen conditions indicated unacceptably low recombination rates. However, when recombination was initiated above freezing conditions and then cooled to below freezing, recombination continued until the system was shut down after 35 hours.

- ***Recommendations***. The report made recommendations consistent with the calculated probable-maximum radiolytic hydrogen-oxygen generation rate of 0.11 liter per hour per 800 kilograms in TMI-2 core debris and noted the following: (•) Two or more catalyst beds should be located in each of the TMI-2 core debris canisters so that, after dewatering and closing the canisters, at least 100 grams of the specified mixed catalyst would not be submerged in water at any given time, regardless of canister orientation. (•) Each catalyst bed should consist of 80-percent Engelhard Deoxo-D nuclear grade catalyst and 20-percent Atomic Energy of Canada Limited silicone-coated catalyst. Additional Engelhard catalyst should be used to fill any oversized beds.

4.2.12 Hydrogen Control in Handling, Shipping, and Storage of Wet Rad Waste
(INEL, GEND-052, February 1986)

This report [41] provided pertinent information concerning the handling and shipping of wet radioactive wastes, which resulted from experience at TMI-2. The report described engineering tools, procedures, and precautions that were intended to ensure the safe handling, shipping, and storage of wet radioactive wastes. A stepwise procedure was presented that permitted the individual investigator to evaluate the potential for flammable gas generation and to minimize potential hazards, with the intent of meeting NRC requirements. In addition, the report presented details of types of effective catalysts, development of catalyst beds, catalyst bed location features, and catalyst bed design criteria.

- ***Corrective Procedures***. The evaluation process identified containers that could be safely shipped and containers that would require some other action before shipment. Corrective procedures, when treated iteratively and selectively, provided the optimum route to safe shipping conditions. The procedures of varying difficulty and cost included: (•) shortening the preparation and shipping schedule; (•) purging/evacuating before shipping; (•) purging and pressurizing with diluent gases; (●) increasing void volume (removal of liquids, venting to cask); (●) reducing target materials; (●) reducing source terms (source removal, chemical poisoning); and (●) using catalytic recombiners.

- ***Catalytic Bed Design Criteria***. The catalytic recombiners designed for the TMI-2 core debris canisters that contained wet radioactive material had the following criteria: (●) One catalyst bed must be exposed to the gas/vapor space at all times. (●) Very wet systems should contain catalyst beds composed of 80-percent Engelhard Deoxo-D and 20-percent silicone-coated Engelhard and Atomic Energy of Canada Limited catalysts or a test-proven equivalent. This catalyst mix was wet resistant and recovered rapidly after being submerged in water. (●) The amount of exposed catalyst required was proportional to the gas generation rate. The recommended ratio of mixed catalyst to the gas generation rate (in milliliters of catalyst per milliliters of hydrogen and oxygen gas produced per hour) was 1.0. Thus, a generation rate of 50 milliliters per hour required a 50-milliliter bed of the mixed catalyst. Each bed should contain this volume of missed catalyst, even when multiple beds were needed to prevent submersion. (●) The recommended ratio of bed volume to the exposed (screened) area in cubic centimeters was 1.0. This resulted in a 1-centimeter-thick bed of the mixed catalyst, when screened on one side.

These mixed-bed recombiners were projected to maintain the oxygen concentration below about 0.5 percent, or the hydrogen concentration below about 1.0 percent, even under the very wet conditions in a maximum loaded TMI-2 canister. Further, the testing was performed using hydrogen and oxygen generation rates that were a factor of 3 times higher than those used to size the beds. Tests conducted at temperatures below freezing showed that the catalyst beds, as designed, would remove hydrogen and oxygen gases at the design rate for at least a few weeks under these temperature extremes.

4.2.13 Calculation of Gas Generation in Sealed Radioactive Waste Containers
(INEL, GEND-041, May 1986)

The NRC issued Information Notice No. 84-72, “Clarification of Conditions for Waste Shipments Subject to Hydrogen Gas Generation,” in September 1984. [42] This notice required waste generators to demonstrate, either by tests or measurements, that a combustible mixture of gases was not present in radioactive waste shipments; otherwise, the waste must be vented within 10 days of shipping. A task force, formed by the Edison Electric Institute to evaluate these NRC requirements, developed a calculation method to quantify hydrogen gas generation in sealed containers. The GEND-041 report [43] provided the calculation method, along with comparisons to actual measured hydrogen concentrations from EPICOR II resin liners vented during their preparation for shipment. As a result of this development, the NRC recently altered certain waste shipment certificates of compliance to allow calculations, tests, and measurements as acceptable means of determining combustible gas concentration. This modification was due in part to work described in this report.

The Electric Power Research Institute (EPRI) demonstrated the use of this calculation method at TMI using 28 airtight EPICOR II liners for the demonstration. EPRI used a desktop computer with a spreadsheet program to compare the predicted hydrogen concentration from the INEL calculation method to the hydrogen concentration actually measured when the EPICOR II liners were vented during their preparation for shipment, after nearly 3 years of storage on site. On average, the predicted hydrogen concentration was within 20 percent of the measured values.

4.2.14 Catalyst Tests for Hydrogen Control in Canisters of Wet Rad Waste
(INEL, GEND-062, August 1987)

This report [44] described a unique test system for determining the effectiveness of catalyst beds to chemically combine hydrogen and oxygen gases in closed containers and without forced convection. The test system was used to determine the effects of: (●) catalyst types and mixtures; (●) catalyst bed size, geometry, and locations; (●) cover gas type and pressure; (●) catalyst bed wetting and drying; and (●) various additives and contaminants on catalyst performance. The catalyst test program, conducted under the direction of the DOE, provided a substantive basis for the design of passive catalyst beds in containers of wet radioactive materials. The objective was to select the catalyst types, quantities, arrangements, and environments that would reliably prevent the buildup of flammable mixtures of radiolytic hydrogen and oxygen gases in the debris canisters under normal, off-normal, and accident conditions.

The test program demonstrated that a mixture of specific Engelhard and Atomic Energy of Canada Limited (AECL) catalysts performed better than either catalyst performed separately. When the Engelhard catalyst was dry and well exposed to the reactive gases, its effectiveness was a factor of about 100 greater than when it was dripping wet. The AECL wet-proof catalyst was not as effective as the Engelhard catalyst, but the catalyst was not as sensitive to the presence of water. The various additives and contaminants to which the catalyst might be exposed during its life cycle had little effect on catalyst performance and contaminants appeared to be effectively removed by the rinsing processes.

4.3 Data Collection Activities

4.3.1 Axial Power Shaping Rod Insertion Test

- ***Purpose***. To individually move the eight axial power shaping rod assemblies in the core to provide insight into the extent of core and upper plenum damage. This early insight provided time to factor this information into plans for subsequent inspections for the head, upper plenum, and core removal.

- ***Evaluation: Hydrogen***. [45] The licensee's safety evaluation concluded that flammable concentrations in the reactor coolant system (RCS) would be avoided so long as the concentration of dissolved nitrogen was greater than that of oxygen (in cubic centimeters per kilogram (cc/kg)) by approximately a factor of 10. The control rod drive housing and reactor vessel upper plenum could contain hydrogen gas. However, the evaluation concluded that this gas, if present, would not constitute a hazard for this test. The composition in the gas spaces of the RCS was estimated from determinations of dissolved gas content of RCS water samples and the known solubility behavior of hydrogen, oxygen, and nitrogen in aqueous systems. Using simple gas laws, calculations confirmed that flammable concentrations in the RCS would be avoided as long as the concentration of dissolved nitrogen ($C(N_2)$) was about a factor of 10 greater than the concentration of dissolved oxygen ($C(O_2) < 0.098\ C(N_2)$ in cc/kg).

During the period of axial power shaping rod motion, primary coolant chemistry would be monitored every 24 hours. If the ratio of oxygen and nitrogen concentrations became greater than 0.098, testing would be stopped temporarily until the ratio was decreased below 0.098. Since $C(N_2)$ was expected to range about 6 cc/kg, this corresponded to a dissolved oxygen concentration of 0.59 cc/kg (0.48 parts per million). If $C(N_2)$ increased above the 6 cc/kg concentration level, the permissible dissolved oxygen concentration could be similarly increased.

- ***NRC Review***. [46] The NRC's evaluation focused on criticality.

4.3.2 Camera Insertion through Reactor Vessel Leadscrew Opening

- ***Purpose***. To insert a camera into the reactor vessel through a control rod drive mechanism (CRDM) leadscrew opening to provide the first visual assessment of conditions inside the reactor vessel. This activity was known as "Quick Look."

- ***Evaluation: Hydrogen***. [47] The licensee's safety evaluation concluded that the gas vented from the reactor coolant system (RCS) after depressurization and from the CRDM leadscrew withdrawal; therefore, cutting operations would not present a flammable mixture in the containment building.

  - *RCS Venting*. The venting of the RCS would result in the release of trapped gases including hydrogen to the containment building atmosphere. The total quantity of hydrogen in the RCS was estimated based on the following assumptions: (●) RCS water had not been processed. (●) The volume of free gas in the RCS was determined by a test that conservatively ignored differences in water head. (●) The ratio of hydrogen to nitrogen in the free gas was the same as the ratio in the RCS samples. (●) Henry's Law was applied for dissolved gases versus free gases.

    The RCS sample results indicated about 7 and 9 cubic centimeters per kilogram of hydrogen and nitrogen, respectively. Based on Henry's Law, results showed that about 40 percent of free gas would be hydrogen, and the remaining 60 percent would be nitrogen. Using the Ideal Gas Law ($PV = nRT$), about 370 standard cubic feet (scf) of free gas were in the RCS, of which 40 percent or 145 scf was hydrogen. Such a high concentration of hydrogen indicated that the RCS water contained about 700 to 800 scf of hydrogen; therefore, the total quantity of hydrogen in the RCS was conservatively estimated at 1000 scf.

The gas vented from the RCS was assumed to be 100-percent hydrogen and would be discharged into a dilution flow stream, with a minimum dilution factor of 25. This would prevent the discharge of a flammable hydrogen mixture to the containment building atmosphere. The discharge would be directed away from where personnel would be located.

- *Leadscrew Removal*. During leadscrew withdrawal and cutting operations, the RCS would be at atmosphere pressure. The RCS surface areas exposed to the atmosphere via the open CRDM motor tube would be less than 4 cubic inches. The evaluation concluded that such a low release rate would not present a hydrogen flammability hazard when vented directly to the containment building.

- ***NRC Review: Hydrogen***. [48] The NRC's safety evaluation considered the potential for a combustible gas burn in the containment building. The venting of the RCS before pressurization and lowering of the water level would result in the discharge of hydrogen to the containment building atmosphere. The NRC reviewed the licensee's procedures for the prevention of a combustible gas burn in the building during the venting. The venting of hydrogen from the RCS would be into a dilution flow stream with a dilution factor of at least 25. The discharge would take place in an area away from working personnel (inside one of the D-rings [j]). There would be no sources of ignition near the area of hydrogen discharge, as the discharge would effectively occur in a confined pipe duct. Lastly, the containment building purge system would be operating to prevent the buildup of any hydrogen gas inside the building.

4.3.3 Reactor Vessel Underhead Characterization (Radiation Characterization)

- ***Purpose***. To characterize radiation under the reactor vessel head to ensure that adequate radiological protection measures would be taken to keep radiation exposures ALARA during head removal. To achieve this objective, measurements were necessary with the reactor vessel head still in place.

- ***Evaluation: Hydrogen***. [49, 50] The licensee's safety evaluation concluded that hydrogen would not accumulate under the reactor vessel head. Previous analyses of reactor coolant system (RCS) liquid and RCS high-point gas samples indicated that RCS hydrogen evolution had not produced combustible gas mixtures in the RCS high points. Expected RCS hydrogen evolution rates during underhead characterization activities would remain below the level required to produce combustible gas mixtures in the RCS high points. Calculations of the potential flow of gas up the control rod drive mechanism (CRDM) with the manipulator tube installed resulted in an exchange rate of more than 5500 cubic feet per day. This corresponded to a free volume turnover of the gas space in the reactor vessel head of about seven volume changes per day. The evaluation concluded that with one CRDM manipulator tube open to the building atmosphere and the remaining CRDM closures inverted in their respective drives,

[j] D-rings were shield enclosures around the steam generator compartments; they were so named because of their shape.

sufficient air circulation would exist under the reactor vessel head to ensure that no hydrogen accumulation would occur.

- ***NRC Review: Hydrogen***. [51] The NRC's safety evaluation stated that combustible gas burn accumulations were performed for previous evaluations. The results of these evaluations combined with actual measurements and observations provided high confidence that the proposed underhead characterization study could adequately deal with this issue.

4.3.4 Reactor Vessel Underhead Characterization (Core Sampling)

- ***Purpose***. To obtain up to six specimens of the core debris using specially designed tools. These tools would be lowered into the reactor to extract samples of the loose debris and to load the samples into small, shielded casks for offsite shipment and analysis. The analysis of the samples would: (●) identify the composition of the particulate core debris; (●) determine particle size; (●) determine fission product content; (●) determine fission product leachability from the debris; (●) analyze the drying properties of the debris; and (●) determine whether pyrophoric materials existed in the core debris. This examination was a follow-on activity to the underhead characterization study.

- ***Background***. Three shipping casks [k] were considered [52] for shipping the core debris samples to the laboratory. The selected cask was the modified and recertified Model CNS 1-13C Type B shipping cask, with the sample in a Specification 2R container. An earlier version of this cask was used to ship spent submerged demineralizer system liners. [53]

- ***Evaluation: Hydrogen***. [54] The licensee's safety evaluation noted that if particulate debris had chemically bound water (water of hydration), then radiolytic gas generation could severely complicate offsite shipping of the material. This condition was encountered in preparing submerged demineralizer system liners that contained zeolite for shipment. The debris sample program consisted of activities similar to those performed during the core probe into the debris bed, which had been safely performed before. The evaluation concluded that the debris sample program would not present any undue risk.

- ***NRC Review***. [55] Editor's Note: The NRC's safety evaluation report did not specifically address this evaluation topic.

4.3.5 Core Stratification Sample Acquisition (NA)

[k] Editor's Note: While large shipping containers of radioactive materials may often be referred to as "shipping casks," the proper term for such containers, when loaded with contents and in their transportation configuration, is "package." See 10 CFR Part 71, "Packaging and Transportation of Radioactive Material," Section 71.4, "Definitions."

4.3.6 Use of Metal Disintegration Machining System to Cut Reactor Vessel Lower Head Wall Samples

- ***Purpose***. To remove metallurgical samples from the inner surface of the lower reactor vessel head using the metal disintegration machining (MDM) system. An international research program used these samples to study core melt and reactor vessel interactions. This program was conducted after the reactor vessel was defueled.

- ***Evaluation: Hydrogen***. [56, 57] The licensee's safety evaluation considered the potential for hydrogen evolution and submerged combustion.

  - *Hydrogen Evolution*. The MDM cutting equipment would generate hydrogen and oxygen gas during operation. The safety concern was that hydrogen gas could reach combustible concentrations on the work platform and in the containment building. During sampling activities, the shielded work platform would cover the reactor vessel. An off-gas system was designed to provide an air inflow through the top of the work platform. This system diluted gases that would evolve during sampling activities before the gases were released into the containment building. Therefore, any hydrogen evolved (calculated to be less than 1 standard cubic foot per minute) would be diluted by the off-gas treatment system, as required. Thus, the hydrogen would not reach a combustible concentration in the containment building. Other hydrogen-related safety issues, such as radiolytic generation of hydrogen in the canister transfer shield, in the fuel handling building, or in the containment building, were discussed in and bounded by the evaluations in the safety evaluation report [58] for defueling the reactor vessel.

  - *Submerged Combustion*. The MDM process would generate electrical sparks between an electrode and the material being cut. This heat source was not expected to create a combustion concern since the sparks were being generated underwater. However, theoretically, combustion of hydrogen and oxygen produced in the MDM process could occur between the electrode and material being cut. Combustion of fuel debris was not considered credible because no significant amounts of fuel were expected to be present in the material being cut by the MDM process, and no identified ignition was introduced with plasma arc cutting in and around fuel debris in the lower core support assembly.

- ***NRC Review: Hydrogen***. [59] The NRC's safety evaluation considered the potential for evolution and collection of combustible gases. The evaluation noted that potential hydrogen evolution would be less than 1 standard cubic foot per minute. The NRC's evaluation agreed with the licensee's evaluation that the existing defueling work platform off-gas system was adequate to preclude buildup of combustible concentrations of hydrogen. The NRC required that the off-gas system be functional and operating during MDM cutting operations. Any combustion at the cutting head would be localized and self-extinguishing due to being submerged under 40 feet of borated water.

4.4 Pre-Defueling Preparations

4.4.1 Containment Building Decontamination and Dose Reduction Activities (NA)

4.4.2 Reactor Coolant System Refill

- ***Purpose***. To refill the reactor coolant system (RCS) to the top of the hot legs in order to purge oxygen and to provide an RCS water level that would permit operation of the once-through steam generator (OTSG) recirculation/cleanup system. To operate the OTSG recirculation/cleanup system, the secondary-side water level in the OTSG must be raised to the vicinity of the upper tubesheet to minimize the chance of unborated water leakage from the OTSGs to the RCS.

As an added measure of protection against system overpressurization, the pressurizer would not be vented. This protective measure provided a surge volume for increases to the RCS or for inadvertent introduction of pressurization to the RCS, such as by activating pumps or changing valve lineups.

- ***Evaluation: Hydrogen***. [60, 61] The licensee's safety evaluation considered the potential for the buildup of a flammable gas mixture in the RCS. The occurrence of a flammable mixture was not considered credible for the following reasons: (●) The release rate of hydrogen from the RCS was less than 0.010 cubic foot per day. (●) Most of the hydrogen released in the reactor vessel would accumulate in the center control rod drive mechanism (CRDM). (●) The CRDM contained a nitrogen-hydrogen atmosphere but no oxygen. This gas mixture would not be contaminated with building air before refill; thus, it is a nonflammable mixture. (●) Hydrogen that did not accumulate in the center CRDM but ended in other CRDMs would not reach a 4-percent hydrogen concentration for many years. Head removal would occur before the CRDMs reached a flammable mixture.

- ***NRC Review: Hydrogen***. [62] The NRC's safety evaluation considered the hazard from combustible gases. The evaluation stated that by pressurizing the RCS to about 70 pounds per square inch gauge, combustible gases generated by radiolysis within the damaged reactor core would remain dissolved in the pressurized coolant. These potential combustible gases would be safely vented and purged from the RCS before opening the system, as was done before the Quick Look task.

4.4.3 Reactor Vessel Head Removal Operations

4.4.3.1 Polar Crane Load Test (NA)

4.4.3.2 First-Pass Stud Detensioning for Head Removal (NA)

4.4.3.3 Reactor Vessel Head Removal Operations

- ***Purpose***. To prepare for reactor vessel head removal to perform the lift and transfer of the head and conduct the activities necessary to place and maintain the reactor coolant system (RCS) in a stable configuration for the next phase of the recovery effort.

- ***Evaluation: Hydrogen***. [63] The licensee's safety evaluation concluded that hydrogen evolution would be acceptably low. As documented in a previous evaluation, [64] analyses of RCS liquid and RCS high-point gas samples indicated that hydrogen evolution had not produced combustible gas mixtures in the RCS. Nevertheless, during the underhead characterization, the reactor vessel head would be vented so there should be no significant accumulation of hydrogen in the vessel before the head lift.

During the venting process, there would be sufficient dilution of the vented gas to preclude the accumulation of a combustible gas mixture in the containment building. During and following the head lift activities, the reactor vessel would be open to the containment building atmosphere. This allowed the small quantity of evolved hydrogen to diffuse into the containment building atmosphere without reaching combustible mixture concentrations in the RCS or in the containment building.

- ***NRC Review: Hydrogen***. [65] The NRC's safety evaluation concluded that there was little potential for a combustible gas reaction during reactor vessel head lift activities. As a result of the radiolytic decomposition of reactor coolant water into gaseous hydrogen and oxygen, there was the potential for combustible gas mixtures to form underneath the head. Those mixtures required a hydrogen concentration of at least 4 percent in the presence of oxygen at 5 percent or greater. Accordingly, the licensee measured the hydrogen generation rate from the reactor coolant during prior cleanup activities and noted the rate was very low (0.01 cubic foot per day or less). Additionally, during head lift, the reactor pressure would be in the depressurized, vented condition, and any generated hydrogen would rapidly diffuse into and mix in the containment building atmosphere. During the underhead characterization study conducted in 1983/1984, the reactor vessel was in a condition (i.e., depressurized and vented) that maintained the head lift conditions for a period of 10 months. This experience demonstrated that the measures were adequate to prevent the buildup of combustible gases.

4.4.4 Heavy Load Handling inside Containment (NA)

4.4.5 Heavy Load Handling over the Reactor Vessel (NA)

4.4.6 Plenum Assembly Removal Preparatory Activities (NA)

4.4.7 Plenum Assembly Removal (NA)

4.4.8 Makeup and Purification Demineralizer Resin Sampling

- ***Purpose***. To obtain resin samples from the two makeup and purification demineralizers. Resin samples were required to characterize the present resin conditions for the development of a technically sound resin removal and disposal program.

- ***Evaluation: Hydrogen***. [66] The licensee's safety evaluation concluded that based, in part, on the gas analysis, the hydrogen generation rates were not sufficient to generate explosive hydrogen levels even if allowed to collect for 30 days.

  - *Gas Sampling*. The analyses of the gas samples from the tests showed high hydrogen content, low oxygen content, and varying amounts of organic molecules. These results were consistent with those observed during the EPICOR II prefilter liner venting and gas generation studies. Following the acquisition of gas samples, the demineralizers were purged of their hydrogen content by repeated pressurization and venting using nitrogen. The resulting atmosphere contained less than 2 percent residual hydrogen. The demineralizers were then allowed to remain in a pressurized condition and then sampled. They would be analyzed periodically for hydrogen buildup.

  - *Hydrogen Generation Rate*. Using the model developed from the EPICOR II prefilter liner venting studies, an estimate of the hydrogen generation rate was calculated. The results of the gamma spectrometry on Demineralizer-A indicated a cesium-137 content of 3400 curies ± 2500 curies. The hydrogen gas generation constant from the EPICOR work was about $6x10^{-6}$ liter per hour per curie of total activity. Using this rate constant and the makeup demineralizer activity, a calculated hydrogen generation rate of 0.96 liter per day would result. This would yield a pressure increase of nearly 3 pounds per square inch on Demineralizer-A due to hydrogen generation per year. The absence of significant quantities of water had been confirmed by gamma scans and Compton spectrometry.

  - *Explosive Concentrations*. If during the sampling operations, the tank atmospheres were permitted to exchange with the ambient atmosphere, the resulting gases would remain nonexplosive (i.e., less than 2 percent hydrogen). This would remain true for a period of 34 days even if the EPICOR hydrogen gas generation rate was assumed, and the oxygen content was at 20 percent (present oxygen levels were less than 1 percent). While there would be momentary openings of the depressurized demineralizers for the purpose of inserting tools, it was not planned to purge the demineralizers with air; thus, minimum oxygen would be introduced. The capability of repressurization and venting would exist for the purpose of inerting after sampling.

- ***NRC Review***. Editor's Note: The NRC's safety evaluation was not located.

4.4.9 Makeup and Purification Demineralizer Cesium Elution

- ***Purpose***. To remove most of the radioactivity from the resins while they were in the demineralizers to the extent that standard resin sluice procedures could complete the task. The scope of this evaluation included only the first phase of a three-phase process for disposition of the makeup and purification of resins. This first phase included the rinse and elution of the demineralizer resins. The latter two phases would include the sluicing, removal, solidification or other packaging and disposal of these resins. Separate safety evaluations would address the latter phases.

- ***Evaluation: Hydrogen***. [(67)] The licensee's safety evaluation concluded that radiolytic gas generation would not present any hazard. Hydrogen gas was generated in the demineralizers as a result of radiolytic breakup of water in a radiation field. This generation rate was measured in Demineralizer-B at 1.4 liters per day and was calculated to be no more than 0.25 liter per day for Demineralizer-A. These rates of hydrogen generation would not inhibit any phase of the operations relating to this rinse and elution process. Hydrogen generated inside the tanks presented no detonation problem because the small generation rate coupled with removal of gas via the waste gas decay system would not allow buildup of hydrogen to the unsafe level required for detonation. The existing auxiliary building ventilation system would remove any hydrogen released from the tank into the cubicle.

- ***NRC Review: Hydrogen***. [(68)] The NRC's safety evaluation stated that radiolytic generation of hydrogen gas in the demineralizer with the highest cesium loading was measured to be less than 1.4 liters per day. The low generation rate in conjunction with continuous venting of the demineralizers to the waste gas system would preclude buildup of combustible gas concentrations. Instrumentation was provided to monitor pressure in the vessels, which would allow early detection of any gas buildup in the event of waste gas system failure. Additionally, all cubicle spaces and control stations would be well ventilated to prevent accumulation of combustible gases in the unlikely event of system leakage.

4.5 Defueling Tools and Systems

4.5.1 Internals Indexing Fixture Water Processing System (NA)

4.5.2 Defueling Water Cleanup

4.5.2.1 Defueling Water Cleanup System

- ***Purpose***. To remove organic carbon, radioactive ions, and particulate matter from the reactor vessel and fuel transfer canal/spent fuel pool "A" (FTC/SFP-A). The defueling water cleanup system (DWCS) actually comprised two independent subsystems (sometimes called trains): the reactor vessel cleanup system and the FTC/SFP-A cleanup system. The DWCS was totally contained within areas that had controlled ventilation and could be isolated. This limited the environmental impact of the system during normal system operations, shutdown, or postulated accident conditions.

- ***Evaluation: Hydrogen***. [69] Editor's Note: The hydrogen evaluation was focused on its effect on radiological releases as a motive force for relieving gases and particulates through the safety relief valve of the submerged demineralizer system liner. Refer to NUREG/KM Chapter 10 on radiological release evaluations.

- ***NRC Review***. [70] Editor's Note: The NRC's safety evaluation report did not specifically address this evaluation topic.

4.5.2.2 Cross-Connect to Reactor Vessel Cleanup System (NA)

4.5.2.3 Temporary Reactor Vessel Filtration System (NA)

4.5.2.4 Filter-Aid Feed System and Use of Diatomaceous Earth as Feed Material

- ***Purpose***. To add a feed material into the filter canisters to promote the buildup of cake on the filter media, thereby significantly improving the performance of the defueling water cleanup system filter canisters. A filter-aid feed system that used diatomaceous earth (DE) as the feed material was installed as an ancillary system to the defueling water cleanup system (DWCS).

- ***Evaluation: Hydrogen***. [71] The licensee's safety evaluation considered gas generation control. Gas generation in the defueling canisters (including the DWCS filter canisters) was controlled by the presence of a recombining catalyst in the canisters. The use of DE as a filter-aid material would not inhibit the performance of the catalyst in the defueling canisters. DE was inert and would not chemically react with the catalyst. Additionally, the very characteristics and consistency of DE, which made the material an ideal filter-aid material, prevented the material from isolating the catalyst from generated hydrogen and oxygen. This was true even if the DE were to settle on the catalyst retainer screens or the catalyst material (i.e., filter-aid DE would not "pack down" to form an impenetrable barrier against gases).

- ***NRC Review: Hydrogen***. [72] The NRC's safety evaluation concluded that the addition of the DE, which was composed of primarily silicon dioxide, to the filter canisters would not adversely affect the operation of the catalytic recombiners nor would the material affect the predicted radiolytic gas generation rate in a closed canister.

4.5.2.5 Use of Coagulants

- ***Purpose***. To demonstrate the use of coagulants and body-feed material to improve the performance of the defueling water cleanup system (DWCS) filter canisters in maintaining water clarity. Operating experience with the DWCS had not achieved the desired level of clarity in the reactor coolant system (RCS) water to support defueling operations within the reactor vessel. The DWCS filters required changeout because of high differential pressure without the expected high filter throughput. The root cause of shortened filter canister life was expected to be the presence of hydrated metallic oxides in colloidal suspension within the RCS that were plugging the filter media. The addition of the coagulant with body-feed was expected to agglomerate the colloids to filterable sizes, thus forming a filter cake on the filter media.

- ***Background***. The first coagulant evaluated (73) was about 20 weight percent of $C_8H_{16}NCl$ and 80 weight percent unborated water when undiluted. The body-feed material was diatomaceous earth. The second alternate coagulant evaluated (74) that showed a greater potential as a filter aid was the polymer melamine-formaldehyde. The undiluted solution of this coagulant was about 8 percent of the polymer and 92 percent unborated water. The expected dosage of the undiluted solution to the DWCS processing stream was 10 to 20 parts per million (ppm) with a maximum dosage of 50 ppm. Both safety evaluations were similar; the first evaluation is discussed in this NUREG/KM.

- ***Evaluation: Hydrogen***. (75,76) The licensee's safety evaluation noted that canister shipping requirements that could be impacted by the addition of coagulant and body-feed material in the defueling canisters, were criticality and gas generation and control. Canister criticality evaluations would be submitted for NRC approval before shipping the filter canisters. The body-feed material, which was essentially silicon dioxide, was shown in a previous safety evaluation report (77) not to affect the recombiner catalyst installed in the canisters to control radiolysis-generated hydrogen and oxygen. Radiolytic breakdown of the coagulant could generate additional hydrogen and other gases. However, the limiting gas concentration for determination of the allowable storage and shipping time for dewatered canisters would be hydrogen. The allowable storage and shipping time would be determined based on actual hydrogen appearance rates obtained from gas samples of dewatered canisters. Testing would also verify that a catalyst exposed to a coagulant with a concentration of 50 parts per million had no adverse impact on the recombiner catalyst installed in the canisters.

- ***NRC Review***. (78) Editor's Note: The NRC's safety evaluation report did not specifically address this evaluation topic.

4.5.2.6 Filter Canister Media Modification (NA)

4.5.2.7 Addition of a Biocide to the Reactor Coolant System

- ***Purpose***. To evaluate the effects of adding hydrogen peroxide as a biocide to the reactor coolant system (RCS) to aid the removal of biological contamination. This biological growth reduced water clarity and fouled the water processing system filters.

- ***Evaluation: Hydrogen***. (79) The licensee's safety evaluation noted that biocide compatibility with RCS chemistry and processing capability was paramount in consideration for use in the

RCS. Consequently, any deleterious effect on canister catalyst effectiveness was addressed in the licensee's safety evaluation with regard to using hydrogen peroxide (H_2O_2) as a disinfectant in the RCS. The effect of the low concentration of H_2O_2 on the effectiveness of the defueling canister catalyst had been checked through laboratory testing. Catalyst performance was tested in the DOE's Hanford Operations with simulated TMI-2 RCS fluid at 500 parts per million of H_2O_2 without lasting serious detrimental effects.

- ***NRC Review: Hydrogen***. [(80)] The NRC's safety evaluation noted that laboratory testing showed that hydrogen peroxide did not cause a permanent degradation of the catalyst material used in the defueling canisters. The agency concluded that the proposed addition of hydrogen peroxide in sufficient quantity to attain a 200-part-per-million residual in the reactor coolant system would not pose a significant risk to the health and safety of the public or the occupational workforce and would not involve an unreviewed safety question.

4.5.3 Defueling Canisters and Operations

4.5.3.1 Defueling Canisters: Filter, Knockout, and Fuel

- ***Purpose***. To provide loading, handling, and storage of the canisters (filter, knockout, and fuel) for the long-term storage of core debris, ranging from very small fines to partial length fuel assemblies.

- ***Evaluation: Hydrogen***. [(81)] The licensee's safety evaluation considered hydrogen evolution during the design of the defueling canisters. A generic feature of the canisters was the recombiner catalyst packages incorporated into the upper and lower heads of all the canisters. The catalyst recombined the hydrogen and oxygen gases formed by radiolytic decomposition of water trapped in the damp debris. This reduced the buildup of internal pressure in the canister and kept the gases below the flammability limit. The redundant locations ensured that an adequate amount of catalyst was available for any canister orientation where hydrogen might be generated (e.g., an accidental drop that left a canister upside down). Test results by the DOE's Hanford Operations, which were documented in GEND-051, [(82)] "Evaluation of Special Safety Issues Associated with Handling the Three Mile Island Unit 2 Core Debris," showed that the catalyst would perform effectively while dripping wet but not when submerged. Specifically, the evaluation considered: (●) minimum catalyst quantity; (●) overpressure protection; (●) gas generation rate; (●) time to reach design pressure; (●) stuck-open relief valve; and (●) hydrogen ignition.

  - *Minimum Catalyst Quantity*. A total of 200 grams of catalyst was initially installed in each canister. Then, extra catalyst was installed in the catalyst beds to fill remaining voids. The 200-gram quantity was determined from the catalyst tests conducted by the DOE's Hanford Operations. These tests used 100 grams of catalysts and a hydrogen/oxygen generator that simulated the maximum gas generation of 0.076 liter per hour hydrogen. Additionally, the beds were designed to meet the shape and volume requirements established by the tested catalyst beds. A total of at least 200 grams of catalyst was installed in the canister to ensure that at least 100 grams was above the maximum water level for all canister orientations. At least 100 grams of catalyst was at either end of the canister, and the bed arrangement was symmetrical at each end.

- *Overpressure Protection*. Each canister was installed with two safety relief valves. The relief setpoint for the first valve was set at 25 pounds per square inch gauge (psig) pressure. To address the issue of canister pressurization resulting from failure of the 25-psig relief valve, a second relief valve was installed on the canisters. This second relief valve would ensure that canister pressure would not exceed the design limit of 150 psig and would make the canister single-failure proof with regards to pressurization. This second valve would also be installed in a manner that would eliminate common mode failure of the two pressure relief valves.

- *Gas Generation Rate*. The maximum predicted gas generation rate in a canister was determined by two separate models: (●) maximum theoretical gas generation rate and (●) maximum realistic gas generation rate. The maximum theoretical gas generation rate was determined by the DOE's Hanford Operations for the purpose of developing the catalytic recombiner bed design (refer to GEND-051 for details). The licensee determined the maximum realistic gas generation rates for use in predicting canister internal pressures during periods when the canisters were water solid. Both models were based on the 1968 paper [83] "Radiolytic Decomposition of Water in Water-Moderated Reactors Under Accident Conditions." Since the quantity of contaminants in the reactor coolant system was small, the generation of other gases from the radiolytic decomposition of these contaminants was not expected to be significant.

  Using the maximum realistic gas generation rate of 0.0075 liter per hour and assuming no recombination or scavenging of oxygen, the relief valve with a 25-psig lift setpoint was estimated to first open in about 25 days for the worst case canister. Released gas would be vented through the pool water directly to the containment building or fuel handling building and was of such a small quantity that it would cause no combustion concerns in the atmosphere of these buildings.

- *Time to Reach Design Pressure*. The recombiner catalyst was ineffective when it was underwater. An evaluation showed that it took 139 days for a non-dewatered canister to reach 150 psig if the 25-psig relief valve failed in the closed position. A similar concern existed for the dewatered canister should a significant amount of oxygen scavenging occur and the 25-psig relief valve failed closed. Assuming no recombination (i.e., complete oxygen scavenging), the canister would reach the design pressure in 4286 days for the worst-case canister.

- *Stuck-Open Relief Valve*. There was the possibility that fuel debris could be released into the pool water if the relief valve should fail open while the canister was being stored. If contaminants were released into the pool, the defueling water cleanup system (DWCS) could be used as necessary to limit the contamination level of the water. Therefore, a failed-open relief valve would not pose a safety concern. Additionally, given that it was planned, although not required, to dewater the canisters shortly after they were loaded, pressurization of the canisters caused by nitrogen/oxygen generation would be minimal, and the relief valve was not expected to open.

- *Hydrogen Ignition*. Although not considered a credible event, the consequences of hydrogen ignition inside a canister were evaluated. The maximum pressure that could be reached inside a canister under normal conditions, because of the 25-psig relief valve, was about 42 pounds per square inch (psi) absolute pressure. This pressure included the 25-psig set pressure and 5 feet of water submergence. Under the assumption that the recombiner catalyst did not function properly, a flammable mixture of hydrogen and oxygen could accumulate within a canister. If ignition of this mixture was postulated, then an overpressurization of the canister could occur. The ultimate stresses that would be reached for various canister components at the estimated pressures included: (●) 2160 psi for the canister shell, (●) 2900 psi for the fuel canister bolts, and (●) 2500 psi for the threaded connections. Considering the large margin that would exist between these pressures and the maximum, normal condition canister pressure (i.e., about a factor of 50), the overpressurization resulting from an ignition of hydrogen within the canister was not expected to affect the overall canister integrity.

- ***NRC Reviews: Hydrogen***. [84] The NRC's safety evaluation noted that after filling a canister with fuel debris, water would remain in the canister. Before dewatering, the canister would be completely flooded with reactor coolant system water. Following dewatering, the canister would contain residual water that was entrained in the fuel debris, as well as a certain amount of free "slosh" water not removed by the dewatering system. Since this water would be in direct contact with fuel and fission products that contained debris, without benefit of the fuel cladding to provide shielding from alpha and beta radiation, there could be significant hydrogen and oxygen generation from radiolytic decomposition of the water. Gas generation would result in internal pressure buildup and production of combustible gas mixtures inside the canisters. The DOE's Hanford Operations performed studies to predict the rate of gas generation and to develop suitable catalytic recombiners to control the gas concentrations. The NRC's safety evaluation considered: (●) gas generation rate; (●) minimum catalyst quantity; (●) catalyst test insights; and (●) catalyst positions.

- *Gas Generation Rate*. The rate of gas generation was shown to be a function of the following: (●) amount of ionizing radiation emitted by debris in a canister; (●) fraction of the energy absorbed in the water; (●) ratio of peak-to-average decay heat energy in the fuel debris, and (●) amount of gas produced per unit of energy. Using the empirical relationship that was confirmed experimentally, the maximum theoretical gas generation rate was predicted as 0.114 liter per hour of hydrogen plus oxygen in stoichiometric proportions. The licensee's evaluation stated that there was significant conservatism in this calculation and provided what was considered a maximum realistic generation rate based on more probable conditions in the core debris. The licensee's predicted maximum realistic gas generation rate was 0.0075 liter per hour.

The conservatisms in the theoretical predictions were as follows: (●) The maximum quantity of fuel in a canister used in the calculations (800 kilograms) would not include allowances for residual water or for weighing accuracy; in the "realistic" prediction, this quantity would decrease. (●) In calculating the fraction of energy absorbed in the water, it was conservatively assumed that large amounts of water were present for absorption rather than using the maximum amount of water that could possibly be present in a filled canister. (●) The amount of gas produced per unit of absorbed energy assumed no oxygen scavenging (i.e., chemical removal) that would produce excess hydrogen and resultant back-reactions. (●) The ratio of peak-to-average decay heat energy was based on the most active region of an undamaged core and did not account for possible dispersal of the material from this core region during the accident.

The NRC reviewed the basis for the gas generation rates and concurred that there was significant conservatism in the theoretical generation rate. However, the data presented in the licensee's technical evaluation report [85] for the defueling canister were insufficient to justify the use of the licensee's lower predicted "realistic" rate or to accurately quantify the conservatisms in the theoretical calculations. Therefore, the NRC's safety evaluation was based on the maximum theoretical gas generation rate of 0.114 liter per hour.

- *Minimum Catalyst Quantity*. Following a series of tests by the DOE's Hanford Operations, the catalyst chosen for use in the defueling canisters was a mixture of 80-percent Engelhard Deoxo-D nuclear grade catalyst and 20-percent silicone-coated catalyst from Atomic Energy of Canada Limited. GEND-051 [86] documented details of the catalyst test program. The test program involved a catalyst bed similar to that in the canisters. The catalyst bed was installed in a test chamber where hydrogen and oxygen were admitted at a controlled rate. The test chamber's pressure and temperatures were monitored, and its internal atmosphere was sampled and analyzed.

  The tests demonstrated that the designed catalyst beds containing 100 grams of catalyst in the required proportions were capable of maintaining the chamber atmosphere below 1.2-percent hydrogen and 0.6-percent oxygen, while recombining the gases at a rate of 0.3 liter per hour of hydrogen plus oxygen in stoichiometric proportions. This showed significant margins of safety from the lower flammability limits of 5-percent oxygen and 4-percent hydrogen, and from the maximum theoretical gas generation rate of 0.114 liter per hour.

- *Catalyst Test Insights*. Other testing of canister catalysts yielded the following insights: (●) Catalysts did not function when immersed in water. After immersion and being "drip dried" in a 100-percent relative humidity atmosphere, the catalysts began recombination at a reduced rate. The rate increased and reached full capacity within a short time as the heat generated by the recombination reaction caused further drying of the catalyst. (●) Chemical species expected to come in contact with the catalyst from the reactor coolant system or during canister fabrication did not have any deleterious effects on the catalyst performance. (●) Freezing conditions during transportation did not stop the recombination reaction once started.

- *Catalyst Positions*. The catalyst beds installed in the defueling canisters were designed so that at least 100 grams of the catalyst would not be immersed in water regardless of canister orientation as long as the canister was no more than half full of free water. Four recombiner packages, each containing 25 grams of catalyst, were attached symmetrically about the axis of the inner surface of the lower canister head in all types of canisters. The upper head of the fuel canister had one large-diameter flat catalyst bed containing 100 grams of catalyst on the inner surface. The knockout and filter canisters contained two symmetrically located beds containing 50 grams each of catalyst in the upper heads. All catalyst cartridges were welded in place and structurally designed to remain intact and functional, provided they were not immersed, during any postulated drop accident. The catalyst material was covered by a retainer screen that held it in place but allowed free diffusion of gas to the catalyst surface and diffusion of water vapor away from the catalyst.

- *Conclusion*. Based on a review of the licensee's evaluation and available literature on radiolytic decomposition, the NRC's safety evaluation concluded that the maximum theoretical gas generation rate was predicted with considerable conservatism. The evaluation further concluded that the designed catalytic recombiners had acceptable margins of safety and provided reasonable assurance that combustible gas mixtures would not develop in the filled canisters after dewatering.

4.5.3.2 Replacement of Loaded Fuel Canister Head Gaskets

- ***Purpose***. To replace head gaskets on the loaded fuel canisters in spent fuel pool "A" (SFP-A) because of excessive leakage. The replacement of gaskets involved: (●) moving a loaded canister from the SFP-A storage rack to the dewatering station rack with the canister handling bridge; (●) removing the canister head; and (●) transferring the head to a worktable. At the worktable: (●) the head was rotated for access to the gaskets; (●) the metallic gaskets were removed; (●) new synthetic gaskets were installed; and (●) the head was inspected before reinstallation.

- ***Evaluation: Hydrogen***. [87] The licensee's safety evaluation concluded that no pressure buildup within the canister could occur and hydrogen evolution was not a safety concern. The presence of fuel in the canister could result in radiolytic decomposition of the water within the canister. The temporary cover placed on the open canister to minimize the potential for the spread of contamination to SFP-A did not provide a seal. Thus, the canister would be open to the pool, readily releasing any hydrogen generated to the fuel pool water and then to the ventilation system of the fuel handling building.

- ***NRC Review***. [88] Editor's Note: The NRC's safety evaluation report did not specifically address this evaluation topic.

4.5.3.3 Use of Debris Containers for Removing End Fittings

- ***Purpose***. To use modified fuel canisters as "debris containers" for removing fuel assembly upper end fittings, control component spiders, or other structural material from the reactor vessel. This activity was performed to expedite access to the vacuumable fuel and debris in the core. The modified fuel canister did not have internal neutron-absorbing plates, concrete filler, recombiner catalyst, dewatering capability, or a relief valve. After the debris containers were loaded, they would be closed and stored in the spent fuel pool "A" racks until final dispositioning of the containers and their contents. There were no plans to utilize these debris containers for shipment. Since these canisters would not have relief valves installed (a prerequisite for shipping), they could be easily be identified.

- ***Evaluation: Hydrogen***. [89] The licensee's safety evaluation concluded that no pressure buildup within the container could occur. The presence of fuel in a container could result in radiolytic decomposition of the water within the containers. However, given the minimal amount of fuel in each container, the hydrogen generation rate was expected to be minor. With the container vented, the minor amounts of gas generated would be readily released to the fuel pool water and then to the ventilation system of the fuel handling building.

- ***NRC Review***. [90] Editor's Note: The NRC's safety evaluation report did not specifically address this evaluation topic.

4.5.3.4 Fuel Canister Storage Racks (NA)

4.5.3.5 Canister Handling and Preparation for Shipment

- ***Purpose***. To transfer defueling canisters from spent fuel pool "A" (SFP-A) to the shipping cask for offsite shipment. Defueling canisters were transferred to locations within the containment building and fuel handling building using a transfer shield. The transfer of canisters to the shipping cask used a different device called a "fuel transfer cask."

- ***Evaluation: Hydrogen***. [91, 92] The licensee's safety evaluation of hydrogen evolution focused on the verification of the catalyst function before shipment. Two verification approaches were required for each canister—the verification of gas control and the verification of dewatering. Acceptance criteria were developed and modified to ensure that the hydrogen/oxygen recombiner catalyst was not submerged in any defueling canister orientation.

  - *Background*. Hydrogen generation inside a defueling canister was controlled through the use of a recombiner catalyst located at both ends of the canister. The catalyst recombines hydrogen and oxygen gases to water. To ensure that half of the total 200 grams of catalyst was above the maximum water level for all canister orientations (called the basic acceptance criterion), dewatering was performed to remove enough water to ensure that a minimum of 50 percent of the catalyst was not submerged in any canister orientation. The canister was weighed before and after it was dewatered to determine the actual volume of water removed. If this amount exceeded 50 percent of the empty free volume of water removed, then the canister was considered to be sufficiently dewatered for offsite shipment. Further, before shipment, each canister would be monitored for gas control.

In early 1987, the NRC approved [93] a licensee request [94, 95] to reduce the dewatering criteria from 50 percent to 25 percent of the canister void volume for fuel canisters and to the level required to ensure exposure of at least 25 grams of recombiner catalyst in the filter and knockout canisters. This approval was based on various recombiner catalyst tests that demonstrated a minimum factor of safety of 6 for 100 grams of approved catalyst. This corresponded to a safety factor (SF) of 1.5 for 25 grams of exposed catalyst. Revision 4 of the safety evaluation report (SER) for canister handling and preparation for shipment proposed a different method that allowed fluctuation in the following variables: catalyst SF equating to 100 grams of catalyst, minimum quantity of exposed catalyst, and decay heat load of the canister based on the canister payload weight.

- *Verification of Gas Control*. Before shipment, each canister would be monitored for gas control. The safety evaluation provided two methods for verifying gas control inside the canister to ensure the catalyst function in the defueling canisters before shipment, including gas sampling and pressure and temperature measurements.

  - *Gas Sampling Method*. The canister could be sampled for the presence of hydrogen and oxygen to determine gas appearance rates after a set holding period.

    - *Sampling Procedure*. The sampling would take place after the dewatering and following an adequate holding period. The dewatering would occur either in the vessel or at the dewatering station. Following the dewatering, the canisters would be brought to the storage racks where they would remain until sampled. The period of time that they were allowed to sit would be determined by sampling each type of canister several times. This would determine optimum holding times and general gas appearance rates. The holding time was estimated to be between 2 and 4 weeks.

      Following the holding period, an evacuated sample vessel of 150 to 300 cubic centimeters was connected to a long-handled tool and then connected to the purge inlet connection on the canister. The sample was obtained by opening a remote valve connected to the sampling vessel, and the vessel was then retrieved. The volume of the sample would ensure that the tool was purged while also obtaining a part of the top volume of the canister. This sample would be conservative since the possible stratification of the gases would tend to migrate to the top of the canister.

      After the sample was taken, the canister could be dewatered again. The argon cover gas pressure would be checked and brought to about 2 atmospheres, any remaining relief valves would be removed, and the canister purge and drain lines would be plugged. The canister was then taken back to the storage racks inside SFP-A, after performance of the bubble test for canister integrity and weighing of the dewatered canisters, as described in Section 2.3.1 of the SER.

- *Allowable Storage Times*. The contents of the sample vessel would be analyzed for the concentration of hydrogen, oxygen, and nitrogen. The nitrogen concentration was obtained to determine air in-leakage into the sample vessel. The oxygen associated with the air was discounted from the gas generation rate. The gas appearance rate would be determined by taking the hydrogen and oxygen concentrations and dividing by the time elapsed between initial dewatering and obtaining the sample. This rate was used to determine the allowable storage time at TMI for canisters that had their relief valves removed (with the relief valves in place, the canisters could be stored in SFP-A indefinitely). The allowable storage time started with the canister final dewatering or purging (when the relief valves would be removed) and ended with the departure of the shipping cask. This represented the period that the canister could remain stored at TMI, while still ensuring that the canister could be shipped without violating the allowable gas concentrations. This time would be determined by the following equation: [$A = (C/2R) - P$], where A was the allowed storage time in days; C was the maximum allowed concentration of hydrogen and oxygen in percent; R was the gas appearance rate in percent per day; and P was the shipping period in days.

- *Worst Case Canister*. In a calculation for a worst case canister ready for shipment, the pressure in the canister following a 1-year buildup of radiolytic gases was 50 pounds per square inch gauge (psig) pressure. This worst case canister was a dewatered canister with no allowable storage time at TMI-2 (i.e., term *A,* for the equation given in the preceding paragraph, was 6 days for cask loading time). This pressure of 50 psig was much less than the maximum internal canister pressures calculated in the SER [96] for the Model 125-B fueling canister shipping cask.

- *Exceeding Allowable Storage Time*. If the canister was stored for a period of time that approached the allowable storage time, the canister could be purged again and stored further at TMI for the previously determined allowable storage time. If the sample taken showed a higher than acceptable gas concentration (i.e., a lower than allowable storage time), the canister was brought back to the dewatering station and the 150-psig relief valve was replaced. The canister was then returned to the storage rack and allowed to sit for the appropriate waiting period. After the waiting period, the canister was sampled again, and a new appearance rate was calculated based on this and the previous sample. If this rate produced an acceptable concentration, the canister was considered ready for shipment. If this rate was unacceptable, steps would be taken to resolve the problem on a case-by-case basis.

– *Pressure/Temperature Measurements*. An alternative method for verifying gas control was to measure canister pressure and temperature following the initial dewatering. After a holding period, the pressure and temperature would be checked again. If the pressure increase was within certain limits when appropriate temperature corrections were applied, the canister was considered ready for shipment (refer to the table in Section 2.3.1 of the SER). This method could not be used if gas release from the canister was suspected before measurement of the pressure.

- *Verification of Dewatering*. Dewatering was performed to purge the canister of removable freestanding water. Completion of dewatering was detected by observing a steady stream of argon flow through the canister in the dewatering system. Following dewatering, the void volume was determined by the volume of water removed, which was obtained by the difference of the canister weights before and after dewatering. The acceptance criterion for canister dewatering was a dewatered canister void volume that ensured a sufficient quantity of exposed catalyst in any canister orientation.

- *Acceptance Criterion*. An acceptable quantity of exposed catalyst was 50 percent more than the required catalyst amount.

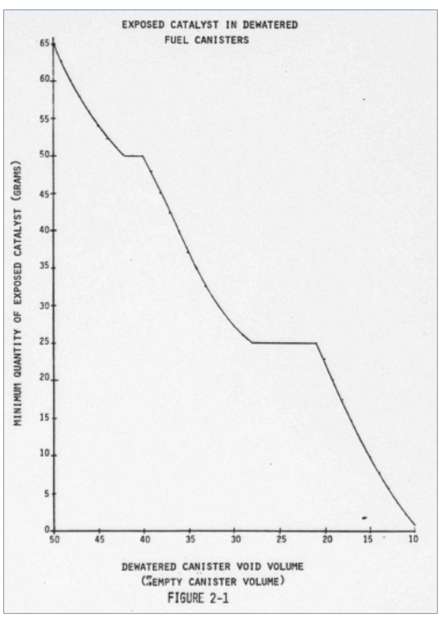
  - *Sufficient Catalyst Quantity*. The required quantity of catalyst was determined by laboratory testing and depended on the *radiolytic gas generation rate* within the canister. Laboratory testing involved subjecting 100 grams of catalyst to contamination that could occur during canister fabrication and loading. Chemical additions that would improve defueling water cleanup system filter performance and control microbiological growth in the reactor coolant system were also included. The contaminated catalyst was placed in a chamber and was then brought to 2 atmospheres of argon. Stoichiometric hydrogen and oxygen were injected into the chamber at a constant rate of 0.3 liter per hour. During the test, oxygen concentrations and catalyst recombination rates were measured and recorded.

  - *Effectiveness Safety Factor*. The effectiveness of the 100 grams of catalyst was expressed as an SF. The SF was defined as the product of two ratios at the time of peak oxygen concentration: [SF = ((measured combination rate)/(required minimum recombination rate)) x ((allowable maximum oxygen concentration)/(measured oxygen concentration))]. The *required minimum recombination rate* was equal to the expected radiolytic gas generation rate in the canister, and the *allowable maximum oxygen concentration* was equal to 5 volume percent. For conservatism, the expected radiolytic gas generation rate was assumed to be the probable maximum generation rate of 0.11 liter per hour (per 100 watts of decay heat) calculated in GEND-051 [97] and referred to as the "maximum theoretical rate" in the technical evaluation report [98] for defueling canisters. Since the gas generation rate was proportional to the decay head load (W) in the canister, the SF was inversely proportional to W. Therefore, the SF for 100 grams of catalyst in a canister with decay head load of W (watts) was [(100/W) x SF]. The required quantity of exposed catalyst was [(100 grams/100/W) x SF = W/SF grams]; thus, a sufficient quantity of exposed catalyst was [1.5 x W/SF grams].

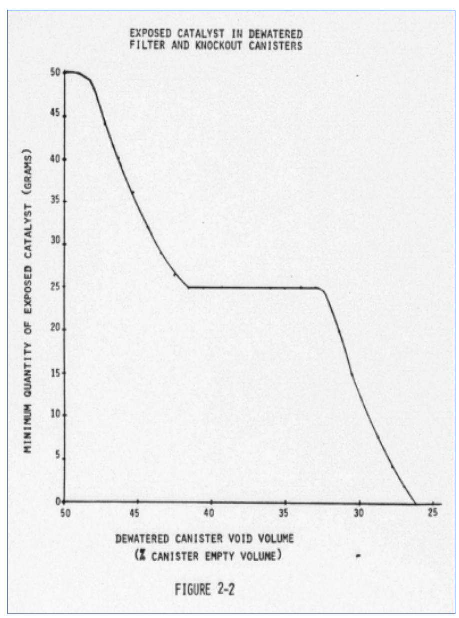
– *Exposed Catalyst Curves*. The SER provided two plots, Figure 2-1 for fuel canisters and Figure 2-2 for knockout and filter canisters. These figures showed the minimum quantity of exposed catalyst for any canister orientation as a function of dewatered canister void volume. Compliance with the canister dewatering acceptance criterion was achieved when the minimum quantity of exposed catalyst for the calculated dewatered canister void volume exceeded the sufficient quantity of catalyst defined above.

– *An Example*. As an example, previous laboratory testing of catalysts resulted in an SF of about 6. Assuming 100 watts of decay heat load in a canister, the sufficient quantity of exposed catalyst would be [(1.5) x (100)/(6) = 25 grams]. Figure 2-1 shows the acceptance criterion for fuel canisters with 100 watts of decay heat to be a void volume greater than 21 percent of the canister empty volume. Figure 2-2 shows the acceptance criterion for knockout or filter canisters with 100 watts of decay heat to be a void volume greater than 32 percent of the canister empty volume. Calculations showed that a canister payload of about 2150 pounds of TMI-2 core debris was equivalent to 100 watts of decay heat for the remainder of the fuel shipping program. Therefore, the sufficient quantity of catalyst could be reduced to reflect the canister decay heat load based on the actual canister payload weight.

– *Adjustments*. Further laboratory testing of the catalyst would be performed whenever new potential catalyst contaminants were considered. The results of these future tests would then be used to determine the appropriate dewatered canister void volume to ensure compliance with the dewatering acceptance criterion.



SER Figure 2-1. Minimum quantity of exposed catalyst for any fuel canister orientation as a function of dewatered canister void volume.



SER Figure 2-2. Minimum quantity of exposed catalyst for any knockout and filter canister orientation as a function of dewatered canister void volume.

- *Conclusion*. If compliance with the dewatering acceptance criterion could not be demonstrated using the above methods, then the canister would be evaluated on a case-by-case basis. If the evaluation did not conservatively show that the dewatering acceptance criterion was met, appropriate corrective action would be taken in accordance with approved site procedures. No canister would be shipped unless the dewatering acceptance criterion was met.

- ***NRC Review: Hydrogen***. [99, 100] The NRC's safety evaluation concluded that the canisters could be shipped in the Model 125-B shipping cask in compliance with the cask certificate of compliance. The procedures of the canister handling and preparation for shipment program involved the verification of gas control to determine that catalytic recombiners were functioning and the verification of dewatering to determine that the catalytic recombiners remained operable regardless of canister orientation.

- *Verification of Gas Control.* [101] The canisters were designed with catalytic recombiners to prevent the buildup of radiolytically generated combustible gases during shipment. The shipping cask certificate of compliance required that both hydrogen and oxygen concentrations not exceed 5 percent by volume during twice the expected shipping time. The certificate of compliance further required that the elapsed time between canister closure and purging and completion of shipment could be no more than the period of time when the canister gas concentration was below these specified limits.

  The licensee would determine the gas generation rate in each canister by one of two methods. Either a gas sample would be obtained from a dewatered canister, or the canister pressure would be measured and compared to the pressure recorded at the time of final dewatering. These checks would be performed after the canister was dewatered and allowed to remain in the storage pool for a period of time. The length of time necessary to reach the maximum allowable gas concentration would be calculated from the gas appearance rate. This time period would be used to determine a maximum onsite storage time to allow shipment within the time constraints specified in the certificate of compliance. The gas monitoring program would be implemented through appropriate procedures reviewed and approved by the NRC.

- *Verification of Dewatering.* [102] Canister dewatering was required to ensure that there was sufficient void volume in the loaded fuel canister for the accumulation of radiolytic gases without overpressurizing a canister. In addition, there must be sufficient void volume to ensure that at least half of the installed catalytic recombiners were not submerged in free water regardless of canister orientation. At all times, the volume within the canisters must be equal to or greater than one-half the free empty volume of the canister.

  The licensee proposed two methods to verify that this specification was met. The first involved quantitative measurements. The weight of a filled and flooded canister would be compared to the weight of the canister after dewatering. The difference would be the weight of water removed and could be used to calculate the remaining canister void volume. The NRC determined that this method, if implemented through appropriately controlled procedures, was acceptable to ensure that the canister void volume met the design specifications. The second method proposed by the licensee was a qualitative method that removed water from the canister by purging with an inert gas. The NRC determined that this method did not provide for an acceptable quantitative determination to verify conformance to the design specifications.

- *Acceptance Criterion*. (103) The acceptance criterion for canister dewatering was a dewatered canister void volume that ensured a sufficient quantity of exposed catalyst in any canister orientation. Revision 5 of the licensee's SER (104) changed this criterion to allow greater flexibility in determining the minimum quantity of recombiner catalyst that should be exposed in the canister gas space during shipping of a loaded canister. The revised criterion required a minimum of 25 grams of exposed catalyst in any canister orientation. This was based on conservative predictions of the maximum theoretical gas generation rates, the maximum catalyst poisoning (i.e., minimum catalyst recombination capacity), and a maximum theoretical decay heat loading of 100 watts per canister. This was discussed in detail in the licensee's previous (105, 106) evaluations. Revision 5 would allow a further adjustment in the quantity of exposed catalyst for canisters that were loaded with debris with a decay heat rate of less than 100 watts. The NRC determined that this was acceptable because gas generation was shown to be directly proportional to decay heat load, and recombination rate was directly proportional to the quantity of catalyst. In addition, the quantity of catalyst was 1.5 times the quantity needed to recombine the maximum theoretical amount of radiolytically produced gas assuming the worst possible contamination of the catalyst. The licensee's proposal would allow a further reduction in the amount of catalyst when laboratory testing could confirm and quantify less contamination. The revision would also provide for increasing the required amount of catalyst when a greater degree of catalyst contamination was found. The NRC concluded that this flexibility in the defueling program was acceptable and still provided a suitable level of conservatism. This conclusion was based on the actual gas appearance rates that had been seen to date to be nearly a factor of 10 below the theoretical maximum that was used in the calculation of the minimum catalyst quantity.

4.5.3.6 Canister Dewatering System

- ***Purpose***. To remove and filter the water from submerged defueling canisters and to provide a transfer path to the defueling water cleanup system for processing. The dewatering system also provided the cover gas for canister shipping.

- ***Evaluation: Hydrogen***. Editor's Note: The licensee's safety evaluations of the canister dewatering system were provided in the safety evaluation reports (107, 108) for canister handling and preparation for shipment (see above).

- ***NRC Review: Hydrogen***. Editor's Note: The NRC's safety evaluations of the canister dewatering system were provided in the safety evaluation reports [109, 110] for canister handling and preparation for shipment (see above).

4.5.3.7 Use of Nonborated Water in Canister Loading Decontamination System (NA)

4.5.4 Testing of Core Region Defueling Techniques (NA)

4.5.5 Fines/Debris Vacuum System (NA)

4.5.6 Hydraulic Shredder (NA)

4.5.7 Plasma Arc Torch

4.5.7.1 Use of Plasma Arc Torch to Cut Upper End Fittings (NA)

4.5.7.2 Use of Plasma Arc Torch to Cut the Lower Core Support Assembly (NA)

4.5.7.3 Use of Plasma Arc Torch to Cut Baffle Plates and Core Support Shield (NA)

4.5.7.4 Use of Air as Secondary Gas for Plasma Arc Torch

- ***Purpose***. To replace nitrogen gas with air as the secondary gas to improve plasma arc torch performance by achieving longer and more efficient cuts.

- ***Evaluation: Hydrogen***. [111] The licensee's safety evaluation concluded that no additional hydrogen gas would evolve because of this change. Air had been used as a secondary cutting gas commercially with no known adverse effects. The physical change essentially was to go from 100-percent nitrogen to a mixture that was 78-percent nitrogen, 21-percent oxygen, and 1-percent argon. The work platform off-gas system would be operated according to NRC requirements [112, 113] to exhaust the plasma gas effluents from the area above the reactor vessel water surface to the containment building purge system. Additionally, based on the same NRC requirements, the containment building purge system would be operated whenever cutting was in progress. During the first planned cuts using air as the secondary gas, samples would be taken from above the water surface to be analyzed for toxic substances. Further, the configuration of the system and administrative procedures should preclude the potential for inadvertent cross connection of the nitrogen and service air systems.

- ***NRC Review***. Editor's Note: The NRC's safety evaluation could not be located.

4.5.8 Use of Core Bore Machine for Dismantling Lower Core Support Assembly (NA)

4.5.9 Sediment Transfer and Processing Operations (NA)

4.5.10 Pressurizer Spray Line Defueling System (NA)

4.5.11 Decontamination Using Ultrahigh Pressure Water Flush (NA)

4.6 Evaluations for Defueling Operations

4.6.1 Preliminary Defueling (NA)

4.6.2 Early Defueling

- ***Purpose***. To remove end fittings, structural materials, and related loose debris that would not involve removal of significant amounts of fuel material, to remove intact segments of fuel rods and other pieces of core debris, and to remove loose fuel fines (particles) by vacuuming operations.

- ***Evaluation: Hydrogen***. [114] Editor's Note: The licensee's safety evaluation for early defueling was practically identical to its subsequent safety evaluation report [115, 116] for bulk defueling; therefore, the evaluation text is not included in this section. For details, please refer to the next section.

- ***NRC Review: Hydrogen***. [117] The NRC's safety evaluation concluded that acceptable methods of combustible gas control should exist during early defueling activities. A previous NRC safety evaluation [118] concluded that the design of defueling canisters was acceptable for the control of combustible gases. The evaluation also concluded that the consequences of an unlikely combustion event during a canister transfer, whether inside a canister or within the canister transfer system, would not pose an unacceptable risk.

  - *Off-Gas System*. During early defueling activities in the reactor vessel, including dewatering of canisters, any hydrogen or other gases generated or released would collect in the air space below the defueling work platform. The defueling off-gas system would be operated as needed to create an airflow through the work platform; into the internals indexing fixture enclosure, where the air would mix with collected gases; and out to the containment building after passing through a filtration unit. Operation of the off-gas system would dilute any accumulated hydrogen gas, preventing a combustible concentration from being reached.

  - *Overpressure Protection and Discharge*. Previous [119, 120] NRC safety evaluations addressed detailed provisions for controlling hydrogen gas buildup in loaded defueling canisters. Each type of canister was designed with catalytic recombiners to limit the concentration of hydrogen gas to prevent combustion or canister overpressurization. Dewatering of canisters was necessary to expose sufficient catalyst for recombination to be effective. Loaded defueling canisters would be provided with two relief valves, with setpoints at 25 pounds per square inch gauge (psig) pressure and 150 psig, in preparation for the canisters' removal and storage. If the canisters were not dewatered before transfer to the fuel handling building, the radiolytic decomposition of water would result in the generation of hydrogen and oxygen; however, the relief valves would open to prevent canister overpressurization. Except for a brief time during transfer, the canisters would be underwater; thus, the opening of the relief valves would release small quantities of combustible gas to the water in the spent fuel pool or the fuel transfer canal. These small quantities of gas would be diluted by the large volumes of air in the containment building and fuel handling building; therefore, this gas would not reach a combustible concentration.

4.6.3 Storage of Upper End Fittings in an Array of 55-Gallon Drums

- ***Purpose***. To use shielded 55-gallon drums to store end fittings and other structural material removed from the reactor vessel.

- ***Evaluation: Hydrogen***. [121, 122] The licensee's safety evaluation concluded that a small amount of hydrogen generation due to radiolysis would not present a safety concern. Assuming that each drum container would store about 20 to 30 kilograms of fuel, the maximum calculated hydrogen generation rate from this fuel would be less than 4 liters of hydrogen per year. Release of this small quantity to the containment building would not appear to raise a concern about explosion; however, to preclude pressurization of the container, the container covers would be vented.

- ***NRC Review***. [123, 124] Editor's Note: The NRC's safety evaluation report did not specifically address this evaluation topic.

4.6.4 Defueling (Also Known as "Bulk" Defueling)

- ***Purpose***. To develop defueling tools and to conduct activities necessary to remove the remaining fuel and structural debris located in the original core volume. An additional activity was to address vacuuming of debris from the lower head of the vessel, a primary activity associated with lower head defueling.

- ***Evaluation: Hydrogen***. [125, 126] The licensee's safety evaluation of hydrogen concerns included off-gas hazards to the shielded work platform and hydrogen generation in the defueling canisters, canister storage racks, and canister transfer shield.

    - *Shielded Work Platform Off-Gassing*. During defueling activities, the reactor vessel would be covered by the shielded work platform. An off-gas system was designed to provide an air inflow through the top of the work platform. This system diluted gases that were released or evolved during defueling activities before they were released into the containment building. Any hydrogen evolved during defueling would be diluted by the off-gas treatment system, as required; thus, hydrogen would not reach a combustible concentration in the containment building.

    - *Defueling Canisters*. While the canisters or debris containers were being transported in the canister transfer shield or were in storage in either the fuel transfer canal inside the containment building or spent fuel pool "A", radiolytic generation of hydrogen could occur. Subsequently, the hydrogen could be released via vent openings or the relief valves. Any hydrogen released would be to either the fuel handling building or containment building, depending on the transfer or storage location. The hydrogen would be diluted by the large surrounding atmospheres of these buildings. Consequently, the release of hydrogen from the canisters or debris containers would not result in a combustible concentration of hydrogen in either building.

- *Canisters in the Transfer Shield*. The combustion of hydrogen within the canister transfer shield was not expected to occur. The canisters or debris containers would be in the transfer shield for short periods of time (i.e., during transfer from the reactor vessel to the deep end of the refueling canal or during canister handling in the fuel handling building). However, the licensee recognized that a canister or debris container could be in a transfer shield for extended periods of time. Even if a hydrogen generation rate within a single canister was postulated such that the rate was sufficient to cause a canister relief device to discharge to the transfer shield, there were no ignition sources inherent in the design or operation of the transfer shield. Also, the top of the transfer shield was vented to the surrounding building volume through a vent area of about 15 square inches. Given the provisions of the shield design, the licensee concluded that a hydrogen combustion incident in the canister transfer shield was very unlikely. However, if hydrogen combustion did occur, the resultant loadings were expected to be within the structural capabilities of the canister and the transfer shield.

- ***NRC Review: Hydrogen***. [(127)] The NRC determined that the safety consideration of hydrogen control was adequately addressed in previous agency safety evaluations and that the NRC's earlier conclusions applied to the proposed activities. Editor's Note: Refer to the NRC's safety evaluation report [(128)] for early defueling.

4.6.5 Use of Core Bore Machine for Bulk Defueling

- ***Purpose***. To use the core stratification sample acquisition (core bore) tooling as a defueling tool so that other defueling tools could more effectively break up and remove the remaining core debris. The core bore tool used a solid-faced bit to perforate the hard crust region of the core, down to the lower grid support structure, at multiple locations. The defueling work platform orientation system was used to position the drill mechanism with restrictions.

- ***Evaluation: Hydrogen***. [(129)] Editor's Note: The licensee's safety evaluation report did not specifically address this topic.

- ***NRC Review: Hydrogen***. [(130)] Editor's Note: The NRC review of the licensee's proposal was documented in the agency's safety evaluation report [(131)] for bulk defueling.

4.6.6 Lower Core Support Assembly Defueling

- ***Purpose***. To dismantle and defuel the lower core support assembly and to partially defuel the lower reactor vessel head. Structural material removed from the lower core support assembly included the: (●) lower grid rib assembly; (●) lower grid forging; (●) distribution plate; and (●) in-core guide support plate. The lowest elliptical flow distributor and the gusseted in-core guide tubes would not be removed under this proposed activity. The use of the core bore machine, plasma arc cutting equipment, cavitating water jet, robot manipulators, automatic cutting equipment system, and other previously approved tools and equipment was included in this activity and associated safety evaluations.

- ***Evaluation: Hydrogen***. (132) Generation of small quantities of hydrogen gas generation (less than 0.1 standard cubic foot per minute) would be a byproduct of the plasma arc cutting tool operation underwater. This hydrogen would be diluted by the off-gas treatment system, as required. Therefore, a combustible concentration would not occur within the containment building. Other hydrogen-related safety issues were bounded by the evaluations in the licensee's report (133) on limits of foreign materials allowed in the reactor coolant system during defueling activities.

- ***NRC Review***. (134) Editor's Note: The NRC's safety evaluation report did not specifically address this evaluation topic.

4.6.7 Completion of Lower Core Support Assembly and Lower Head Defueling

- ***Purpose***. To remove the elliptical flow distributor and gusseted in-core guide tubes of the lower core support assembly and to defuel the reactor vessel lower head.

- ***Evaluation: Hydrogen***. (135, 136) Generation of small quantities of hydrogen gas generation (less than 0.1 standard cubic foot per minute) would be a byproduct of the plasma arc cutting tool operation underwater. This hydrogen would be diluted by the off-gas treatment system, as required. Therefore, a combustible concentration would not occur within the containment building. Other hydrogen-related safety issues were bounded by the evaluations in the licensee's report (137) on limits of foreign materials allowed in the reactor coolant system during defueling activities.

- ***NRC Review***. (138) Editor's Note: The NRC's safety evaluation report did not specifically address this evaluation topic.

4.6.8 Upper Core Support Assembly Defueling

- ***Purpose***. To cut and move the baffle plates and to defuel the upper core support assembly. This evaluation addressed the following activities: (●) cutting the baffle plates for later removal;

(●) removing bolts from the baffle plates; (●) removing the baffle plates; and (●) removing core debris from the baffle plates and core former plates.

- ***Evaluation: Hydrogen***. [139] Generation of small quantities of hydrogen gas generation (less than 0.1 standard cubic foot per minute) would be a byproduct of the plasma arc cutting tool operation underwater. This hydrogen would be diluted by the off-gas treatment system, as required. Therefore, a combustible concentration would not occur within the containment building. The building purge system was utilized to remove potentially toxic by-product gases produced buring plasma arc torch operation. Other hydrogen-related safety issues were bounded by the evaluations in the licensee's report [140] on limits of foreign materials allowed in the reactor coolant system during defueling activities.

- ***NRC Review***. [141] Editor's Note: The NRC's safety evaluation report did not specifically address this evaluation topic.

4.7 Evaluations for Waste Management

4.7.1 EPICOR II

- ***Purpose***. To decontaminate accident-generated, intermediate-level radioactive wastewater being held in tanks in the auxiliary building. Later, the system was used to polish effluents from the submerged demineralizer system during the cleanup of highly radioactive water from the containment building sump, reactor coolant system, and reactor coolant drain tanks. Following the decommissioning of the submerged demineralizer system, EPICOR II was used to clean residual wastewater from decontaminating the structures and systems.

- ***Background: Hydrogen***. An unexpected problem with the storage of spent EPICOR II liners that became a hazard and problem at TMI and the U.S. nuclear industry was the discovery of gas generation inside heavily loaded (radioactive) demineralizer liners. These liners used resin-based media for ion exchange. During shipping preparations of a heavily loaded prefilter liner to the DOE's Battelle Columbus Laboratories for characterization, a flash ignition of hydrogen occurred at the EPICOR II cask loading station in the TMI-2 waste packaging and handling facility. Hydrogen and oxygen gas generated by radiolytic decomposition of residual water in the liners became a safety concern during handling, transportation, and reception of the liners. [142]

The INEL designed and built a device to sample and vent the liners remotely and add recombiner catalyst. This remotely operated vent tool removed the vessel vent plug while maintaining a sealed environment around the storage module cell. The liners were purged of hydrogen gas, and the hydrogen gas generation rate was quantified before shipment to comply with U.S. Department of Transportation regulations. [143]

The DOE had studied the generation of hydrogen gas as a result of radiolysis of water before the TMI-2 accident; however, these studies were limited to high-level and transuranic wastes.

The NRC sponsored later studies of hydrogen generation in EPICOR II vessels, which resulted in changes to the certification of shipping casks. In Information Notice 84-72, [144] the NRC required plants to demonstrate, by tests or measurements, that combustible mixtures of gases were not present in radioactive waste shipments; otherwise, the waste was to be vented within 10 days of shipping. A task force, formed by the Edison Electric Institute to evaluate these NRC requirements, developed a calculational method to quantify hydrogen gas generation in sealed containers. The Electric Power Research Institute then demonstrated this calculational method using a desktop computer at TMI-2, and the NRC accepted this method. [145]

- ***Evaluation: Hydrogen***. Editor's Note: The licensee's safety evaluation was not located.

- ***NRC Review: Hydrogen***. [146, 147] Editor's Note: Hydrogen generation was not a known concern at TMI during the time that the NRC's environmental assessments were developed.

4.7.2 Submerged Demineralizer System

4.7.2.1 Submerged Demineralizer System Operations (NA)

4.7.2.2 Submerged Demineralizer System Liner Recombiner and Vacuum Outgassing System

- ***Purpose***. To eliminate the potential of a combustible hydrogen and oxygen mixture existing in the submerged demineralizer system (SDS) liners and to facilitate the ultimate shipment and burial of the SDS liners. The liner recombiner and vacuum outgassing system (LRVOS) was designed to remove moisture by evaporation from the zeolite beds of SDS spent liners. This operation dried the beds but did not remove the water in the zeolite.

- ***Evaluation: Hydrogen***. [148, 149] The licensee's safety evaluation and supplemental information provided the following safety basis for the proposed LRVOS relating to hydrogen hazards.

  - *Background*. The hydrogen/oxygen recombiner catalyst that was used in the SDS spent liners was made of a platinum/palladium-coated alumina formed into tiny cylindrical shapes. The recombiner catalyst was added to the liner after the liner was vacuum dried to a predetermined value and repressurized with an inert gas. A special tool was developed to allow the recombiner catalyst to be added remotely with the liner pressurized.

  - *Testing*. The LRVOS was satisfactorily tested by the DOE's Hanford Operations using an SDS liner loaded with zeolite and simulating the gas generation condition found in a radioactivity-loaded SDS liner. The purpose of the test study was to evaluate the ability to prepare an SDS liner for addition of the recombiner catalyst and to confirm the effectiveness of the catalyst (platinum/palladium) to recombine hydrogen and oxygen. The catalyst would minimize the pressure rise due to radiolytic generation of these gases from the water in the liner.

  - *Water Content*. To ensure adequate catalyst performance during the postulated accident conditions, the water content in the SDS liner would be reduced to less than 120 pounds.

This water content was based on feasibility testing performed by the DOE's Hanford Operations under simulated accident conditions.

- *Recombiner Catalyst*. The amount of catalyst to be added to each liner would be 236 grams, based on testing performed by the DOE's Hanford Operations. This amount of catalyst was based on the allowance for 1 percent by weight holdup of the catalyst in the catalyst insertion tool and the catalyst bed volume available. This volume of catalyst was significantly greater than the minimum amount needed to ensure recombination of the hydrogen and oxygen gases being generated by the decomposition of water in the SDS liners.

- *Hydrogen Detonation*. The safety evaluation concluded that the probability of occurrence or consequences of an accident would not increase. Previous calculations on spent SDS liners showed that the integrity of the liner would not be breached if the hydrogen and oxygen gas pressure was held below 10 pounds per square inch gauge (psig) pressure, should a detonation occur. This was based on the liner hydro pressure of 530 psig and the peak pressure of a hydrogen detonation being a factor of 20 above the initial liner absolute pressure. The actual number was 26.5 pounds per square inch absolute pressure (or 11.8 psig). When the catalyst was added, the liner pressure would be monitored after isolating the liner. If pressure were to rise to 5 pounds per square inch absolute, a sample would be obtained to determine the cause of the pressure increase.

- *Additional Testing*. An additional test would be performed on an SDS liner that was used for the processing of highly radioactive containment building sump water. A subsequent safety evaluation would be conducted to demonstrate that SDS liners could be safely shipped.

- *Contingency*. In the unlikely event that the catalyst did not work as anticipated, an increase of hydrogen and oxygen with a combustible mixture could be postulated. This situation would be precluded by the following measures: (●) SDS spent liners with catalyst added would be monitored before they were shipped to detect a failure. (●) A gas sample would be taken to analyze the cause of the pressure increase. (●) The liner would then be vented to the SDS off-gas header to preclude any further gas buildup or dangerous situation.

- ***NRC Review: Hydrogen***. [150] The NRC's safety evaluation considered the potential for gas ignition and breach of the SDS liner integrity, among other issues. The NRC concluded that no combustible gas mixtures would be created inside the SDS liner. This conclusion was based on reducing hydrogen and oxygen concentrations and limiting liner pressure during storage.

- *Gas Concentration*. The radiolytic gas generation rate of hydrogen and oxygen within the SDS liners was a function of both the curie loadings and the residual water content. To ensure that the gas accumulation and the potential energy source, if ignited, would not cause an SDS liner to lose integrity, procedures would be in effect to limit the internal pressure of spent SDS liners to less than 10 psig. This would be done by continuously venting the spent SDS liner via the vent header to the off-gas separator tank. The operation of the LRVOS when connected to the SDS liners would be predominantly at low pressure with the SDS liner and LRVOS pressures ranging from 30 inches of mercury (vacuum) to 5 psig. This pressure range corresponded to the normal operating configurations of LRVOS. The operations included: (●) vacuum drying to remove the free water in the SDS liner; (●) insertion of the catalytic recombiner pellets; (●) gas inerting of the liner with nitrogen or argon; and (●) monitoring and sampling the SDS liner to ensure that sufficient catalyst exists to recombine the hydrogen and oxygen.

- *Pressure Buildup*. The NRC reviewed all existing SDS gas generation data and pressure buildup rates in conjunction with the LRVOS design criteria and operational configuration. The evaluation concluded that if ignition occurred, system pressures would be controlled such that the SDS liner and LRVOS would be maintained. Before connecting the LRVOS with a spent liner, the liner would be dewatered and purged with nitrogen, thereby ensuring the hydrogen and oxygen gas inventories would be minimal (less than 0.2 percent). The vacuum drying operation of the LRVOS would continue to purge gases and water vapor from the liner, which should preclude the collection and accumulation of combustible gas mixtures. Procedural controls would also be established to require the nitrogen purging of the SDS liners if the liner was isolated from the SDS vent header or LRVOS for more than 5 hours. This requirement, which was based on twice the maximum expected gas generation rate, would also ensure that no combustible gas mixtures would be created.

- *Conclusion*. The NRC concluded that the SDS LRVOS could be operated safely with the controls and acceptance criteria provided in the licensee's safety evaluation. There was reasonable assurance that the LRVOS would safely perform its intended function on spent SDS liners. The evaluation concluded that the risk to the health and safety of the public and the occupational workforce was minimal. The use of the LRVOS would not change the boundary of accidents previously considered in the SDS safety evaluation report. Additionally, the environmental effects from the operation of the LRVOS fell within the scope of conditions previously considered in the PEIS.

4.7.2.3 Shipment of Spent Submerged Demineralizer System Liners Containing Special Gas Passification Systems

Editor's Note: Although shipments of fuel debris and other radioactive wastes were outside the scope of this NUREG/KM, the safety evaluation of hydrogen concerns during preparation for shipment of submerged demineralizer system (SDS) liners is included below, given its uniqueness to TMI-2.

- ***Purpose***. To ensure that combustible gas mixtures were not generated and vacuum conditions were maintained during SDS liner handling and shipping periods. The special passification systems included the use of a catalyst recombiner, a vacuum drying system for dewatering the liner, and the installation of a special relief valve device on each SDS liner before shipment. The vacuum-dried liner condition ensured that sufficient water removal occurred and optimized conditions for the catalyst to recombine the hydrogen and oxygen gases produced by the radiolysis of the residual water in the liner. The special relief device provided overpressure protection for the spent SDS liner during extended offsite storage periods at the DOE's Hanford Operations facility.

- ***Evaluation: Hydrogen***. [151] In support of the licensee's intent to begin shipping spent SDS liners, the licensee's safety evaluation presented the process for using and monitoring passification systems in preparation for shipments.

  - *Catalyst Tests*. To ensure that combustible mixtures of gas were not present in the liner during shipment, a hydrogen/oxygen catalyst was added to the vent screen of the liner via a 1.5-inch quick-connect coupling connection on the vent nozzle. The DOE's Hanford Operations testing demonstrated the effectiveness of the catalyst at recombining hydrogen and oxygen gases under various liner orientations and internal pressures ranging from the vapor pressure of water to 1 atmosphere. The tests included an additional rate of hydrogen and oxygen gas addition of 3 liters per hour. This rate was more than 2 times the measured rate of 1.25 liters per hour that was generated by the highest curie loaded SDS liner without a catalyst. In all tests with a catalyst, the liner internal pressure stabilized, and no detectable hydrogen was found when internal liner gases were sampled.

  - *Catalyst Insertion*. Once the catalyst was added to the liner, the liner would be pumped down to approximately the vapor pressure of water and then isolated on a pressure/vacuum gauge to verify proper catalyst operation. The gas generation for the test SDS liner would be monitored for 14 days before shipment. Subsequent liners would be monitored for a sufficient period to ensure proper catalyst operation. The pressure of the liner when loaded into the shipping cask would be approximately the vapor pressure of water. Any liners that were determined to leak would be handled on a case-by-case basis for shipping. Pressures during the isolation period would be monitored, and the pressure increase rate would be calculated to determine the shipping window. To detect any water leakage into the liner, the liner would be weighed periodically during the isolation period.

  - *Cask Preparation*. At the completion of the isolation period or upon reaching an upper limit of 10 pounds per square inch absolute (psia) pressure on the liner pressure rise, the liner gas inventory would be sampled to determine concentrations of combustible gases. This modified the previous 5-psia limit and was based on experience gained during the testing of the SDS liner. These results, along with the calculated pressure increase rate data, would be used to verify compliance with the shipping cask certificate of compliance. The shipping cask void space would be inerted with nitrogen before each shipment to below 5-percent oxygen volume at standard temperature and pressure. This action would ensure that oxygen did not enter the liner in the unlikely event a leak should develop between the liner and shipping cask cavity during shipment.

- ***NRC Review: Hydrogen***. [152] The NRC's safety evaluation reviewed the three criteria for shipping the recombiner-loaded spent SDS liners: (●) residual water content of the liner was not to exceed 120 pounds during handling and shipment; (●) hydrogen gas concentration was limited to no more than 5 percent by volume at standard temperature and pressure (i.e., no more than 0.063 gram-mole per cubic foot at 14.7 psia and 70 degrees Fahrenheit); and (●) if the hydrogen criteria could not be demonstrated (due to nonstoichiometric production), the oxygen gas concentration would be limited to no more than 5 percent by volume.

- *Residual Water*. The NRC's safety evaluation reviewed two independent methods (i.e., measured and calculated) for verifying that the residual water content criteria were met and approved the methodology. The measured method, which incorporated pre- and post-weighing of the vacuum-dried liner, provided an accurate measurement (± 5 pounds) of the water removed. The more conservative water weight of 220 pounds (as measured by the DOE's Hanford Operations recombiner tests) for a bulk dewatered (nitrogen-purged) liner also provided additional conservatism in residual water content measurement. The calculated method that would be used as a backup technique was also adequate. The NRC required both methods for verification of each SDS liner. Based on the residual water criteria of 120 pounds and conservatisms in the recombiner tests for both normal operation and potential accidents, the NRC concluded that these weight verification methods provided adequate measurements and safety margins for shipments.

- *Hydrogen Concentration*. The gas composition criteria were consistent with the requirements [153] of the CNS-1-13C II shipping cask certificate of compliance. Each spent SDS liner would be monitored and sampled to demonstrate compliance with these requirements. The NRC reviewed procedures and hardware for liner sampling and gas analyses and approved the proposed techniques and analytical capability. In addition, each spent SDS shipment would include the nitrogen inerting of the shipping cask void space, which would provide additional safety assurance of noncombustible conditions within the waste package. Since each SDS liner would be prepared for shipment in accordance with NRC-approved procedures and also would meet the requirements of the Type B shipping cask certificate of compliance (designed to withstand transport accident), the NRC concluded that controls and safety features were adequate to ensure compliance with hydrogen concentration limits.

- *Shipment Tests and Analysis*. In accordance with the NRC's previous approval of SDS liner shipments, [154] each shipment would be subject to individual analysis and tests to demonstrate compliance with all applicable NRC and U.S. Department of Transportation regulations. These tests and analyses would include: (●) verification of curie loadings; (●) verification of residual water content; (●) gas analysis; (●) dose rate survey; and (●) related information. The NRC would continue to review the above information and inspect each shipping container individually before actual shipment.

- *Conclusion*. The NRC approved the licensee's proposed plan, subject to the submittal of individual information packages with supporting data, for shipping of recombiner-loaded SDS liners. Additionally, the NRC recognized that the DOE would take possession of the SDS liners at the TMI site boundary and would be the designated shipper of record. The NRC's TMI project office staff would continue to complete radiological survey verification and provide this information to the DOE.

4.8 Endnotes

Reference citations are exact filenames of documents on the Digital Versatile Discs (DVDs) to NUREG/KM-0001, Supplement 1, (155) "Three Mile Island Accident of 1979 Knowledge Management Digest: Recovery and Cleanup," issued June 2016, DVD document filenames start with a full date (YYY-MM-DD) or end with a partial date (YYYY-DD). Citations for references that are not on a DVD begin with the author organization or name(s), with the document title in quotation marks. Contents of the DVDs are available at the Idaho National Laboratory Web site (https://tmi2kml.inl.gov).

---

[1] GEND-INF-023-VOL-4, ANALYSIS OF THE THREE MILE ISLAND UNIT 2 HYDROGEN BURN (1983-03)

[2] GEND-052, Hydrogen Control in Handling, Shipping, and Storage of Wet Radioactive Waste (1986-02)

[3] GEND-052, Hydrogen Control in Handling, Shipping, and Storage of Wet Radioactive Waste (1986-02)

[4] NUREG-0591, Environmental Assessment, Use of EPICOR-II at TMI-2 (1979-08)

[5] (1979-10-19) NRC Approve EPICOR II Procedures and Design and Construction Details

[6] (1979-10-22) NRC Order (10-18-1979), Operation of EPICOR II to decontaminate Intermediate Water in Aux Bldg

[7] Brookhaven National Laboratory (BNL), "Radiation Effects on Ion Exchange Materials," BNL-50781, November 1977 [Available at osti.gov]

[8] (1980-05-05) BNL Report, Leachability, Structural Integrity, and Radiation Stability of Resins Solidified in Cement

[9] BNL, "Radiation Effects on Ion Exchange Materials," BNL-50781, November 1977 [Available at osti.gov]

[10] (1980-05-15) NRC, Evaluation of EPICOR II Wastes Under Handling, Storage, Trans. and Disposal Conditions

[11] (1980-07-02) GPU Response to NRC (05-15-1980), Evaluation of EPICOR II Wastes

[12] (1980-10-23) EGG Report, EPICOR, Summary of Studies on Stability of Ion Exchange Resins in Radiation Environs

[13] (1980-12-04) GPU, EPICOR II Liner Evaluation Status

[14] GEND-015, CHARACTERIZATION OF EPICOR II PREFILTER LINER 16 (1982-08)

[15] GEND-INF-025, Development of a Prototype Gas Sampler for EPICOR II Prefilter Liners (1982-09)

[16] (1980-04-10) GPU Technical Evaluation, Submerged Demineralizer System

[17] (1981-03-11) GPU Technical Evaluation, SDS, Revised

[18] (1980-09-15) NRC, Advises Detailed Safety Evaluation for SDS Necessary

[19] NUREG-0683, Vol. 1, PEIS-Decontamination and Disposal of Radioactive Wastes Resulting from TMI-2 (1981-03)

[20] (1981-04-24) GPU Response to NRC (Discussions), Potential for Explosive Mixtures in SDS Vessels

[21] (1981-05-06) GPU, Clarification of 04-24-1981 Submittal, Radiolysis Data for Irradiated Zeolite Systems

[22] DOE-NE-0012, Evaluation of Increased Cesium Loading on SDS Zeolite Beds (1981-05)

[23] NUREG-0557, Evaluation of Long-Term Post-Accident Core Cooling of TMI-2 (1979-05)

[24] NUREG-0683, Vol. 1, PEIS-Decontamination and Disposal of Radioactive Wastes Resulting from TMI-2 (1981-03)

[25] (1981-08-04) GPU, Status of EPICOR II Prefilter Liners
[26] GEND-031, Submerged Demineralizer System Processing of TMI-2 Waste Water (1983-02)
[27] DOE-NE-0012, Evaluation of Increased Cesium Loading on SDS Zeolite Beds (1981-05)
[28] GEND-029, Preparations to Ship EPICOR Liners (1983-06)
[29] GEND-015, CHARACTERIZATION OF EPICOR II PREFILTER LINER 16 (1982-08)
[30] GEND-027, CHARACTERIZATION OF EPICOR II PREFILTER LINER 3 (1983-04)
[31] GEND-INF-015, PRELIMINARY CHARACTERIZATION OF EPICOR II PREFILTER 16 LINER (1981-11)
[32] GEND-INF-025, Development of a Prototype Gas Sampler for EPICOR II Prefilter Liners (1982-09)
[33] GEND-053, EPICOR-II Resin Degradation Results from First Resin Samples of PF8 and PF20 (1985-12)
[34] NUREGCR-4150, TMI-2 EPICOR-II Resin Degradation Results from First Resin Samples of PF-8 and PF-20 (1985-07)
[35] NUREGCR-4608, EPICOR-II Resin Degradation Results from Second Samples of PF-8 and PF-20 (1986-10)
[36] NUREGCR-5594, Radiation degradation in EPICOR-2 ion exchange resins (1990-09)
[37] GEND-035, Submerged Demineralizer System Vessel Shipment Report (1984-06)
[38] GEND-048, COMMERCIAL DISPOSAL OF HIGH INTEGRITY CONTAINERS WITH EPICOR-II PREFILTERS (1985-09)
[39] EGG-TMI-6304, Design Analysis, High-Integrity Container for Disposal of EPICOR-II Prefilter Liners (1983-06)
[40] GEND-051, Evaluation of Special Safety Issues Associated with Handling the TMI-2 Core Debris (1985-06)
[41] GEND-052, Hydrogen Control in Handling, Shipping, and Storage of Wet Radioactive Waste (1986-02)
[42] USNRC, "Clarification of Conditions for Waste Shipments Subject to Hydrogen Gas Generation," Information Notice No. 84-72, September 10, 1984 [Available at nrc.gov]
[43] GEND-041, Predict Combustible Gas Generation in Sealed Radioactive Waste Containers (1986-05)
[44] GEND-062, Catalyst Tests for Hydrogen Control in Canisters of Wet Rad Waste (1987-08)
[45] (1982-06-15) GPU Safety Evaluation, Axial Power Shaping Rod Insertion Test
[46] (1982-05-17) Order Amendment
[47] (1982-07-06) GPU Safety Evaluation, Insertion Camera Through Reactor Vessel Leadscrew Opening, Rev. 2
[48] (1982-07-13) NRC Review, Control Rod Drive Mechanism Quick Look Camera Inspection (re 07-06-1982) (2)
[49] (1983-05-19) GPU Safety Analysis, Radiation Characterization Under Reactor Vessel Head (No Polar Crane), Rev. 0
[50] (1983-05-25) GPU Safety Analysis, Radiation Characterization Under the Reactor Vessel Head, Rev. 0
[51] (1983-07-13) NRC Review, Reactor Vessel Underhead Characterization (re Various Letters)
[52] (1983-07-20) GPU Safety Evaluation, Underhead Characterization, Core Sampling Addendum
[53] USNRC, "Three Mile Island Accident of 1979 Knowledge Management Digest: Recovery and Cleanup," NUREG/KM-0001, Supplement 1, June 2016 [Available at nrc.gov]
[54] (1983-07-20) GPU Safety Evaluation, Underhead Characterization, Core Sampling Addendum

[55] (1983-08-19) NRC Safety Evaluation, Addendum to the Underhead Characterization Study (re 07-30-1983)
[56] (1989-08-18) GPU Safety Evaluation, Remove Metallurgical Samples from Reactor Vessel SER
[57] (1989-10-20) GPU Safety Evaluation, Remove Metallurgical Samples from Vessel, Rev. 1 (re 08-18-1989) (effective pages)
[58] (1986-05-15) GPU Safety Evaluation, Defueling SER, Rev. 10
[59] (1989-11-28) NRC Safety Evaluation, Reactor Vessel Lower Head Metallurgical Sampling (re 08-18-1989)
[60] (1982-11-22) GPU, Inform Plans to Refill RCS
[61] (1982-12-14) GPU, Changes in Plan to Vent Pressurizer During RCS Refill (re 11-22-1982)
[62] (1982-12-14) NRC Review, Plan to Refill and Pressurize RCS to Purge Gases (re 11-22-1982
[63] (1984-03-09) GPU Safety Evaluation, Head Removal, Rev. 5
[64] GPU Nuclear Corporation, "RCS Hydrogen Evolution," Letter 4410-82-L-G042 to NRC, November 8, 1982 [ADAMS Accession No. ML20028A024]
[65] (1984-07-17) NRC Safety Evaluation, Reactor Pressure Vessel Head Lift (re -3-09-1984)
[66] (1983-02-25) GPU Safety Evaluation, Makeup and Purification Demineralizer Resin Sampling
[67] (1984-07-19) GPU Safety Evaluation, Makeup and Purification Cesium Elution SER
[68] (1984-09-19) NRC Safety Evaluation, Makeup and Purification Demineralizer Cesium Elution (re 07-19, 08-29-1984)
[69] (1989-05-12) GPU Technical Evaluation, Defueling Water Cleanup System, Rev. 12
[70] (1985-08-06) NRC Safety Evaluation, Defueling Water Cleanup System (re various letters)
[71] (1986-06-30) GPU Safety Evaluation, Operation of Filter-Aid Feed System
[72] (1986-07-09) NRC Review, Operation of Filter-Aid Feed System, Use of Diatomaceous Earth (re various letter)
[73] (1986-12-15) GPU Safety Evaluation, Addition of Coagulants to the Reactor Coolant System, Rev. 0
[74] (1986-12-31) GPU, Use of Different Coagulants (re 12-15-1986)
[75] (1986-12-15) GPU Safety Evaluation, Addition of Coagulants to the Reactor Coolant System, Rev. 0
[76] (1986-12-31) GPU, Use of Different Coagulants (re 12-15-1986)
[77] (1986-06-30) GPU Safety Evaluation, Operations of Filter-Aid Feed System and Use of Diatomaceous Earth as Feed Material
[78] (1987-03-27) NRC Review, Criticality Safety Evaluation for Coagulants (re 12-11-1986, 02-20, 01-05-1987)
[79] (1986-05-05) GPU Safety Evaluation, Addition of a Biocide to the TMI-2 Reactor Coolant System, Rev. 0
[80] (1986-05-06) NRC Review, Addition of a Biocide to the TMI-2 Reactor Coolant System (re 05-05-1986)
[81] (1985-09-10) GPU Technical Evaluation, Defueling Canister TER (re 08-15-1985)
[82] GEND-051, Evaluation of Special Safety Issues Associated with Handling the TMI-2 Core Debris (1985-06)
[83] Turner, S.E., "Radiolytic Decomposition of Water in Water-Moderated Reactors Under Accident Conditions," *Reactor and Fuel Processing Technology*, Vol. 12, No. 1, 1968
[84] (1985-11-05) NRC Safety Evaluation, Defueling Canisters (re various letters)
[85] (1985-09-10) GPU Technical Evaluation, Defueling Canister TER (re 08-15-1985)
[86] GEND-051, Evaluation of Special Safety Issues Associated with Handling the TMI-2 Core Debris (1985-06)
[87] (1986-04-11) GPU Safety Evaluation, Replacement of Loaded Fuel Canister Head Gaskets

[88] (1986-04-29) NRC Review, Replacement of Loaded Fuel Canister Head Gaskets (re 04-11-1986)
[89] (1986-01-17) GPU, Plan to Use Modified Debris Canisters for removing End Fittings, etc
[90] (1986-01-21) NRC Review, Plan to Use Modified Fuel Canisters to Remove End Fittings, etc (re 01-17-1986)
[91] (1987-04-02) GPU, Safety Evaluation, Canister Handling and Preparation for Shipment SER, Rev. 4 (Effective Pages Only)
[92] (1988-03-29) GPU Safety Evaluation, Canister Handling and Preparation for Shipment, Rev. 6, (Effective Pages Only)
[93] (1987-01-07) NRC Review, Reduced Defueling Canister Void Volume (re 11-10, 12-30-1986)
[94] (1986-11-10) GPU, Alternative Dewatering Acceptance Criterion for Defueling Canisters
[95] (1986-12-30) GPU, Additional Info on Alternative Dewatering Acceptance Criteria for Defueling Canisters
[96] Pacific Nuclear Systems, Inc., "Safety Analysis Report for the NuPac 125-B Fuel Shipping Cask," Rev. 0, June 1985 [ADAMS Accession No. ML20132E700]
[97] GEND-051, Evaluation of Special Safety Issues Associated with Handling the TMI-2 Core Debris (1985-06)
[98] (1986-03-06) GPU Technical Evaluation, Defueling Canister TER, Rev. 2
[99] (1986-06-20) NRC Safety Evaluation, Canister Handling and Preparation for Shipment Program (re 02-17, 06-11-1986)
[100] (1988-03-03) NRC Review, Canister Handling and Preparation for Shipment Program TER (re various letters)
[101] (1986-06-20) NRC Safety Evaluation, Canister Handling and Preparation for Shipment Program (re 02-17, 06-11-1986)
[102] (1986-06-20) NRC Safety Evaluation, Canister Handling and Preparation for Shipment Program (re 02-17, 06-11-1986)
[103] (1988-03-03) NRC Review, Canister Handling and Preparation for Shipment Program TER (re various letters)
[104] (1987-05-26) GPU Safety Evaluation, Canister Handling and Preparation for Shipment, Rev. 5 (Effective Pages Only)
[105] (1986-12-30) GPU, Additional Info on Alternative Dewatering Acceptance Criteria for Defueling Canisters
[106] (1987-10-21) GPU, Plan to Discontinue Defueling Canister Gas Sampling at INL (re 01-07-1987)
[107] (1987-04-02) GPU, Safety Evaluation, Canister Handling and Preparation for Shipment SER, Rev. 4 (Effective Pages Only)
[108] (1987-05-26) GPU Safety Evaluation, Canister Handling and Preparation for Shipment, Rev. 5 (Effective Pages Only)
[109] (1986-06-20) NRC Safety Evaluation, Canister Handling and Preparation for Shipment Program (re 02-17, 06-
[110] (1988-03-03) NRC Review, Canister Handling and Preparation for Shipment Program TER (re various letters)
[111] (1989-02-08) GPU Safety Evaluation, Use of Air as Secondary Gas for Plasma Arc Torch
[112] (1988-04-01) NRC Safety Evaluation, Lower Core Support Assembly Defueling (re 01-18-1988)
[113] (1987-08-20) NRC Review, Use of Plasma Arc Torch to Cut Upper End Fittings (re various letters)
[114] (1985-10-10) GPU Safety Evaluation, Early Defueling of the TMI-2 Reactor Vessel, Rev. 4
[115] (1986-05-15) GPU Safety Evaluation, Defueling SER, Rev. 10

[116] (1986-05-20) GPU Safety Evaluation, Defueling SER, Rev. 10 (Correction)
[117] (1985-11-12) NRC Safety Evaluation, Early Defueling of the TMI-2 Reactor Vessel (10-10-1985)
[118] (1985-11-05) NRC Safety Evaluation, Defueling Canisters (re various letters)
[119] (1985-09-10) GPU Technical Evaluation, Defueling Canister TER (re 08-15-1985)
[120] (1985-11-05) NRC Safety Evaluation, Defueling Canisters (re various letters)
[121] (1986-08-16) GPU, Storage of Upper End Fittings in Shielded 55 Gal Drums
[122] (1986-09-09) GPU Response to NRC (Discussion), Storage of Upper End Fittings (re 08-16-1986)
[123] (1986-09-10) NRC Review, Plans to Store Upper End Fittings (re various letters)
[124] (1986-09-25) NRC Safety Evaluation, Storage of Upper End Fittings in an Array of 55 Gallon Drums (re various letters)
[125] (1986-05-15) GPU Safety Evaluation, Defueling SER, Rev. 10
[126] (1986-05-20) GPU Safety Evaluation, Defueling SER, Rev. 10 (Correction)
[127] (1986-07-24) NRC Review, Core Region Bulk Defueling Operations Limited to the Core Region (re 05-15, 07-23-1986)
[128] (1985-11-12) NRC Safety Evaluation, Early Defueling of the TMI-2 Reactor Vessel (10-10-1985)
[129] (1986-07-23) GPU, Plan to Use Core Bore Tooling as Defueling Tool
[130] (1986-07-24) NRC Review, Core Region Bulk Defueling Operations Limited to the Core Region (re 05-15, 07-23-1986)
[131] (1986-07-24) NRC Review, Core Region Bulk Defueling Operations Limited to the Core Region (re 05-15, 07-23-1986)
[132] (1988-01-18) GPU Safety Evaluation, Lower Core Support Assembly Defueling, Rev. 2
[133] GPU Nuclear Corporation, “Report on Limits of Foreign Materials Allowed in the TMI-2 Reactor Coolant System During Defueling Activities,” Rev. 1, 15737-2-N09-002, October 1985 [ADAMS Accession No. ML19302G716]
[134] (1988-04-01) NRC Safety Evaluation, Lower Core Support Assembly Defueling (re 01-18-1988)
[135] (1988-06-06) GPU Safety Evaluation, Completion of Lower Core Support and Lower Head Defueling, Rev. 0
[136] (1988-06-27) GPU Safety Evaluation, Completion of Lower Core Support and Lower Head Defueling, Rev. 1
[137] GPU Nuclear Corporation, “Report on Limits of Foreign Materials Allowed in the TMI-2 Reactor Coolant System During Defueling Activities,” Rev. 1, 15737-2-N09-002, October 1985 [ADAMS Accession No. ML19302G716]
[138] (1988-12-01) NRC Safety Evaluation, Lower Core Support and Lower Head Defueling (re 06-06-1988)
[139] (1988-09-15) GPU Safety Evaluation, Completion of Upper Support Core Assembly Defueling, Rev. 0
[140] GPU Nuclear Corporation, “Report on Limits of Foreign Materials Allowed in the TMI-2 Reactor Coolant System During Defueling Activities,” Rev. 1, 15737-2-N09-002, October 1985 [ADAMS Accession No. ML19302G716]
[141] (1989-04-04) NRC Safety Evaluation, Completion of Upper Core Support Assembly Defueling (re 9-15-1988)
[142] Skillman, Gordon R., “The 35th Anniversary of the Three Mile Island Nuclear Power Plant Accident of 1979: Working at TMI During and Following the Accident,” video seminar presented at the U.S. Nuclear Regulatory Commission, Rockville, MD, March 2014

[Retrieved from NUREG/KM-0001 at https://tmi2kml.inl.gov/HTML/Search.html?folder=0d-Video-35th]

[143] GEND-INF-025, Development of a Prototype Gas Sampler for EPICOR II Prefilter Liners (1982-09)

[144] USNRC, “Clarification of Conditions for Waste Shipments Subject to Hydrogen Gas Generation,” Information Notice 84-72, September 10, 1984 [Available at nrc.gov]

[145] Electric Power Research Institute, “TMI-2 Waste Management Experience,” EPRI-TR-100640, April 1992

[146] NUREG-0591, Environmental Assessment, Use of EPICOR-II at TMI-2 (1979-08)

[147] NUREG-0683, Vol. 1, PEIS-Decontamination and Disposal of Radioactive Wastes Resulting from TMI-2 (1981-03)

[148] (1982-10-13) GPU Safety Evaluation, SDS Liner Recombiner and Vacuum Outgassing System

[149] (1982-11-11) GPU Response to NRC (Discussions), SDS Liner Recombiner and Vacuum Outgassing System

[150] (1982-11-23) NRC Review, SDS Liner Recombiner and Vacuum Outgassing System

[151] (1982-12-17) GPU Response to NRC (05-19-1982), Shipment of Spent SDS Liners Requiring Passification Systems

[152] (1982-12-23) NRC Review, Shipment of Spent SDS Liners Containing Gas Passification Systems

[153] (1982-05-21) NRC, Certificate of Compliance 9152, Rev. 1, for Model CNS-1-13C II Shipping Cask

[154] (1982-05-19) NRC Review, Shipment of SDS Waste Vessels (re 05-14-1982)

[155] USNRC, “Three Mile Island Accident of 1979 Knowledge Management Digest: Recovery and Cleanup,” NUREG/KM-0001, Supplement 1, June 2016 [Available at nrc.gov]

5 PYROPHORICITY SAFETY EVALUATIONS

TABLE OF CONTENTS

Note: "NA" (not applicable) means the licensee and the NRC determined that the safety topic was not important for the activity.

5.1 Introduction

5.1.1 Background

The broad subject of pyrophoricity covers essentially the entire fuel cycle, ending with storage or disposal. This document focuses on the evaluations and the application of pyrophoricity safety measures for postaccident TMI-2 conditions and the associated defueling operations. For the TMI-2 activities, pyrophoricity was a subject of intense interest early in the defueling preparations. Interested parties provided analytical support based on experimental results and observed conditions of fine fuel debris dispersed throughout the reactor vessel and reactor coolant system.

The accident produced high fuel temperatures that resulted in massive oxidation and fragmentation of the reactor core. The general state of the core damage was well documented by video camera inspections and core debris sampling. The top part of the reactor core was oxidized and fragmented, leaving about a 1.5-meter-deep void at the top of the reactor core. The surface of the remaining rubble bed consisted of sand- and gravel-like granular debris with an approximate depth of 1 meter. There were regions of once-molten core materials in eutectoid phases that consisted primarily of zirconium-uranium-oxygen and regions of largely undamaged fuel assembly stubs under the loose debris.

- ***Concerns***. The TMI-2 PEIS [(1)] related to the decontamination and disposal of radioactive wastes at TMI-2 suggested in 1981, about 2 years following the accident, that pyrophoric materials might have been present within the core debris. These materials could present a safety concern during reactor defueling.

  - *Pyrophoric Materials*. The pyrophoricity issue primarily concerned the existence of metals associated with the core debris that could exist in pyrophoric forms. The principal ceramic materials (uranium dioxide and zirconium dioxide) would not support combustion; thus, these materials were determined to not exhibit pyrophoric characteristics. Generally, metals capable of exothermic reaction with oxygen can exhibit pyrophoric behavior. The list of metals commonly referred to as combustible metals includes those from periodic table groups IA, IIA, IVA, and IIB (i.e., lithium, sodium, potassium, magnesium, calcium, zinc, thorium, uranium, plutonium, titanium, zirconium, and hafnium).

  - *Zirconium Content*. The TMI-2 active core region was composed of various ceramics, metals, and alloys, with the principal metallic constituents being Zircaloy-4 [(a)] (18.8 weight percent), control rod alloy silver-indium-cadmium (2.2 weight percent), stainless steel (1.3 weight percent), and Inconel (1.0 weight percent). The metal of principal interest with respect to pyrophoricity at TMI-2 was Zircaloy-4, an alloy of which the major constituent was zirconium (98 weight percent). Other materials for which pyrophoricity was considered included zirconium-uranium dioxide solids and zirconium hydride thought to have been formed from chemical reactions during the accident. In addition to Zircaloy metal, the principal compounds of zirconium associated with the core debris were most likely zirconium

[a] Editor's Note: Zircaloy is a trademark name of a common subgroup of zirconium alloys used for fuel rod cladding.

dioxide and zirconium hydride (not necessarily stoichiometric) and formations of zirconium-uranium solids from Zircaloy-uranium dioxide fuel chemical reactions. Zirconium dioxide was in its maximum oxidation state. Zirconium hydride, even if stoichiometric, could thermodynamically undergo oxidation and was potentially reactive.

The planned recovery programs at TMI-2 included head lift, plenum removal, and other operations associated with fuel removal and cleanup of the reactor. These operations, along with reactor defueling and fuel canister handling, would expose new core internals and core debris surfaces. If any pyrophoric material was created during the accident, these operations would provide the most likely occasion for pyrophoric events. Consequently, engineering assessments examined the pyrophoric potential of these operations.

- *Conclusion*. It was known before the accident that finely fragmented metallic zirconium was pyrophoric. Even though evidence showed that most of the zirconium was oxidized by the accident, the fragmented Zircaloy cladding caused concern because, if core debris was exposed to air, there was a possibility of a pyrophoric reaction and a consequent large metal fire. In analytical studies, samples of fine core debris were subjected to ignition tests. The studies indicated no potential for pyrophoric reactions. When the water level was lowered to the top of the plenum assembly before head lift, it uncovered fine debris resting on top of the plenum and allowed air into the system; however, no pyrophoric reactions occurred. During the entire reactor defueling operations, including sawing and plasma arc cutting of core materials, no pyrophoric reactions were observed. [2, 3]

5.1.2 Chapter Contents

This chapter presents pyrophoricity safety evaluations of the postaccident TMI-2 and defueling operations. It describes the many studies performed, thereby enabling the reader to understand the thinking of the analysts at the time; the expectations and the reality; the uncertainties in data; measurement and mitigation methods; and the high-level timeline of cleanup activities.

The evaluations presented in this chapter ensured that all activities that could generate pyrophoric reactions from fuel debris were addressed and consequences evaluated; controls were maintained in accordance with the requirements of the plant’s license, technical specifications, procedures, and applicable regulatory requirements; and adequate contingencies were developed for normal operations and accident conditions.

Key activities that were initially of concern with pyrophoricity included: (●) reactor vessel underhead characterization; (●) core (bore) stratification sample acquisition; (●) reactor vessel head removal operations; and (●) early defueling. Results of research studies and experience with these early activities proved that pyrophoricity was not a safety concern.

Section 2 summarizes the key studies that supported the safety evaluations. The remaining sections present the safety evaluation for each applicable cleanup activity or system. Section 8 presents additional safety evaluations that are unique to this safety topic at TMI-2. Section 9 lists endnotes for references cited throughout this chapter.

5.2 Key Studies

This section provides a high-level chronological progression of pyrophoricity analyses and safety evaluations performed following the TMI-2 accident to support cleanup and defueling activities. Each subsection below summarizes a document. The intent is to help the reader understand the thinking that occurred with the progression of time and increase in knowledge of the postaccident plant conditions throughout defueling.

5.2.1 Programmatic Environmental Impact Statement
(NRC, NUREG-0683, Vol. 1, March 1981)

This report [4] was an overall study of the activities necessary for decontaminating the facility, defueling, and disposing of the radioactive wastes. The NRC reviewed the potential for fire hazards due to zirconium hydride formation.

The agency concluded that zirconium hydride ignition was unlikely and based its conclusion on the following considerations: (●) Review of data on core conditions indicated there would not be significant amounts of zirconium hydride present in the reactor vessel. (●) Only finely fragmented zirconium hydride, in powder form, when exposed to air (oxygen) would be pyrophoric. The presence of hydrided Zircaloy cladding in the powdered state would be readily identified by visual inspection with underwater video monitoring during the core inspection (before any defueling operation). Precautionary actions would be taken if hydrided Zircaloy was found to ensure that no fire hazard would occur. (●) Defueling operations would be performed with water coverage, given that zirconium hydride would not ignite underwater.

5.2.2 TMI-2 Leadscrew Debris Pyrophoricity Study
(Pacific Northwest Laboratory, GEND-INF-044, April 1984)

This report [5] examined debris removed from the surface of a control rod drive mechanism leadscrew to assess the potential for the debris becoming pyrophoric. Elemental analyses were performed to identify candidate crystalline phases that could be pyrophoric, and x-ray diffraction was used to determine if any of these phases was actually present. These analyses observed that none of the candidate crystalline phases was found. Based on differential scanning calorimetry, no exothermic reactions were observed upon heating the debris to about 500 degrees Celsius in air. Particle size distributions of the debris were obtained from analyses of micrographs of the debris and a light blockage instrument. These analyses indicated that most of the debris volume contained particles larger than 10 micrometers; however, the majority of the particles were smaller than 10 micrometers. Gamma spectroscopy indicated that after debris removal, most of the radioactivity in the debris and on the leadscrew was due to mixed fission products such as cesium-137 and cesium-134.

5.2.3 TMI-2 Pyrophoricity Studies

(INEL, GEND-043, November 1984)

This report [6] summarized the pertinent literature and experimental data available from TMI-2 recovery program efforts. To quantify the pyrophoric potential associated with the accident-related materials from TMI-2, core debris material was collected from the TMI-2 reactor vessel for experimental evaluation of its pyrophoric characteristics. Three types of materials from within the TMI-2 reactor vessel were acquired for chemical analysis and pyrophoric potential evaluation: leadscrew debris deposits, plenum cover debris, and core debris.

The following conditions were observed: (●) The leadscrew located immediately above the center fuel assembly at core grid location H8 was removed from the reactor, and loose debris adhering to the leadscrew was collected for analysis and pyrophoricity testing. (●) Television camera inspection revealed that a small accumulation of fine particulate debris existed on the plenum cover (the large horizontal surface on top of the reactor plenum structure). A vacuum suction device that drew a slurry of reactor water and plenum cover debris into a collection bottle obtained samples of this material for testing. The slurry was filtered to concentrate the solids for testing. (●) A number of samples of the gravel-like core debris were acquired from the surface and from beneath the surface of the rubble bed, using a specially designed sampling tool. Specimens were withdrawn into shielded transfer casks and shipped to offsite laboratories for evaluation.

The results of these evaluations of the pyrophoric potential of core debris specimens are summarized below for leadscrew deposits, plenum cover debris, and core debris. Conclusions follow.

- ***Leadscrew Deposits***. Three sections, each about 30 centimeters long, were cut from the middle-threaded portion of the H8 leadscrew (Type 17-4 PH stainless steel) for examination and analysis by the Pacific Northwest Laboratory (PNL), Babcock & Wilcox Company's Lynchburg Research Center, GPU, and INEL. [b] PNL provided the most direct information on the pyrophoric potential of the leadscrew deposits by performing differential scanning calorimetry on a specimen of the deposit. Using this thermal analysis technique, PNL determined that the sample did not undergo an exothermic reaction upon heating in air up to about 500 degrees Celsius. The only notable observation from this analysis was a phase change between 310 and 450 degrees Celsius.

PNL also performed x-ray diffraction analysis of the leadscrew debris to determine whether any candidate pyrophoric materials were present (i.e., zirconium metal, zirconium hydride). The x-ray diffraction analysis indicated that these phases were not present in detectable quantities. Based on chemical and thermal characterizations of the debris, PNL concluded that the possibility of a pyrophoric process involving the debris from the leadscrew was very small.

[b] Editor's Note: PNL is currently known as Pacific Northwest National Laboratory; INEL is currently known as Idaho National Laboratory.

(Refer to the GEND-INF-044 report, [7] "TMI-2 Leadscrew Debris Pyrophoricity Study," for further details of the examination.)

- ***Plenum Cover Debris***. A series of rapid tests was developed that could be implemented in the limited, onsite (i.e., at TMI) radioactive material handling facilities. These tests were consistent with the generally listed ignition sources for metals. Specifically, the selected tests were: (•) spark test, in which a high-voltage spark source from a Tesla coil was applied to a portion of the particulate from a sample; (•) strike test, in which a sample under impact in an enriched oxygen environment was observed; and (•) flame test, in which a propane torch flame was passed over a portion of the particulate from a sample. All these tests were pilot-ignition tests; therefore, these tests exceeded the pyrophoric definition. The flame test was an extreme pilot-ignition test.

The plenum cover sample consisted of a small quantity of reactor coolant water, with a few millimeters of fine particulate material on the bottom of the sample container and a reddish-brown suspension above the solids. The plenum cover sample was subdivided and subjected to two spark tests, one strike test, and two flame tests. The tests were documented on videotape. No plenum cover samples exhibited any pilot ignition. Following these tests, the reactor water level was lowered below the plenum cover, exposing this entire surface to air. No pyrophoric reactions were observed.

- ***Core Debris***. Given that core debris samples were needed for a variety of analyses and that the samples were exposed to air following their removal from the reactor vessel, the pilot-ignition tests were limited to nondestructive Tesla coil spark tests (although two small samples were exposed to a propane torch flame test to confirm their inertness). The pilot-ignition tests were performed on core debris specimens that were sieved to produce sample portions with particles within a specific size range. The tests concentrated on the larger particles because the ignition of debris fines could not be construed as a major concern. The tested specimens were both dry and moist. A moist specimen was defined by having about 10 weight percent water added to the sample. This value was about midrange of the 3 to 16 weight percent range, which was given as the amount of water that enhances pyrophoricity. No pilot ignition was observed in any of the spark tests that included the smallest particles, which had been expected to ignite because of their high surface-to-volume ratios.

Two additional pilot-ignition tests, using a flame as a source of pilot ignition, were performed on individual pieces of material less than 4000 micrometers in size from the sample that was obtained at the H8 leadscrew location, at a depth of about 56 centimeters into the core rubble bed (also known as the 8H-B sample). Both of these pieces weighed less than 1 gram. One of the pieces appeared to be a cladding tubing fragment and the other to be a porous ceramic material. No observed pilot ignition occurred in the flame tests with either of these particles.

- ***Conclusion***. The general recommendation, from a review of the literature, was that educated caution must be used in handling potentially combustible metallic materials such as zirconium-bearing materials. Chemical analyses and pilot-ignition tests on a variety of accident-generated materials (leadscrew deposits, plenum cover debris, and core rubble bed debris) indicated no tendency toward pyrophoricity or pilot ignition. Other observations made during recovery activities (i.e., exposure of the entire plenum cover to air, exposure of bulk core debris samples to air during shipping, handling in air of leadscrews containing surface deposits of particulate debris) also indicated no pyrophoricity for core debris. During tests conducted on simulated core material (nonradioactive powders, such as zirconium, zirconium oxide, Zircaloy-2, Zircaloy-2 hydride, iron, uranium dioxide, and a mixture of these materials), sparking was detected from isolated particle surfaces, but no bulk pilot ignition or sustained propagation was observed.

These observations indicated that reactor bulk defueling could proceed without extraordinary procedures for handling pyrophoric material. Other factors inherent in the defueling process further reduced any pyrophoric hazard. These factors included the following: (●) Defueling would be conducted at ambient temperatures, and use of extreme sources of ignition (i.e., cutting torches) was not planned. (●) Oxidized debris and oxidized core materials would act as a diluent to any pyrophoric materials. (●) Defueling would occur underwater, with the large amount of water acting as a heat sink for any temperature rises that would occur during defueling.

The report concluded that there was minimal concern for a pyrophoric event or sustained propagation during underwater defueling of the TMI-2 reactor vessel. A previous evaluation [8] by the NRC of the TMI-2 pyrophoricity hazard also reached this conclusion. The report considered the possibility of surface oxidation of freshly exposed metallic surfaces during defueling and how such oxidation would generate hydrogen gas from radiolytic decomposition of water from wet fuel debris; however, the data indicated the probability of this was low. Hydrogen gas accumulations were a potential concern; therefore, the report concluded that evaluations should ascertain the hydrogen accumulations and eliminate them, if necessary, through engineering controls (e.g., venting, controlled combustion, absorption, or other engineered safety features).

The uranium-zirconium alloy material and fuel rod stubs represented a potential concentration of partially oxidized metallic zirconium. Accordingly, the report recommended that the following additional studies be performed: (●) pilot-ignition tests and chemical analyses on positively identified specimens of alloy material obtained from existing or future core debris samples and (●) pilot-ignition tests on positively identified specimens of partially oxidized Zircaloy cladding (from rod stub assemblies or fragmented fuel rod cladding pieces) obtained from existing or future core debris samples. [c]

[c] Editor's Note: For both recommended studies, core debris samples used in the studies were those in existence at the time of the report (November 1984) or those that would be taken some time after the report's publication.

5.2.4 Evaluation of Special Safety Issues Associated with Handling the TMI-2 Core Debris

(DOE's Hanford Operations, GEND-051, June 1985)

This report [9] described the methods, techniques, configurations, and conditions that maximized safety and minimized cost and scheduling needs while resolving safety issues relating to the handling of TMI-2 core debris.

- ***Background***. Finely divided metal from a variety of sources was present in the TMI-2 core debris. Zircaloy, cadmium, indium, silver, and stainless steel had all been identified. Zircaloy, an alloy containing 98 percent zirconium, originated primarily from the fuel pin cladding. About 23,000 kilograms of zirconium were originally contained in the reactor core; experts believed that about half of this zirconium had oxidized [10] and that most of the oxidized zirconium was degraded to rubble. Similarly, experts believed that about half of the 93,000 kilograms of uranium oxide had overheated, fractured, reduced to rubble, and mixed with the other core debris. Core debris samples revealed that the early estimate of the particle size distribution of the rubble [11] was reasonably consistent with later estimates. The report stated that about 80,000 kilograms of the fuel debris could have been in the form of full or partial fuel assemblies, frozen agglomerates, or in pieces (greater than 1 inch or 2.5 centimeters) too large for removal with hydraulic vacuuming techniques. It was also determined that about 40,000 kilograms of the debris was probably in rubble sizes less than 1 inch (2.5 centimeters). Possibly half of that debris was thought to be trapped in the matrix between the fuel pins of reasonably intact fuel assemblies and likely to be removed with those assemblies. Only about 5000 kilograms of the debris was believed to be less than 800 micrometers in size. Assuming that some of this debris would be retained in "knockout" canisters and that some would remain trapped within the matrix of fuel assemblies, the report concluded that less than 3000 kilograms of debris would likely be removed in "filter" canisters.

The pyrophoricity potential of the segregated core materials was limited primarily to fine particles. Any debris greater than a few hundred micrometers had a very low potential to be pyrophoric. [12] Based on size alone, the fine materials (less than a few hundred micrometers) found primarily in filter canisters had the greatest pyrophoricity potential. However, even for the fine materials, the potential for pyrophoricity was very small since essentially all of the particles were almost certainly well oxidized at this point. Of the seven small samples of core debris examined by differential thermal analysis calorimetry techniques at Rockwell, only one oxidized significantly, indicating that it was not well oxidized before the analysis. The oxidation occurred between 500 and 900 degrees Celsius.

- ***Results***. The results of these evaluations of the pyrophoric potential are summarized below for zirconium, generic handling and storage of zirconium powder, and other pyrophoricity tests. Conclusions follow.

  - *Zirconium*. Numerous published laboratory experiments had proven that the presence of water did not change the ignition temperature of zirconium. [13] When wet zirconium reached its ignition temperature, it could extract oxygen from water, liberating hydrogen in the process. Dry zirconium burned with a quiet, white-hot flame; wet zirconium burned violently, tossing burning debris into the air.

Despite the inability of water to raise the ignition temperature and the violence of combustion of wet zirconium, water was still universally used to stabilize powdered zirconium. The water successfully acted as a heat sink and typically prevented the zirconium from reaching its combustion temperature. The amount of water used was generally greater than 25 percent of the weight of zirconium. If the water was allowed to boil away during extended heating, the zirconium could burn as described above.

The effectiveness of water as a heat sink was dramatically illustrated by an experiment in which pyrophoric powder (in a drum) was covered by water and the drum was placed between four drums of dry powder. The dry powder was ignited, and the resulting heat melted the central drum down to the water line but did not boil away enough water to ignite the wet zirconium powder.

- *Handling and Stor*age of Zirconium Powder. As a result of these and other experiments, the recommendations for handling and storage of zirconium powder evolved into the following simple points. The following recommendations were not necessarily applicable to the TMI-2 core debris: (•) handle and collect all fine particles less than 850 micrometers underwater; (•) separate zirconium powder from other combustible materials; (•) maintain low moisture content (less than 3 percent by weight) or submerge completely; (•) avoid use of carbon dioxide or water fire extinguishers; (•) avoid dust accumulations; (•) avoid ignition sources; (•) use solid diluents, such as sand and oxides of uranium, in a one-to-one or greater ratio; (•) use argon (or other noble gas) as a cover gas; (•) provide pressure relief for closed systems to minimize rupture potential; (•) separate zirconium work areas from other work areas; (•) exercise extreme care in opening sealed containers; and (•) avoid high temperatures (typically those above the ambient boiling temperature of water).

  Massive pieces of zirconium metal could be safely stored for long periods in or out of water. Massive pieces were defined as sheets 0.3-millimeter thick or thicker and fragments or pieces of which the smallest dimension exceeded 3 millimeters. For particles smaller than massive pieces but greater than 60 micrometers, submerged storage was advised to minimize the chance of a significant temperature rise. For pure zirconium and zirconium alloy powders less than 60 micrometers, storage underwater or in inert gas was mandatory.

- *Pyrophoricity Tests at the DOE's Hanford Operations*. The DOE's Hanford Operations completed a number of laboratory investigations to identify the combustion behavior of finely divided zirconium in shipping containers under drip-dry (i.e., the wet condition of a previously submerged substance after the water was allowed to drain) and bound-water-only (i.e., the wet condition of a substance after the free unbound water was removed) conditions. These investigations concluded that wet zirconium fines did not burn when struck by a spark in an atmosphere of argon with less than 3-percent (by volume) oxygen or in an argon atmosphere initially containing 3-percent oxygen and 4-percent hydrogen (although, for unknown reasons, the hydrogen and oxygen did react). Vacuum-dried zirconium fines showed no reaction to a spark in an argon atmosphere containing 3-percent oxygen and 4-percent hydrogen. However, wet zirconium fines could burn in an argon atmosphere with more than 3-percent oxygen. No reaction occurred when a spark was struck in an argon atmosphere to fine zirconium powder that was wetted with a solution of hydrogen peroxide (an intermediate in the radiolytic production of hydrogen and oxygen). Based on these results, it appeared advisable to maintain materials that might be pyrophoric in an argon atmosphere with less than 3-percent oxygen.

- ***Conclusion***. The TMI-2 core debris was not believed to be pyrophoric. The characteristics that tended to mitigate pyrophoricity in the TMI-2 core debris are summarized as follows: (●) Zircaloy-clad fuel pins and Zircaloy cladding hulls had been demonstrated to be noncombustible (burning was not self-supporting). (●) Most of the core debris consisted of uranium dioxide and zirconium dioxide, which were not combustible. (●) Most of the core debris (uranium dioxide and zirconium dioxide) acted as a solid diluent that reduced the combustibility of any pyrophoric metal particles. (●) Particulate matter in the core debris was heated to high temperatures during the loss-of-coolant accident. Any reactions that could occur at a high temperature should have occurred during the accident. (●) Considerable oxidizing, melting, alloying, and agglomerating of metals occurred in the loss-of-coolant accident so that pure metal fines were probably relatively scarce. (●) No mechanical processes that produced unoxidized fine materials had occurred. Therefore, unoxidized metal surfaces would be relatively scarce. If the defueling process created fresh metal surfaces, particularly when the process created very small chips and fines such as from sawing and grinding operations, the pyrophoricity potential would increase.

The planned core debris environment during removal, handling, and shipping essentially eliminated any pyrophoricity potential during these activities. The core debris would be placed in canisters while underwater, and an inert gas would blanket the core debris in the canisters after dewatering.

These core debris characteristics and environmental controls provided a reasonable basis for believing that no pyrophoricity incident would be encountered while handling the core debris. The report concluded that if any pyrophoric condition occurred during the testing program, it would not impact the defueling and shipping procedures that were being considered at the time; however, it could significantly affect procedures during canister opening, when the debris could be exposed to air.

5.2.5 TMI-2 Core Debris Grab Samples Analysis of First Group of Samples
(INEL, GEND-INF-060, Vol. 1, July 1985)

This report [14] examined the extent and nature of the damage and postaccident condition of the core to assist in assessing the tooling and procedures required to defuel the TMI-2 reactor. One of the principal reactor recovery issues addressed by the core debris examinations was determination of the extent of pyrophoric materials present in the debris, if any.

Pyrophoricity pilot-ignition tests were performed on selected core debris materials to evaluate potential safety hazards to core recovery operations. To demonstrate the procedure, preliminary tests were performed on zirconium hydride powder using a small Tesla coil rated at 50,000 volts. The ignition of this powder was recorded by both videotape and still photography before beginning the actual core debris pyrophoricity tests. An additional method used to produce higher temperatures with a propane torch was tested on the same type of powder before beginning the actual pyrophoricity tests. [d]

The sieve fractions from TMI-2 core Samples 3 and 6 were chosen for pyrophoricity testing. Tests were performed on material removed from size (sieve) fractions ranging from 30 to 4000 micrometers. The quantity of material used for each test ranged from 0.25 to 0.5 gram. This small quantity of material was used to keep radiation exposure of personnel within reasonable limits. Tests were performed under both dry and wet conditions. The dry condition was attained by warming the material for about 30 minutes at 100 degrees Celsius. The wet condition was attained by adding two drops of water to the material.

Tests were also performed on individual wet and dry portions of the samples. No visible pilot ignition was observed for any size fraction of either sample. One particle, of what appeared to be core structural material, from each of the two samples was exposed to the propane torch, and no pilot ignition was observed for either particle.

5.2.6 TMI-2 Core Debris Grab Sample Examination and Results

(INEL, GEND-INF-075, September 1986)

This report [15] presented examination results of 10 core debris grab samples at INEL. The examinations included physical, metallurgical, chemical, and radiological analyses. The phenomenon of pyrophoricity was investigated to evaluate the possibility of a pyrophoric reaction during removal and storage of the damaged TMI-2 core. The report concluded that pyrophoric reactions would be unlikely during defueling. This conclusion was based on a review of literature on pyrophoricity, which concentrated on experiences of the nuclear industry with zirconium metal, combined with evaluation of core material reactions at the high temperatures that occurred during the TMI-2 accident.

A series of ignition tests was performed at INEL on samples of core materials removed from all size (sieve) fractions between 30 and 4000 micrometers of Samples 3 and 6. [e] The samples were tested under both wet and dry conditions, and ignition of the material was attempted using both a Tesla coil and a propane torch. The ignition of the materials did not occur in any of the cases. (Refer to GEND-043, [16] “TMI-2 Pyrophoricity Studies,” for further details of the tests.)

---

[d] Six samples of particulate debris had been removed from the TMI-2 core rubble bed in fall 1983. The samples were sieved to determine the particle size distribution of each sample and the weight of each sieved fraction (i.e., the portion of the sample particles within a particular size range, determined by the sieves used in the process).

[e] Editor’s Note: Similar to the study described in GEND-INF-060, the samples for this study were also sieved to determine the particle size distribution in each sample. Thus, a size (sieve) fraction is a portion of the sample containing particles with a particular size range, determined by the sieves used in the process.

The DOE's Hanford Operations conducted differential thermal analyses on fragmented chips from seven particles (refer to Appendix H to GEND-INF-075, Part 2 [17]). These particles generally showed little thermal activity, although one particle produced a large, broad release of energy (exotherm) of 761 calories per gram, spanning nearly 500 degrees Celsius, starting at about 550 degrees Celsius (Appendix H). This exotherm occurred at much higher temperatures than those observed for samples of zirconium powder and partially oxidized zirconium powder. The report concluded that if the exotherm was due to the oxidation of zirconium, the zirconium in the sample must have been coated with a thick noncombustible layer, possibly an oxide, that protected the sample at lower temperatures. Appendix H contained the results of this test.

5.3 Data Collection Activities

5.3.1 Axial Power Shaping Rod Insertion Test (NA)

5.3.2 Camera Insertion through Reactor Vessel Leadscrew Opening (NA)

5.3.3 Reactor Vessel Underhead Characterization (Radiation Characterization)

- ***Purpose***. To conduct radiation characterization under the reactor vessel head to ensure that adequate radiological protection measures would be taken to keep radiation exposures ALARA during head removal. To achieve this objective, measurements were necessary with the reactor vessel head still in place.

- ***Evaluation: Pyrophoricity***. [18] The underhead characterization program required the lowering of the reactor coolant level in the reactor vessel. Because some of the vessel internals (e.g., leadscrew support tubes, control rod guide assembly tubes, and upper plenum cover plate) would be uncovered and exposed to air, the issue of pyrophoricity was addressed. The licensee's safety evaluation concluded that a pyrophoric reaction was an unlikely event while the reactor coolant level was lowered, for the reasons outlined below:

  - *Larger Particle Sizes*. As evidenced by various incidences of zirconium fires, the rapid burning of zirconium metal was usually restricted to fines of 100 micrometers or less. The formation of zirconium powder during the accident was highly improbable because of the dynamics of the accident.

  - *Protective Oxide Layer*. Even if zirconium fines were formed, they would have been partially or completely transformed to zirconium oxide. This chemical form was a very stable material with no pyrophoric properties. The fines would be more oxidized than larger metal pieces because of the high surface-area-to-volume ratio. At a minimum, the outer surface of any such particle would exhibit an oxide layer due to the oxidation that occurred as the particles were exposed to the reactor coolant for the past 4 years.

  - *Particle Location*. Flow velocities during and following the accident were such that significant quantities of material of any nature would not have been transferred to the upper plenum. The results of the "Quick Look" and "Quick Scan" tests supported this prediction.

- *Particle Dilution*. Any fines on the plenum would have been diluted with other fully oxidized and nonpyrophoric materials, which would inhibit sustaining a pyrophoric reaction, as evidenced by examination of the material from the surface of the leadscrew.

- *Experience from Leadscrew Removal*. Debris particles from the control rod drive mechanism leadscrew, which was removed during Quick Look, did not ignite during various burn tests. The leadscrew sample was obtained from a section of leadscrew that was close to the plenum cover and in the normal flowpath to the cover. Other samples of residue material from the reactor coolant system that had been extensively handled and examined in air failed to exhibit any observable pyrophoric properties.

- *Experience from Reactor Coolant System Draindown*. During and after the Quick Look program, the upper tubesheets of both steam generators were exposed to an air environment during the reactor coolant system draindown. The material on these tubesheets was expected to be similar to what might be found on the plenum cover. No indications of a pyrophoric occurrence were evident during the 4 months that the tubesheets were exposed to air.

- ***Evaluation: Pyrophoricity (Results)***. [19] The licensee's pyrophoricity evaluations from the underhead characterization program concluded that the possibility of a pyrophoric event due to the presence of zirconium-bearing material was highly unlikely. However, the precautions that would be put in place reflected sufficient prudence to permit proceeding with the underhead characterization program. This conclusion was based on the observations discussed below.

- *Minimum Core Debris on the Plenum Cover*. The Quick Look videotapes showed that the top surfaces of the control rod guide assembly's first and second support plates had only light deposits, typically found in normal plants. This indicated that the plenum cover should also be free of debris. The videotapes also showed that the bottom surface of each support plate was free of deposits. This observation indicated that there was no reason to believe there was any debris on the inside surface of the reactor vessel head.

  Visual examination of the removed leadscrews indicated only a thin layer of material on their surfaces. No substantial buildup of material was observed on or between the horizontal surfaces of the threads. This observation further substantiated the premise that little core debris was carried to the plenum cover during or after the accident.

  This conclusion was based on the flow conditions estimated to exist at the time of the accident. In particular, the principal means by which debris could have reached the plenum top cover and inside surface of the vessel head was by entrainment in fluid flowing upward inside of the control rod guide assemblies. This bypass flow was a small fraction of the total flow. With one reactor coolant pump running, as had occurred after the accident, the vertical velocity within a guide assembly was estimated to be on the order of 0.3 feet per second in the region between support plates. This velocity was low enough to permit most of the entrained fuel debris to settle out before it could reach the top end of a guide assembly. Only small particles, on the order of tens of microns in size or less, could have reached the upper end of the guide assembly. Because of their small size and the core conditions that resulted in their formation, any particles that reached the upper plenum surface were likely to have completely oxidized.

Quick Scan measurements and calculations concluded that the radioactivity levels in the upper plenum were most likely a result of cesium deposition on all vertical and horizontal surfaces, not just debris on the horizontal upper surface of the plenum only.

- *Filter Material and Control Rod Drive Mechanism Leadscrew*. Experimental evidence supported the contention that any core debris on the reactor plenum cover was not pyrophoric. Samples of core debris that collected outside of the vessel on filters in the reactor water letdown-purification system were analyzed. These analyses showed that the filter debris consisted of small particles, from less than 1 and up to 50 micrometers, with a mean particle size of 6 micrometers. The particles were found to be composed primarily of non-fuel-rod components. Over 50 percent of the particles contained stainless steel, Inconel, and silver-indium-cadmium control rod material constituents. Based on a limited sampling, most of the zirconium-bearing particles were reaction products with uranium, control materials, or structural materials. Thus, the zirconium-bearing particles were alloyed as well as physically mixed with other particles, thereby reducing any potential pyrophoricity. Furthermore, although all of the zirconium compounds in the debris had not been identified, zirconium dioxide was confirmed. This supported the contention that the zirconium present in the debris had undergone oxidation and reaction with other materials and was therefore not pyrophoric.

  The licensee examined a small sample (estimated to be 20 to 50 milligrams) of particulate debris obtained during the removal of one of the leadscrews. While the exact origin of this sample was uncertain, it showed no pyrophoric reaction upon air drying. It was statically charged; the static charge did not cause the particles to react. A sample of the shavings generated during the cutting of the leadscrew was obtained. The cut was made at an elevation that corresponded to a position on the leadscrew near the plenum cover. About 100 milligrams of the cuttings containing leadscrew debris were air dried, heated on a hot plate, struck with an electric spark, and heated directly with a flame. No pyrophoricity was indicated; no burning, smoking, or any indication of pyrophoricity was observed. While the quantity of leadscrew debris in the sample was small, it represented a sample where new surfaces were exposed; thus, this debris should have been highly reactive. In addition, neither the cutting operations nor the long-term storage in the containment building of the extracted leadscrews showed signs of a pyrophoric reaction

  A sample of the debris from a 12-inch-long section of the H-6 leadscrew was sent to Babcock & Wilcox for detailed analyses. Preliminary results confirmed the presence of significant quantities of zirconium in the debris along with iron (the major component), uranium, tellurium, copper, and nickel. The principal form of zirconium was identified as an intermetallic oxide of the form $FeZrO_4$.

One 9-inch-long section of the leadscrew was sent to a second private laboratory for detailed analyses. Preliminary results of these tests and analyses indicated the presence of very little unalloyed zirconium. The analysis of this section of leadscrew was specifically directed toward the detection and characterization of any pyrophoric material. In the particles analyzed as of the publication of the reactor vessel underhead characterization report, the zirconium existed mainly in an alloyed form with silver, uranium, or iron. The presence of zirconium dioxide particles had been confirmed. Neither the free metal form nor zirconium hydride had been identified in the particles.

In summary, the analyses of the filter debris and leadscrew samples appeared to confirm that the TMI-2 particulate debris was not pyrophoric.

- *Zirconium Oxidation*. The concern over the potential existence of pyrophoric materials in TMI-2 was focused on the possibility of metallic Zircaloy and zirconium hydride fines existing on the horizontal surface of the reactor plenum cover. The way the fuel deteriorated during a severe accident would make the presence of these species in a pyrophoric form highly unlikely. Zircaloy, being a ductile metal even after irradiation, would not break into small particles under the high-temperature steam environment of a light-water reactor accident. Rather, as the material oxidized, the oxide broke up as a consequence of thermal shock or abrasion.

  Zircaloy that had not substantially oxidized (either to zirconium dioxide or to the oxygen-stabilized alpha phase) retained most of its ductility. Therefore, any zirconium-bearing particles carried to the plenum cover during the TMI-2 accident were expected to have been largely converted to oxide and not pyrophoric. Larger particles from reactor accident experiments (greater than a few millimeters) were sometimes seen metallographically to be only partially oxidized. For such particles, a Zircaloy metal zone was surrounded by layers of an oxygen-stabilized alpha phase and zirconium dioxide. In principle, such particles could be pyrophoric if they were broken and a fresh metal surface was exposed to the air. However, these large particles seemed to be able to dissipate heat, so that they merely oxidized when the fresh metal was exposed to air, instead of burning. Experience from the postirradiation examination of kilogram quantities of fuel debris from light-water reactor accident examinations indicated that zirconium-bearing particles could be collected, handled, sieved, weighed, etc., in both the wet and dry condition without producing any sustained pyrophoric reactions.

- ***Evaluation: Pyrophoricity (Precautions)***. [20] The licensee's safety evaluation concluded that a pyrophoric event was highly unlikely during the underhead characterization program. However, the following precautions would be taken to minimize the potential for a pyrophoric event.

- *Video Inspections*. With the reactor vessel water level above the plenum cover, a video inspection would be performed through the manipulator support tube to better determine the quantity of any core debris on the cover.

- *Sample Examinations*. After the video inspection, samples of the observed debris on the cover would be obtained, provided that sufficient debris was available. At least two samples would be collected. (This could be one sample split into several portions.) One sample would be subjected to an immediate test to ascertain if it would burn, as was done with the leadscrew sample. The other sample would be analyzed for archive purposes in a laboratory to determine its physical and chemical properties. If insufficient debris was available to obtain a sample, the possibility of a pyrophoric reaction was negated.

- ***NRC Review: Pyrophoricity***. (21) The NRC's safety evaluation considered the phenomenon of spontaneous combustion of small pieces of zirconium, zirconium hydride, and partially oxidized zirconium. The review concluded that the pyrophoric reaction would not initiate underwater, but once initiated in air, the reaction could continue underwater. Once the top surface of the plenum was exposed to air, there was a potential for this reaction to occur, and the licensee, various consultants, and the NRC extensively evaluated this potential. The NRC review concluded that the primary system flow dynamics during the 1979 accident were unlikely to have transported large quantities of pyrophoric material to the top of the plenum. If such material had been transported to the void under the reactor vessel head, it would have oxidized sufficiently to render it nonpyrophoric. As a precaution, before lowering the reactor water level below the plenum, a video inspection would be made of a portion of the top surface of the plenum. If any material was present on the surface of the plenum, a sample of the material would be obtained and analyzed for pyrophoricity. The decision to lower the water level below the top of the plenum would be based on the results of the visual inspection and sample analysis.

5.3.4 Reactor Vessel Underhead Characterization (Core Sampling)

- ***Purpose***. To obtain up to six specimens of the core debris using specially designed tools. These tools would be lowered into the reactor to extract samples of the loose debris and to load the samples into small, shielded casks for offsite shipment and analysis. The analysis of the samples would: (●) identify the composition of the particulate core debris; (●) determine particle size; (●) determine fission product content; (●) determine fission product leachability from the debris; (●) analyze the drying properties of the debris; and (●) determine whether pyrophoric materials existed in the core debris. This examination was a follow-on activity to the underhead characterization study.

- ***Background***. Three shipping casks [f] were considered (22) for transporting the core debris samples to the laboratory. The selected cask was the modified and recertified Model

[f] Editor's Note: While large shipping containers of radioactive materials may often be referred to as "shipping casks," the proper term for such containers, when loaded with contents and in their transportation configuration, is "package." See 10 CFR Part 71, "Packaging and Transportation of Radioactive Material," Section 71.4, "Definitions."

CNS 1-13C Type B shipping cask, with the sample in a Specification 2R container. An earlier version of this cask was used to ship spent submerged demineralizer system liners. [23]

- ***Evaluation: Pyrophoricity***. [24] The subject of a possible pyrophoric reaction was addressed in the licensee's safety evaluation report [25] for radiation characterization under the reactor vessel head. The discussion in that report primarily addressed the possibility of a pyrophoric reaction of debris on the plenum cover. For the core debris sampling program, the debris in question was that of the actual core; however, the potential for a pyrophoric reaction of the sample was still considered highly unlikely for the reasons outlined below.

  - *Unlikely Formation of Fines*. As evidenced by various incidences of zirconium fires, the rapid burning of zirconium metal was usually restricted to fines of 100 micrometers or less. The formation of zirconium powder during the accident was extremely unlikely because of the dynamics of the accident.

  - *Dilution of Fines*. Fines that potentially were in the sample would be diluted with other fully oxidized and nonpyrophoric materials. These other materials would inhibit a sustained pyrophoric reaction, as evidenced by examination of the material from the surface of the leadscrew.

  - *Experience from Previous Sample Examinations*. Debris from the control rod drive mechanism leadscrew, which was removed during the Quick Look examination, did not ignite during various burn tests. Other samples of residue material from the reactor coolant system that were extensively handled and examined in air did not exhibit any observable pyrophoric properties.

  - *Hazard Evaluation*. Although the possibility of withdrawing pyrophoric material from the debris bed was expected to be small, the possible consequences of having such material in the debris sample were considered. The nominal size sample to be withdrawn was about 1 cubic inch, as confirmed by testing. Any debris on the external parts of the sampler was expected to be washed off during withdrawal and did not need to be considered. If this sample contained 10-percent zirconium (a fuel assembly was about 10-percent zirconium by volume) and if it were to totally burn when exposed to the air, it was estimated that 123 British thermal units of energy would be released. An explosion of this material was not considered possible. This heat would, assuming perfect heat transfer, raise the specimen holder about 200 degrees Fahrenheit. This was insufficient to damage the holder, the manipulator tube, or the sample cask. Therefore, the evaluation concluded that a possible pyrophoric occurrence would not increase the radiological and safety hazards that were considered in the evaluations for the debris sample program.

- ***NRC Review: Pyrophoricity***. [26] The NRC evaluated the potential for a pyrophoric reaction during the core debris sampling. The evaluation concluded that there was little likelihood that pyrophoric zirconium materials would be present in the core. Additionally, the evaluation verified that should a pyrophoric event occur during sampling, the reaction and associated heat energy would be contained in the sampling mechanism and no undesirable radiological or safety consequences would result.

- *Introduction*. The licensee proposed [27, 28] to carry out an underhead characterization study and core debris sampling. These operations would expose previously water-covered reactor surfaces and samples of the core debris to containment building air. Based on a previous NRC evaluation and a 1957 paper, [29, 30] appreciable amounts of zirconium hydrides could have formed during the TMI-2 incident. In a finely divided form, zirconium hydrides could be pyrophoric. If hydrides were present on reactor surfaces and in the core debris, it was postulated that they could react violently when exposed to containment building air during the proposed underhead characterization and core sampling procedures.

- *Zirconium Hydride Production*. The 1957 paper indicated that bulk zirconium metal or zirconium hydride was normally protected from reaction with air, water, or hydrogen by a tight, impervious surface film of zirconium dioxide. Even as a powder, the metal or hydride could be handled in air at ordinary temperatures without burning. However, incidents had occurred elsewhere in which finely divided zirconium metal or hydrides ignited spontaneously and burned violently in air. [31]

  The large amounts of hydrogen generated during the TMI-2 incident could conceivably have reacted with the hot Zircaloy cladding to produce pyrophoric zirconium hydrides. The 1957 paper stated that pyrophoric zirconium hydrides are formed by the reaction of dry hydrogen gas at 400 degrees Celsius (degrees C) with a Zircaloy surface. During the TMI-2 accident, high-pressure steam was also present. At high temperatures, zirconium hydrides react with steam to form zirconium oxide and hydrogen gas.

  The presence of steam and the temperature conditions during the accident made it unlikely that significant quantities of zirconium hydride in a pyrophoric condition were produced during the accident. In regions of the reactor core below about 400 degrees C, the hydriding reaction would have been slow. In regions above about 700 degrees C, the hydrides would have reacted with steam. Any hydride formed at intermediate temperatures would have been in a narrow band between the uncovered and the cooled regions of the core. In the finely divided condition required for pyrophoricity, it was expected that this material would be dispersed and mixed with inert components of the core debris. The inert diluents would help to dissipate reaction heat and prevent the development of pyrophoric conditions.

- *Sample Examinations*. Analysis of solids filtered from the reactor coolant solution and of the thin films scraped from the surfaces of the leadscrews removed from the reactor head indicated the absence of zirconium metal and hydride particles. Over 50 percent of the particles filtered from the coolant contained stainless steel, Inconel, and silver-indium-cadmium control material constituents. Most of the zirconium-containing materials were reaction products with uranium, control materials, or structural materials. Some zirconium dioxide particles were identified. A larger proportion of zirconium dioxide would have been expected from the reaction of a mixture containing enough zirconium hydride to be pyrophoric.

Similarly, analysis of the thin film scraped from a section of the removed control rod drive mechanism leadscrew showed the presence of some zirconium compounds but the main component was iron, along with some uranium, tellurium, copper, and nickel. The principal form of zirconium was identified as an intermetallic oxide of the form $FeZrO_4$. In the analysis of another section of the leadscrew, zirconium was formed principally in an alloyed form with silver, uranium, or iron. No free zirconium metal or zirconium hydride was found. Some zirconium dioxide particles were identified but not in the amounts that a pyrophoric mixture would have produced.

These analytical results and the observation of only light deposits on the upper reactor surfaces in the Quick Look videotapes indicated a low probability of any significant pyrophoric activity when lowering the reactor coolant level. As a further precaution, while the upper plenum surface was still covered with water, any debris deposits that were found would be sampled and tested for pyrophoricity.

- *Hazard Evaluation*. The licensee calculated that the specimen holder temperature would be increased by about 200 degrees Fahrenheit by the complete oxidation of a 1-cubic-inch sample of core debris containing 10-percent zirconium metal. The NRC independently verified the calculations and agreed that the assumptions and calculational method were conservative and that the burning of this amount of zirconium would not damage the holder, the manipulator tube, or the sample cask. This provided additional assurance that the core debris sampling operations could be carried out safely even in the unlikely event that the debris contained pyrophoric zirconium materials.

- *Conclusion*. Based on the above considerations, the NRC had reasonable assurance that pyrophoric zirconium materials would not be present in the TMI-2 degraded core in sufficient quantities to interfere with the safe execution of the proposed procedures for underhead characterization and core sampling. Therefore, the NRC concluded that the proposed procedures were acceptable.

5.3.5 Core Stratification Sample Acquisition

- ***Purpose***. To perform the activities associated with: (●) installation, operation, and removal of the core boring machine; (●) acquisition of the core samples; (●) transfer of the samples from the machine to the defueling canisters; and (●) viewing of the lower vessel region through the bored holes.

- ***Evaluation: Pyrophoricity***. [32] The licensee's safety evaluation stated that, to avoid the possibility of a hypothetical pyrophoric reaction, core bore samples would be maintained in an essentially oxygen-free environment following withdrawal from the reactor coolant system water. This would be done by purging the core barrel and core sample with an inert gas as they were removed from the reactor. Purging would be done by inserting a wand into the lowest section of the drill tube containing the sample before its being raised from the water. As the water drained out of the drill tube and core barrel, inert gas would replace the water. After the last section of drill tube was removed from the top of the core barrel, a plug would be inserted into the top of the core barrel before transferring the sample to the defueling canister. The evaluation referred to the licensee's safety evaluation report [33] justifying the nonseismic design of TMI-2 postaccident systems for a more detailed discussion of pyrophoric events in the reactor coolant system. The evaluation determined that the heat generated by drill bit friction would not increase the potential for a pyrophoric event since this heat would be readily dissipated by the flush water from the boring tool. The drill unit would be automatically shut down upon loss of flush water.

- ***NRC Review: Pyrophoricity***. [34] The NRC considered the creation of pyrophoric materials followed by a burn as potentially safety significant during the core stratification sampling program.

  - *Observations*. The NRC's review of the proposal noted the following observations.

    - *Protective Oxide Layer*. Bulk zirconium or Zircaloy is normally protected from reaction with air or water by a tight, impervious surface film of zirconium oxide. The operation of core drilling through metallic Zircaloy structures would produce sizable quantities of finely divided metallic Zircaloy particles. Even as a powder, the metal could normally be handled in air at ordinary temperatures without burning. However, when ignited, finely divided zirconium or Zircaloy metal burns violently in air. At high temperatures, the metal could burn in steam and generate hydrogen gas and zirconium dioxide.

    - *Water Cooling*. A review of industry practices showed that the key to preventing pyrophoric reactions was to provide conditions in which the heat of the metal oxidation reaction would be rapidly conducted away from the reacting surface. At low surface temperature, the rate of oxidation would be slow, and a protective oxide film would be formed, preventing a runaway pyrophoric reaction. Finely divided zirconium or Zircaloy particles could be prevented from igniting by submerging them in water. Zirconium powders and machining chips were therefore commonly stored underwater. No cases of spontaneous ignition under these conditions had been reported. Some studies indicated that 25-percent water by volume was enough to provide safe storage for finely divided zirconium.

    - *Core Bore Design*. A review of the proposed core boring operation showed that machining operations on zirconium alloys or zirconium hydride compositions could sometimes be performed with a cooling gas stream directed at the cutting edge of the machine tool to carry away the heat generated by the cutting operation. The rapidly cooled machined chips would not burn. The flush water surrounding the cutting bits of the core drilling tool would provide a very efficient heat sink to dissipate the heat generated by the drilling operation. The drill unit was designed to shut down automatically upon loss of flush water. Therefore, there was reasonable assurance that no pyrophoric reactions would be caused by the heat generated by drill bit friction.

The remaining core sampling operations included replacing the water in the core barrel with inert gas and transferring the core sample into a fuel canister. There was low potential for pyrophoric reactions during these operations since no friction heat would be generated and the core sample would contain little finely divided metallic material.

  - *Reactor Coolant Filtration*. Fines and cutting chips filtered out of the reactor coolant after the core drilling operation were also to be kept submerged in water and would not be subject to friction heat. Further, the metallic components of this material would be mixed with the inert nonflammable components, diluting the pyrophoric potential. Therefore, there would be no significant risk from this source.

  - *Inert Gas Cover (Canisters)*. The provision of an inert gas atmosphere in the defueling canisters containing the core samples and the filtered samples would further protect against the possibility of pyrophoric reactions.

- *Conclusion*. The NRC concluded that the risk of pyrophoric reactions during the removal of loose core debris before the core drilling operations would be low for the reasons given in the agency's safety evaluation report [35] for the underhead characterization and core sampling. Tests on samples of loose core debris showed that the samples contained very little unoxidized material and were not ignitable. The pyrophoric risk was therefore not significant even though large quantities of materials were involved. The risk was further reduced by collecting and storing the loose debris underwater or in an inert gas atmosphere. On the basis of the above evaluation, the NRC concluded that there was no significant risk of pyrophoric reactions during the proposed core stratification sample acquisition activities. The NRC further concluded that the proposed activities were acceptable with respect to pyrophoric issues.

5.3.6 Use of Metal Disintegration Machining System to Cut Reactor Vessel Lower Head Wall Samples

- ***Purpose***. To remove metallurgical samples from the inner surface of the lower reactor vessel head using the metal disintegration machining system. An international research program used these samples to study core melt and reactor vessel interactions. This program was conducted after the reactor vessel was defueled.

- ***Evaluation: Pyrophoricity***. [36, 37] Given that no pyrophoricity problems had been experienced during defueling operations, the licensee's safety evaluation concluded that pyrophoricity concerns during the sampling activities were bounded by evaluations provided in the safety evaluation report [38] on defueling activities.

- ***NRC Review: Pyrophoricity***. [39] The NRC determined that issues related to pyrophoricity were within the bounds of previous agency evaluations.

5.4 Pre-Defueling Preparations

5.4.1 Containment Building Decontamination and Dose Reduction Activities (NA)

5.4.2 Reactor Coolant System Refill (NA)

5.4.3 Reactor Vessel Head Removal Operations

5.4.3.1 Polar Crane Load Test (NA)

5.4.3.2 First-Pass Stud Detensioning for Head Removal (NA)

5.4.3.3 Reactor Vessel Head Removal Operations

- ***Purpose***. To prepare for reactor vessel head removal to perform the lift and transfer of the head and conduct the activities necessary to place and maintain the reactor coolant system (RCS) in a stable configuration for the next phase of the recovery effort.

- ***Evaluation: Pyrophoricity***. [40] The licensee's safety evaluation report [41] for radiation characterization under the reactor vessel head had previously addressed the possibility of a pyrophoric reaction of zirconium fines in the vessel and concluded that such an event was highly unlikely when the reactor coolant level was lowered. Since the RCS level for head removal would be the same as for the underhead characterization program, the report concluded that the likelihood of a pyrophoric reaction during head removal would not increase. As part of the underhead characterization program, samples were taken of the debris on the plenum cover plate. Three tests were performed on two samples to establish if the materials showed pyrophoric characteristics and to illustrate the inertness of the material to ignition. The results of the tests confirmed that the material sampled from the plenum cover displayed no pyrophoric characteristics.

- ***NRC Review: Pyrophoricity***. [42] During reactor vessel head lift activities, when the reactor coolant would be lowered about 1 foot below the plenum cover, the potential existed for a pyrophoric reaction of any material on the cover when it was exposed to air. The NRC extensively evaluated [43] the issue of pyrophoricity as it related to the head lift program. The NRC's response cited the results and experience of the underhead characterization study of 1983/1984 and other cleanup-related studies in concluding the following: (●) Little material (about 1 millimeter in depth) was present on the plenum surface. (●) Flame and spark tests indicated the material on the plenum surface was not pyrophoric. (●) Material filtered from the RCS during the accident lacked any pyrophoric content. (●) Material scraped from control rod drive mechanism leadscrews lacked any pyrophoric content. (●) Samples of material removed from the damaged core had not shown any tendency to undergo a pyrophoric reaction.

Additionally, flame and spark tests on several core samples did not indicate any pyrophoric characteristics in the samples. Accordingly, the NRC concluded that there was little potential for a pyrophoric reaction during the head lift program.

5.4.4 Heavy Load Handling inside Containment (NA)

5.4.5 Heavy Load Handling over the Reactor Vessel (NA)

5.4.6 Plenum Assembly Removal Preparatory Activities (NA)

5.4.7 Plenum Assembly Removal (NA)

5.4.8 Makeup and Purification Demineralizer Resin Sampling (NA)

5.4.9 Makeup and Purification Demineralizer Cesium Elution (NA)

5.5 Defueling Tools and Systems

5.5.1 Internals Indexing Fixture Water Processing System (NA)

5.5.2 Defueling Water Cleanup (NA)

5.5.2.1 Defueling Water Cleanup System (NA)

5.5.2.2 Cross-Connect to Reactor Vessel Cleanup System (NA)

5.5.2.3 Temporary Reactor Vessel Filtration System (NA)

5.5.2.4 Filter-Aid Feed System and Use of Diatomaceous Earth as Feed Material (NA)

5.5.2.5 Use of Coagulants (NA)

5.5.2.6 Filter Canister Media Modification (NA)

5.5.2.7 Addition of a Biocide to the Reactor Coolant System (NA)

5.5.3 Defueling Canisters and Operations (NA)

5.5.3.1 Defueling Canisters: Filter, Knockout, and Fuel (NA)

5.5.3.2 Replacement of Loaded Fuel Canister Head Gaskets (NA)

5.5.3.3 Use of Debris Containers for Removing End Fittings (NA)

5.5.3.4 Fuel Canister Storage Racks (NA)

5.5.3.5 Canister Handling and Preparation for Shipment (NA)

5.5.3.6 Canister Dewatering System (NA)

5.5.3.7 Use of Nonborated Water in Canister Loading Decontamination System (NA)

5.5.4 Testing of Core Region Defueling Techniques (NA)

5.5.5 Fines/Debris Vacuum System (NA)

5.5.6 Hydraulic Shredder

- ***Purpose***. To utilize a hydraulically powered shredder to reduce the size of fuel pins and other core debris and to facilitate the loading of fuel canisters or debris buckets.

- ***Evaluation: Pyrophoricity***. [44] Safety concerns related to this operation, including the potential for a pyrophoric event, were previously addressed in the licensee's safety evaluation report [45] for core stratification sample acquisition. Use of the core bore machine on the lower core support assembly did not alter the consequences of these issues.

- ***NRC Review: Pyrophoricity***. [46] The NRC concluded that the potential safety issues associated with pyrophoricity in the reactor vessel and internals did not significantly differ from those issues associated with previously reviewed and approved operations.

5.5.7 Plasma Arc Torch

5.5.7.1 Use of Plasma Arc Torch to Cut Upper End Fittings

- ***Purpose***. Use of a plasma arc torch to cut the upper end fittings inside the reactor vessel to allow the fittings to be placed directly in fuel canisters for transfer from the reactor vessel to the fuel canal. This was the first use of the plasma arc torch inside the reactor vessel.

- ***Evaluation: Pyrophoricity***. [47] The licensee's safety evaluation for defueling addressed pyrophoricity concerns during defueling activities. This evaluation concluded that it would not be possible to sustain a pyrophoric reaction during planned defueling activities or postulated accidents. The use of the plasma arc torch did not affect the conclusions of the defueling safety evaluation. In addition, offsite testing of the plasma arc torch with 7/16-inch-diameter Zircaloy tubes underwater indicated no pyrophoric tendency; in fact, the flame was self-extinguishing as the torch moved past the rods.

- ***NRC Review***. [48] Editor's Note: The NRC's safety evaluation report did not specifically address this evaluation topic.

5.5.7.2 Use of Plasma Arc Torch to Cut the Lower Core Support Assembly (NA)

5.5.7.3 Use of Plasma Arc Torch to Cut Baffle Plates and Core Support Shield (NA)

5.5.7.4 Use of Air as Secondary Gas for the Plasma Arc Torch (NA)

5.5.8 Use of Core Bore Machine for Dismantling Lower Core Support Assembly (NA)

5.5.9 Sediment Transfer and Processing Operations (NA)

5.5.10 Pressurizer Spray Line Defueling System (NA)

5.5.11 Decontamination Using Ultrahigh Pressure Water Flush (NA)

5.6 Evaluations for Defueling Operations

5.6.1 Preliminary Defueling (NA)

5.6.2 Early Defueling

- ***Purpose***. To remove end fittings, structural materials, and related loose debris that would not involve removal of significant amounts of fuel material, to remove intact segments of fuel rods and other pieces of core debris, and to remove loose fuel fines (particles) by vacuuming operations.

- ***Evaluation: Pyrophoricity***. [49] The licensee's safety evaluation noted that the NRC had previously evaluated the potential for zirconium hydride fires in the PEIS. This evaluation assumed that operations relating to early defueling would be conducted with water coverage. The PEIS concluded that zirconium hydride would not ignite underwater. Since early defueling operations would be conducted underwater, the licensee concurred that there was no potential for a pyrophoric zirconium reaction in the reactor vessel during early defueling.

  - *Pyrophoric Reaction in Canisters*. After the defueling canisters were loaded with debris, the dewatering operation in the fuel handling building would remove enough water from the canister to ensure that more than one-half of the catalytic recombiner was not submerged. When the catalyst was no longer submerged, the catalyst would function to control the hydrogen and oxygen concentrations in the canister. An inert cover gas (e.g., argon) would be used to blanket the core debris in the canister after the dewatering process.

    The concern over pyrophoric materials was focused on the pyrophoric potential of metallic Zircaloy and zirconium hydride fines in the dewatered canisters. The manner in which the fuel deteriorated during the accident made the presence of these species in a pyrophoric form highly unlikely in the known configuration of the core rubble bed. Zircaloy, a ductile metal even after irradiation, would not break into small particles in the high-temperature steam environment of the TMI-2 accident. Rather, as the material oxidized, the oxide would break up as a consequence of thermal shock or abrasion. However, during the early defueling process, it was possible that fresh (i.e., unoxidized) metal surfaces, including small chips and fines, could be created as a result of cutting operations.

- *Conditions for Pyrophoric Reaction*. Many analyses had been conducted since pyrophoric concern was initially raised, and these were summarized in the licensee's technical plan TPO/TMI-127, "Technical Plan for Pyrophoricity," December 1984. Analyses indicated that three conditions were required to initiate and maintain a pyrophoric reaction:

  (1) The pyrophoric material required a high surface-to-volume ratio, to the same degree as powder. Experience indicated that moist zirconium fines of less than 10 microns would burn. However, analysis of core debris had indicated that only about 1.5 percent of the particulate matter was smaller than 50 microns. [g] The early defueling activities were not likely to generate significant quantities of fines in the size range of concern.

  (2) The pyrophoric material required an oxygen-depleted environment, followed by a sudden exposure to oxygen. The surface of the core pyrophoric material had been exposed to oxygen in the water since the accident. Thus, oxidation that had already occurred would limit a pyrophoric reaction to freshly exposed material. The early defueling process was not likely to expose significant quantities of debris in the size range specified above. Further, pyrophoric material in this size range subject to exposure during the early defueling activities was initially underwater, where oxidation was also possible. The rate of this oxidation had been determined to be extremely high; thus, the resulting time for all newly created surfaces of a reactive metal to be oxidized would be small when compared to the time required for canister loading and transfer operations.

  (3) The oxidation rate was required to exceed the heat transfer rate to the surrounding environment. The oxidized debris that mixed with any pyrophoric material would act as a diluent and minimize the potential for ignition and propagation.

- *Test Results*. In addition to the above considerations, tests had been previously conducted on a sample of material that was removed from the plenum to determine its pyrophoricity. Attempts were made to pilot ignite the subject material by conducting a spark test and a flame test. The results found no pyrophoric characteristic for the material tested.

- *Conclusion*. In summary, theoretical analysis and experimental data indicated that the characteristics of the material in the reactor vessel at the time of this evaluation, or as potentially modified during early defueling activities or postulated accidents, were highly unlikely to sustain a pyrophoric reaction. This conclusion was not dependent on continued submergence of the material in water. Therefore, it was not considered reasonable to postulate a pyrophoric reaction of exposed fuel debris as a significant driving force for radionuclide transport.

----

[g] Editor's Note: The licensee's safety evaluation report for early defueling referred to 50-micron particles, whereas the safety evaluation for the recovery seismic design criteria stated 45 microns. Both evaluations referenced the same licensee technical plan for pyrophoricity, which is not available for review.

- ***NRC Review: Pyrophoricity***. [50] Based on tests conducted on core debris samples and the experience of earlier cleanup activities, previous NRC safety evaluations had concluded that the potential for submerged core debris to sustain a pyrophoric reaction was extremely remote. During early defueling activities, core debris collected in canisters could be exposed to gases following dewatering. Argon, an inert gas, would be used in the dewatering process to purge excess water and act as a cover gas in the canister. Some hydrogen and oxygen were likely to be generated in the canisters because of radiolytic decomposition of water.

Although some debris could be exposed to oxygen, the potential for a pyrophoric reaction was still very small for the following reasons: (●) Significant quantities of potentially pyrophoric material (zirconium hydride) were not postulated to exist in sizes small enough (10 microns) to spontaneously ignite. (●) Unoxidized surfaces had to be exposed to an oxygen environment to undergo a pyrophoric reaction and any new surfaces exposed in the course of defueling would be in contact with water, thus oxidizing before canister dewatering occurred. (●) The rate of oxidation had to exceed the heat transfer rate of the material for ignition to occur.

The NRC concluded that the potential for a pyrophoric event during early defueling activities was extremely unlikely. If a pyrophoric event were to occur in a canister, there was a high probability that the canister would be submerged, providing additional protection against the consequences of such an event.

5.6.3 Storage of Upper End Fittings in an Array of 55 Gallon Drums

- ***Purpose***. To use shielded 55-gallon drums to store end fittings and other structural material removed from the reactor vessel.

- ***Evaluation: Pyrophoricity***. [51, 52] The licensee determined that the pyrophoricity concerns and radioactive material releases due to a dropped storage container were bounded by the analysis performed for a dropped fuel canister in the licensee's safety evaluation report [53] for bulk defueling.

- ***NRC Review***. [54, 55] Editor's Note: The NRC's safety evaluation report did not specifically address this evaluation topic.

5.6.4 Defueling (Also Known as "Bulk" Defueling)

- ***Purpose***. To develop defueling tools and to conduct activities necessary to remove the remaining fuel and structural debris located in the original core volume. An additional activity was to address vacuuming of debris from the lower head of the vessel, a primary activity associated with lower head defueling.

- ***Evaluation: Pyrophoricity***. [56, 57] Editor's Note: The licensee's safety evaluation report for early defueling was practically identical to its subsequent safety evaluation report [58] for bulk defueling; therefore, this section does not include the evaluation text. Please refer to the next section for details.

- ***NRC Review: Pyrophoricity***. [59] The NRC's safety evaluation noted that bulk defueling activities were to include considerable sizing and cutting operations, resulting in the creation of smaller particles of core debris, including zirconium compounds. Despite the generation of more finely divided particles, the potential for a pyrophoric reaction remained very low. During pilot-ignition tests conducted on debris samples and during previous defueling operations, no pyrophoric characteristics were observed. Although creation of smaller particles was expected, the increase in the volume of particles in the range of concern, below 50 microns, was not expected to be great. Bulk defueling activities would be conducted underwater; as discussed in the NRC's safety evaluations [60, 61] for early defueling and core stratification sample acquisition, this condition effectively prevented pyrophoric events, even if finely divided zirconium compounds were present. In the safety evaluation of early defueling, the NRC also concluded that the potential for a pyrophoric event in a filled, dewatered defueling canister was acceptably low.

Canister loading, handling, and storage would be conducted in a similar manner during bulk defueling; therefore, the NRC's previous conclusion was applicable. In the safety evaluation of core stratification sample acquisition, the NRC concluded that the use of the core bore machine would not present an unacceptable potential for a pyrophoric event, provided that the drill bit was continuously flushed to rapidly remove the heat generated by drilling. Because the drill unit was designed to shut down upon loss of flush water, removal of the frictional heat generated by drilling was ensured. The NRC concluded that the potential for a pyrophoric event was acceptably low for the proposed core region defueling activities.

5.6.5 Use of Core Bore Machine for Bulk Defueling

- ***Purpose***. To use the core stratification sample acquisition (core bore) tooling as a defueling tool so that other defueling tools could more effectively break up and remove the remaining core debris. The core bore tool used a solid-faced bit to perforate the hard crust region of the core, down to the lower grid support structure, at multiple locations. The defueling work platform orientation system was used to position the drill mechanism with restrictions.

- ***Evaluation: Pyrophoricity***. [62] Pyrophoricity issues were bounded by the licensee's safety evaluation report (SER) [63] for core stratification sample acquisition. Heat generated by the drill bit would be readily dissipated by flush water from the boring tool during drilling operations. Previous operations used a drill bit with a surface area of about 4.7 square inches. The proposed operation would use a bit with a surface area of about 9.62 square inches. Both bits had an outside diameter of about 3.5 inches. Although flush waterflow during both operations was the same (i.e., 3 to 6 gallons per minute), this flow was determined to be sufficient to remove any heat generated during defueling operations with the core bore tool.

- ***NRC Review: Pyrophoricity***. [64, 65] The NRC's safety evaluation imposed a requirement to maintain continuous drill bit cooling; this ensured that pyrophoricity issues would remain within the bounds previously reported in the NRC's SER [66] for core stratification sample acquisition and its SER [67] for bulk defueling. The previous requirement [68] to maintain an operable water level alarm was retained to ensure prompt leak detection in case one of the in-core instrument nozzle welds was breached. Within these stated limitations, safety issues associated with pyrophoricity did not differ from those issues associated with previously reviewed and approved operations.

5.6.6 Lower Core Support Assembly Defueling

- ***Purpose***. To dismantle and defuel the lower core support assembly (LCSA) and to partially defuel the lower reactor vessel head. Structural material removed from the LCSA included the: (●) lower grid rib assembly; (●) lower grid forging; (●) distribution plate; and (●) in-core guide support plate. The lowest elliptical flow distributor and the gusseted in-core guide tubes would not be removed under this proposed activity. The use of the core bore machine, plasma arc cutting equipment, cavitating water jet, robot manipulators, automatic cutting equipment system, and other previously approved tools and equipment was included in this activity and associated safety evaluations.

- ***Evaluation: Pyrophoricity***. [69] The licensee determined that pyrophoricity concerns during LCSA and lower head defueling were bounded by the licensee's safety evaluations for early [70] and bulk defueling [71] of the core region and its technical plan (TPO/TMI-127) for pyrophoricity.

- ***NRC Review***. [72] Editor's Note: The NRC's safety evaluation report did not specifically address this evaluation topic.

5.6.7 Completion of Lower Core Support Assembly and Lower Head Defueling

- ***Purpose***. To remove the elliptical flow distributor and gusseted in-core guide tubes of the lower core support assembly and to defuel the reactor vessel lower head.

- ***Evaluation: Pyrophoricity***. [73, 74] The licensee determined that pyrophoricity concerns during lower core support assembly and lower head defueling were bounded by the licensee's safety evaluations [75] for early and bulk defueling of the core region and the technical plan (TPO/TMI-127) for pyrophoricity.

- ***NRC Review***. [76] Editor's Note: The NRC's safety evaluation report did not specifically address this evaluation topic.

5.6.8 Upper Core Support Assembly Defueling

- ***Purpose***. To cut and move the baffle plates and to defuel the upper core support assembly. This evaluation addressed the following activities: (●) cutting the baffle plates for later removal; (●) removing bolts from the baffle plates; (●) removing the baffle plates; and (●) removing core debris from the baffle plates and core former plates.

- ***Evaluation: Pyrophoricity***. [77] The licensee determined that pyrophoricity concerns during upper core support assembly defueling were bounded by the licensee's safety evaluations [78] for early and bulk defueling of the core region and the technical plan (TPO/TMI-127) for pyrophoricity.

- ***NRC Review: Pyrophoricity***. [79] The NRC's safety evaluation concluded that the remaining fuel was well characterized, including its location. The portion of the core debris most likely to contain pyrophoric materials was removed while the safety controls remained.

5.7 Evaluations for Waste Management (NA)

5.7.1 EPICOR II (NA)

5.7.2 Submerged Demineralizer System (NA)

5.7.2.1 Submerged Demineralizer System Operations (NA)

5.7.2.2 Submerged Demineralizer System Liner Recombiner and Vacuum Outgassing System (NA)

5.8 Other Safety Topics

This section covers other activities relating specifically to pyrophoricity safety that were unique at TMI-2.

5.8.1 Risks to Safe Operation of TMI Unit 1 Resulting from TMI-2 Cleanup

- ***Purpose***. To examine the possible impacts of various categories of events on the integrity of the physical barriers to radioactivity release at TMI-1. These events included fires, explosions, missiles, the release of toxic chemicals, and the release of radioactive materials from TMI-2.

- ***Evaluation: Pyrophoricity***. [80] To hypothesize an event that could lead to excessive releases of radioactive material, the licensee's safety evaluation determined that events directly affecting radioactive materials in relatively large quantities (greater than 1000 curies) had to be considered. At the time of this assessment, most radioactive material was contained within the damaged fuel in the reactor vessel. However, defueling activities were going to transfer this material to the spent fuel pool and ultimately, using shipping casks, to offsite storage. The two basic categories of accidents that could result in release of radionuclides from damaged fuel material were (●) severe overheating due to insufficient cooling, fires, or mechanical damage and (●) criticality. In addition to core heatup resulting from decay heat, fires resulting from zirconium and zirconium hydride ignition were investigated. Investigations had already been performed as documented in the NRC's PEIS [81] and the licensee's data report TPO/TMI-120, "TMI-2 Pyrophoricity Studies," dated June 1984.

- *Results*. The results were summarized as follows: (●) Analysis of TMI-2 core material samples did not show a pyrophoric reaction. (●) Only finely divided zirconium hydride, in powder form, would be pyrophoric when exposed to air (oxygen). (●) The presence of hydrided Zircaloy cladding in a powdered state would be readily identified by visual inspection, and precautions could be taken (samples indicated that this condition did not exist). (●) Defueling operations would be performed underwater (zirconium would not ignite underwater). (●) Realistic particle sizes would not ignite until temperatures exceeded 1000 degrees Fahrenheit.

- *Conclusion*. This information confirmed the low likelihood of approaching the level of a major zirconium hydride ignition for the following reasons: (●) Defueling activities would be performed underwater and ignition would not occur in water. (●) The likelihood of the accidental loss of water cover was extremely low, in either the reactor vessel or spent fuel pool. (●) Even if water was not present, the analysis indicated that fuel temperatures could not attain the values needed for realistic particle sizes to ignite, even if an ignition source were present. The sampling that had been performed indicated the material to be nonpyrophoric.

  Even assuming ignition of the zirconium material, liquefaction of the fuel material could not occur unless the majority of the zirconium material was involved in the reaction. Furthermore, the amount of unreacted zirconium material present was less than in a typical reactor because of the TMI-2 accident.

5.8.2 Seismic Design Requirement of TMI-2 Postaccident Systems

- ***Purpose***. To demonstrate that the systems installed after the 1979 accident, or that were contemplated for later installation for the sole purpose of supporting recovery activities, were not required to meet seismic design requirements.

- ***Evaluation: Pyrophoricity***. [82] The licensee's analyses of postaccident systems and processes considered the consequences of accidents postulated to occur because of a seismic event. The evaluation was conducted in three steps: (1) identification of source terms and driving forces, (2) specification of seismic-induced accident sequences, and (3) consequence analysis.

Potential driving forces that could develop from a seismic event or could credibly occur coincidentally with the event included: (●) pyrophoric fire-related overpressure; (●) combustibles fire-related overpressure; (●) atmospheric differential pressure/air exchange (e.g., open equipment hatch or open buildings); or (●) reactor coolant system steaming due to criticality or reactor vessel draindown. Criticality was a possible result of a postulated core reconfiguration or a boron dilution event. Seismic-induced pyrophoric fire-related overpressure scenarios are presented below. Chapter 11, Section 2, of this NUREG/KM summarizes the licensee safety evaluation report.

- *Pyrophoric fire-related overpressure*. Fire resulting from a pyrophoric reaction was postulated to be caused by metal fines associated with the core debris or debris that had settled on reactor vessel internals. Three possible materials were present in the core that could have undergone a pyrophoric reaction: a uranium-zirconium eutectoid (that was surrounded by an oxygen-stabilized zirconium), zirconium metal, and zirconium hydride. Only three scenarios exposing core materials were postulated: (●) uncovering the core by leakage through the guide tubes, (●) leaking of core materials through failed guide tubes, or (●) exposing materials in a defueling canister because of a handling accident.

  Considerable analysis had been previously conducted after the pyrophoric concern was raised. The results of these analyses were summarized in a presentation to the TMI-2 Technical Assistance and Advisory Group. The analyses indicated that three conditions were required to initiate and maintain a pyrophoric reaction:

  (1) The pyrophoric material required a high surface-to-volume ratio, to the same degree as powder. Experience indicated that moist zirconium fines smaller than 10 microns would burn. However, analysis of core debris had indicated that only about 1.5 percent of the particulate matter was less than 50 microns. [(h)] The early defueling activities were not likely to generate significant quantities of fines in the size range of concern.

  (2) The pyrophoric material required an oxygen-depleted environment, followed by a sudden exposure to oxygen. The surface of the core pyrophoric material had been exposed to oxygen in the water since the accident. Thus, oxidation that had already occurred would limit a pyrophoric reaction to freshly exposed material. The early defueling process was not likely to expose significant quantities of debris in the size range specified above. Further, pyrophoric material in this size range subject to exposure during the early defueling was initially underwater, where oxidation was also possible. The rate of this oxidation had been determined to be extremely high; thus, the resulting time for all newly created surfaces of a reactive metal to be oxidized would be small when compared to the time required for canister loading and transfer operations.

  (3) The oxidation rate was required to exceed the heat transfer rate to the surrounding environment. The oxidized debris that mixed with any pyrophoric material would act as a diluent and minimize the potential for ignition and propagation.

[h] Editor's Note: The licensee's safety evaluation report for early defueling referred to 50-micron particles, whereas the safety evaluation for the recovery seismic design criteria stated 45 microns. Both evaluations referenced the same licensee technical plan for pyrophoricity, which is not available for review.

- *Conclusion (Pyrophoric)*. Theoretical analysis and experimental data indicated that the characteristics of the materials in the reactor vessel, which may have been modified during defueling or postulated accidents, were exceedingly unlikely to sustain a pyrophoric reaction. This conclusion was not dependent on continued submergence of the material in water or on any of the seismically induced accident sequences that had been postulated. Therefore, it was not considered reasonable for a postulated pyrophoric reaction from exposed fuel debris to be a significant driving force for radionuclide transport.

- ***NRC Review***. (83) Editor's Note: The NRC's safety evaluation did not specifically mention pyrophoricity hazards.

5.8.3 Shipping Cask Model 125-B

- ***Purpose***. To develop a shipping cask [i] for the defueling canisters for fuel debris shipments from TMI-2 to the INEL. Six defueling canisters could be loaded into each of the three NRC--certified shipping casks.

- ***Evaluation: Pyrophoricity***. (84) The licensee's special hazards evaluation (85) performed for the TMI-2 core debris shipments considered pyrophoricity of the canisters during shipment. The DOE's Hanford Operations was responsible for the effort. Pyrophoricity was a minor technical concern for the TMI-2 core debris shipments. The concern was due to the possible presence of small particles of zirconium from damage to the Zircaloy-clad fuel pins in the core. Finely divided zirconium was known to spontaneously ignite under certain conditions. However, this was also a concern for defueling of the reactor, and the considerable testing performed determined that the zirconium in the TMI-2 core debris was not pyrophoric, largely because of oxidation. Furthermore, the environment of the core debris during the shipments essentially eliminated the potential for pyrophoricity because of the use of an inert cover gas in the canisters. Limiting the choice of cover gas to only argon, helium, or nitrogen, rather than air, reduced the potential for a pyrophoric event.

- *Pyrophoric Conditions*. The determination that the fuel debris did not have a significant potential for a pyrophoric event during transport and storage was based on a number of considerations. The existence of potentially pyrophoric zirconium and zirconium hydride in the TMI-2 core debris could not be completely ruled out. However, laboratory tests and studies of the TMI-2 core debris had shown that conditions that could initiate or sustain a pyrophoric event were extremely unlikely. For a metallic material to undergo a pyrophoric reaction, the following conditions must be present: (●) Material must be in a chemical form or state that could undergo oxidation. (●) Material had to have a large surface-to-volume ratio, with particle diameters of less than 50 microns. (●) Material was required to be of a sufficient concentration to initiate, sustain, and propagate the ignition. (●) Material must be

[i] Editor's Note: while large shipping containers of radioactive materials may often be referred to as "shipping casks," the proper term for such containers, when loaded with contents and in their transportation configuration, is "package." See section 10 CFR 71.4.

exposed to an oxidizing environment where the heat generation rate was much greater than the heat loss rate.

- *Test Results*. Laboratory tests of TMI-2 core debris samples were performed before removal of the reactor plenum and the start of defueling. In these tests, rigorous attempts were made to ignite the samples. Samples were also mechanically agitated in an attempt to disturb the protective oxide film on the Zircaloy metal to expose a fresh, unoxidized surface. All tests failed to produce a pyrophoric reaction. These tests concluded that the loose core rubble presented little potential for the existence of pyrophoric conditions. The evaluation stated that aggressive defueling techniques, such as drilling, cutting, or sawing, could expose fresh zirconium surfaces. These surfaces could lead to conditions that might support a pyrophoric reaction; therefore, the issue was studied further.

  For the aggressive defueling techniques, freshly exposed unoxidized zirconium surfaces were believed to quickly oxidize. This would result in a protective oxide layer over any newly exposed surfaces because of the high dissolved oxygen content and pH of the reactor coolant. Further, the dilution of the fresh material with nonpyrophoric material was considered to prevent heat buildup sufficient to cause ignition.

- *Conclusion*. The applicant's safety analysis determined that the potential for a pyrophoric event during shipment of the TMI-2 core debris was insignificant. This was based on the following conclusions of several studies: (●) There was little potential for the existence of pyrophoric material in the TMI-2 core debris. (●) As fresh surfaces were exposed during defueling, they would probably oxidize in the reactor coolant system before being exposed to the atmosphere. (●) If potentially pyrophoric surfaces were exposed and did not oxidize, they would probably be in the form of coarse turnings, chips, or shavings that would not present significant pyrophoric potential. (●) Material would be significantly diluted with inert, nonpyrophoric material. (●) Because the mass of material present provided a heat sink, there was very little potential for the material to react and retain sufficient heat to cause ignition. (●) Fuel debris would be shipped in an inert gas environment that would not support a pyrophoric reaction. (●) Even under hypothetical accident conditions during transportation, the maximum temperatures experienced by the fuel debris would not be sufficient to cause ignition of the diluted potentially pyrophoric material.

• ***NRC Review***. (86) No evaluation specific to pyrophoricity concerns was addressed in the NRC's safety evaluation report [j] for the shipping cask Model 125-B certificate of compliance.

---

[j] Editor's Note: While the title of the reference mentions the certificate of compliance, the reference also includes the NRC's safety evaluation report for the package approval and issuance of the certificate.

5.8.4 Frequently Asked Questions about Cleanup Activities at TMI-2 (NUREG-0732, Rev. 1, March 1984) [87, 88]

This question-and-answer report provided answers in nontechnical language to frequently asked questions about the status of cleanup activities at TMI-2.

- **Question: Were explosions possible during the cleanup?**

Answer: There was virtually no possibility of an explosion. Of course, there was always the potential that a hose under high pressure could burst or that cleaning solvents could ignite and injure workers in the immediate vicinity. The licensee enforced a strict program to minimize such a possibility. Also, there had been concern about the possibility of a pyrophoric explosion.

- **Question: What are pyrophoric explosions?**

Answer: Pyrophoric explosions result from the extremely rapid burning of very reactive metals. For example, metallic sodium undergoes pyrophoric burning (or explosion) if wet. Less reactive metals, such as aluminum, magnesium, and zirconium, would undergo pyrophoric reactions if they are finely powdered and exposed to air. Zirconium metal and oxide from the tubes that surround the uranium fuel could undergo pyrophoric reactions if exposed to air, although these materials were underwater and would remain underwater throughout the cleanup. Wet particles could undergo pyrophoric reactions when exposed to air, but such reactions would not take place underwater. Even so, this possibility was investigated for the materials at TMI-2. Samples of reactive metals taken from the structures near the top of the reactor vessel were tested. Based on the results of these investigations and tests, such explosions were considered highly unlikely. Nevertheless, workers were to perform fuel removal tasks underwater to avoid any chance of such reactions.

5.9 Endnotes

Reference citations are exact filenames of documents on the Digital Versatile Discs (DVDs) to NUREG/KM-0001, Supplement 1, [89] "Three Mile Island Accident of 1979 Knowledge Management Digest: Recovery and Cleanup," issued June 2016, DVD document filenames start with a full date (YYY-MM-DD) or end with a partial date (YYYY-DD). Citations for references that are not on a DVD begin with the author organization or name(s), with the document title in quotation marks. Contents of the DVDs are available at the Idaho National Laboratory Web site (https://tmi2kml.inl.gov).

---

[1] NUREG-0683, Vol. 1, PEIS-Decontamination and Disposal of Radioactive Wastes Resulting from TMI-2 (1981-03)
[2] GEND-051, Evaluation of Special Safety Issues Associated with Handling the TMI-2 Core Debris (1985-06)
[3] DOE-ID-10276, TMI-2 Lessons Learned by the DOE 1979-1990 (1990-03)
[4] NUREG-0683, Vol. 1, PEIS-Decontamination and Disposal of Radioactive Wastes Resulting from TMI-2 (1981-03)

[5] GEND-INF-044, TMI-2 LEADSCREW DEBRIS PYROPHORICITY STUDY (1984-04)
[6] GEND-043, TMI-2 PYROPHORICITY STUDIES (1984-11)
[7] GEND-INF-044, TMI-2 LEADSCREW DEBRIS PYROPHORICITY STUDY (1984-04)
[8] (1984-02-17) NRC, Directors Decision Under 10 CFR 2.206 Regarding Pyrophoric Reaction of Zirconium Fines
[9] GEND-051, Evaluation of Special Safety Issues Associated with Handling the TMI-2 Core Debris (1985-06)
[10] GEND-INF-023-VOL-4, ANALYSIS OF THE THREE MILE ISLAND UNIT 2 HYDROGEN BURN (1983-03)
[11] GEND-INF-027, PROPOSED METHODS FOR DEFUELING THE TMI-2 REACTOR CORE (1984-05)
[12] Cooper, T.D., "Review of Zirconium-Zircaloy Pyrophoricity," RHO-RE-ST-31-P, Rockwell Hanford Operations, Richland, WA, November 1984
[13] Cooper, T. D., "Review of Zirconium-Zircaloy Pyrophoricity," RHO-RE-ST-31-P, Rockwell Hanford Operations, Richland, WA, November 1984
[14] GEND-INF-060-VOL-1, TMI-2 CORE DEBRIS GRAB SAMPLES ANALYSIS OF FIRST GROUP OF SAMPLES (1985-07)
[15] GEND-INF-075-PT-1, TMI-2 CORE DEBRIS GRAB SAMPLE EXAMINATION AND RESULTS (1986-09)
[16] GEND-043, TMI-2 PYROPHORICITY STUDIES (1984-11)
[17] GEND-INF-075-PT-2, TMI-2 CORE DEBRIS GRAB SAMPLE SAMPLES EXAMINATION (1986-09)
[18] (1983-05-25) GPU Safety Analysis, Radiation Characterization Under the Reactor Vessel Head, Rev. 0
[19] (1983-05-25) GPU Safety Analysis, Radiation Characterization Under the Reactor Vessel Head, Rev. 0
[20] (1983-05-25) GPU Safety Analysis, Radiation Characterization Under the Reactor Vessel Head, Rev. 0
[21] (1983-07-13) NRC Review, Reactor Vessel Underhead Characterization (re Various Letters)
[22] (1983-07-20) GPU Safety Evaluation, Underhead Characterization, Core Sampling Addendum
[23] USNRC, "Three Mile Island Accident of 1979 Knowledge Management Digest: Recovery and Cleanup," NUREG/KM-0001, Supplement 1, June 2016 [Available at nrc.gov]
[24] (1983-07-20) GPU Safety Evaluation, Underhead Characterization, Core Sampling Addendum
[25] (1983-05-25) GPU Safety Analysis, Radiation Characterization Under the Reactor Vessel Head, Rev. 0
[26] (1983-08-19) NRC Safety Evaluation, Addendum to the Underhead Characterization Study (re 07-30-1983)
[27] (1983-05-25) GPU Safety Analysis, Radiation Characterization Under the Reactor Vessel Head, Rev. 0
[28] (1983-07-20) GPU Safety Evaluation, Underhead Characterization, Core Sampling Addendum
[29] USNRC, "Formation of Zirconium Hydrides in the TMI-2 Incident," Internal Memorandum, June 6, 1979 [ADAMS Accession No. ML19224C127]
[30] Gulbransen, Earl K. and K.F. Andrew, "Reaction of Hydrogen with Pre-Oxidized Zircaloy-2 at 300° to 400°C," *Journal of the Electrochemical Society* 104, 709–712 (1957)
[31] Gulbransen, Earl K. and K.F. Andrew, "Reaction of Hydrogen with Pre-Oxidized Zircaloy-2 at 300° to 400°C," *Journal of the Electrochemical Society* 104, 709–712 (1957)

[32] (1986-06-11) GPU Safety Evaluation, Core Stratification Sample Acquisition, Rev. 4
[33] (1985-04-16) GPU Safety Evaluation, Justifying Non-Seismic Design of TMI-2 Post-Accident Systems, Rev. 0
[34] (1986-05-05) NRC Safety Evaluation, Core Stratification Sample Acquisition SER, Rev. 1 (re 08-30-1985, 12-31-1985)
[35] (1983-08-19) NRC Safety Evaluation, Addendum to the Underhead Characterization Study (re 07-30-1983)
[36] (1989-08-18) GPU Safety Evaluation, Remove Metallurgical Samples from Reactor Vessel SER
[37] (1989-10-20) GPU Safety Evaluation, Remove Metallurgical Samples from Vessel, Rev. 1 (re 08-18-1989) (effective pages)
[38] (1986-05-15) GPU Safety Evaluation, Defueling SER, Rev. 10
[39] (1989-11-28) NRC Safety Evaluation, Reactor Vessel Lower Head Metallurgical Sampling (re 08-18-1989)
[40] (1984-03-09) GPU Safety Evaluation, Head Removal, Rev. 5
[41] (1983-05-25) GPU Safety Analysis, Radiation Characterization Under the Reactor Vessel Head, Rev. 0
[42] (1984-07-17) NRC Safety Evaluation, Reactor Pressure Vessel Head Lift (re -3-09-1984)
[43] (1984-02-17) NRC, Directors Decision Under 10 CFR 2.206 Regarding Pyrophoric Reaction of Zirconium Fines
[44] (1986-03-26) GPU, Plan to Use Hydraulic Shredder to Reduce Core Debris
[45] (1986-06-11) GPU Safety Evaluation, Core Stratification Sample Acquisition, Rev. 4
[46] (1986-04-16) NRC Review, Use of Hydraulic Shredder
[47] (1986-08-27) GPU, Plan to Use Plasma Arc Torch to Cut Upper End Fittings (re 08-18-1986)
[48] (1987-08-20) NRC Review, Use of Plasma Arc Torch to Cut Upper End Fittings (re various letters)
[49] (1985-10-10) GPU Safety Evaluation, Early Defueling of the TMI-2 Reactor Vessel, Rev. 4
[50] (1985-11-12) NRC Safety Evaluation, Early Defueling of the TMI-2 Reactor Vessel (10-10-1985)
[51] (1986-08-16) GPU, Storage of Upper End Fittings in Shielded 55 Gal Drums
[52] (1986-09-09) GPU Response to NRC (Discussion), Storage of Upper End Fittings (re 08-16-1986)
[53] (1986-05-15) GPU Safety Evaluation, Defueling SER, Rev. 10
[54] (1986-09-10) NRC Review, Plans to Store Upper End Fittings (re various letters)
[55] (1986-09-25) NRC Safety Evaluation, Storage of Upper End Fittings in an Array of 55 Gallon Drums (re various letters)
[56] (1986-05-15) GPU Safety Evaluation, Defueling SER, Rev. 10
[57] (1986-05-20) GPU Safety Evaluation, Defueling SER, Rev. 10 (Correction)
[58] (1985-10-10) GPU Safety Evaluation, Early Defueling of the TMI-2 Reactor Vessel, Rev. 4
[59] (1986-07-24) NRC Review, Core Region Bulk Defueling Operations Limited to the Core Region (re 05-15, 07-23-1986)
[60] (1985-11-12) NRC Safety Evaluation, Early Defueling of the TMI-2 Reactor Vessel (10-10-1985)
[61] (1986-05-05) NRC Safety Evaluation, Core Stratification Sample Acquisition SER, Rev. 1 (re 08-30-1985, 12-31-1985)
[62] (1986-07-23) GPU, Plan to Use Core Bore Tooling as Defueling Tool
[63] (1986-06-11) GPU Safety Evaluation, Core Stratification Sample Acquisition, Rev. 4
[64] (1986-07-24) NRC Review, Core Region Bulk Defueling Operations Limited to the Core Region (re 05-15, 07-23-1986)

[65] (1986-10-16) NRC Safety Evaluation, Core Bore Operations (re 09-19-1986)
[66] (1986-05-05) NRC Safety Evaluation, Core Stratification Sample Acquisition SER, Rev. 1 (re 08-30-1985, 12-31-1985)
[67] (1986-07-24) NRC Review, Core Region Bulk Defueling Operations Limited to the Core Region (re 05-15, 07-23-1986)
[68] (1986-07-16) NRC Safety Evaluation, Extended Core Stratification Sample Acquisition SER (re 06-23-1986)
[69] (1988-01-18) GPU Safety Evaluation, Lower Core Support Assembly Defueling, Rev. 2
[70] (1985-10-10) GPU Safety Evaluation, Early Defueling of the TMI-2 Reactor Vessel, Rev. 4
[71] (1986-05-15) GPU Safety Evaluation, Defueling SER, Rev. 10
[72] (1988-04-01) NRC Safety Evaluation, Lower Core Support Assembly Defueling (re 01-18-1988)
[73] (1988-06-06) GPU Safety Evaluation, Completion of Lower Core Support and Lower Head Defueling, Rev. 0
[74] (1988-06-27) GPU Safety Evaluation, Completion of Lower Core Support and Lower Head Defueling, Rev. 1
[75] (1986-05-15) GPU Safety Evaluation, Defueling SER, Rev. 10
[76] (1988-12-01) NRC Safety Evaluation, Lower Core Support and Lower Head Defueling (re 06-06-1988)
[77] (1988-09-15) GPU Safety Evaluation, Completion of Upper Support Core Assembly Defueling, Rev. 0
[78] (1986-05-15) GPU Safety Evaluation, Defueling SER, Rev. 10
[79] (1989-04-04) NRC Safety Evaluation, Completion of Upper Core Support Assembly Defueling (re 9-15-1988)
[80] (1984-09-11) GPU, Assessment of Risks to Safe Operation of TMI-1 Resulting from TMI-2 and Its Cleanup (Report)
[81] NUREG-0683, Vol. 1, PEIS-Decontamination and Disposal of Radioactive Wastes Resulting from TMI-2 (1981-03)
[82] (1985-04-16) GPU Safety Evaluation, Justifying Non-Seismic Design of TMI-2 Post-Accident Systems, Rev. 0
[83] (1986-01-16) NRC Review, Seismic Design Requirements to Post Accident Systems (re 04-16, 07-29, 09-27-1985)
[84] DOE-ID-10400, Historical Summary of the TMI-2 Core Debris Transportation Campaign (1993-03)
[85] GEND-051, Evaluation of Special Safety Issues Associated with Handling the TMI-2 Core Debris (1985-06)
[86] (1986-04-11) NRC, Certificate of Compliance 9200, Rev. 0, for Model 125-B Shipping Cask
[87] NUREG-0732, Rev. 1, Answers to Frequently Asked Questions about Cleanup Activities at TMI-2 (1984-03)
[88] NUREG-1060, Answers to Questions About Updated Estimates of Occupational Doses at TMI-2 (1983-12)
[89] USNRC, “Three Mile Island Accident of 1979 Knowledge Management Digest: Recovery and Cleanup,” NUREG/KM-0001, Supplement 1, June 2016 [Available at nrc.gov]

6 DECAY HEAT REMOVAL SAFETY EVALUATIONS

TABLE OF CONTENTS

Note: “NA” (not applicable) means the licensee and NRC determined that the safety topic was not important for the activity.

6.1 Introduction

6.1.1 Background

The safety topic of decay heat removal was a concern early in the defueling operations. This chapter focuses on the safety evaluations of decay heat generation and removal during postaccident TMI-2 cleanup.

The feasibility of never-before-tested, long-term natural circulation was extensively studied [1, 2, 3, 4, 5] within the first month following the accident. The following year, the licensee and NRC also studied [6, 7, 8] the phenomenon of natural circulation transitioned into a cyclic chugging flow (or "burps"). After 7 months, the NRC tested and approved the transition to loss-to-ambient core cooling (reactor coolant system (RCS) components cooled by ambient containment building atmosphere). This review was documented in a safety evaluation report [9] as part of a recovery technical specification change. The reader should refer to Supplement 1 to this NUREG/KM [10] and accompanying documents for further details of the studies and reviews that supported these modes of decay heat removal, including recovery decay heat removal systems.

Once the preferred RCS decay heat removal mode was identified as loss-to-ambient core cooling, postulated core cooling failure scenarios were limited to draindown of the RCS and breach of the reactor vessel integrity. A draindown event could reduce heat capacity, and a breach could uncover the fuel debris. Given that the decay heat rates were extremely low during defueling, plenty of time would be available to monitor water levels and take appropriate actions. Enhanced heat removal, if required, could be handled by existing systems such as feed and bleed through letdown and the standby pressure control system, refill of the RCS, or mini-decay heat removal system.

6.1.2 Chapter Contents

This chapter presents decay heat removal evaluations of the postaccident TMI-2 and defueling operations. It describes the many studies performed to help the reader understand the thinking of the analysts at the time, the expectations and the reality, uncertainties in the data, and measurement and mitigation methods. It also presents a high-level timeline of cleanup activities.

The evaluations presented in this chapter ensured that all activities that could interrupt the removal of decay heat from fuel debris were addressed and consequences evaluated; controls were maintained in accordance with the requirements of the plant's license, technical specifications, procedures, and applicable regulatory requirements; and adequate contingencies were developed for normal and accident conditions.

Key activities that were initially of concern in decay heat removal included: (●) Quick Look video inspection in the reactor vessel; (●) reactor vessel underhead characterization; (●) core (bore) stratification sample acquisition; (●) reactor vessel head removal operations; (●) defueling operations; and (●) others. As defueling progressed and time passed, decay heat generation rates became less of a safety concern.

Additional evaluations of the loss of decay heat removal scenarios can be found in the chapters of this NUREG/KM supplement on load drop evaluations (Chapter 7) and reactor vessel integrity evaluations (Chapter 8).

Section 2 summarizes the key studies used to support safety evaluations. The following sections present the safety evaluation for each applicable cleanup activity or system. Section 8 lists the endnotes for references cited throughout this chapter.

6.2 Key Studies

The main concern with decay heat removal during defueling and its preparations involved the lowering of reactor coolant level in the reactor coolant system and reactor vessel. The licensee issued two studies for NRC review. The first study supported the Quick Look video inspection inside the reactor vessel in July 1982. The second study was an update that supported the reactor vessel head removal in July 1984. The update took advantage of experiences from the reactor coolant lowering for the Quick Look video inspection in the reactor vessel. Both of these studies are summarized below.

6.2.1 TMI-2 Decay Heat Removal Analysis
(Babcock & Wilcox, April 1982)

This report [a, 11] assessed the thermal status of the reactor core and predicted the thermal response of the system to partial draindown of the reactor coolant system (RCS). The draindown was in preparation of Quick Look video inspection inside the reactor vessel. The criterion used in this study was based on the TMI-2 operating procedure 2102-3.4, Revision 2, "Reactor Coolant System Operation with Core Cooling via Natural Heat Loss," which restricted the average in-core coolant temperature to less than 170 degrees Fahrenheit (degrees F) and all operable thermocouple temperature readings to less than 210 degrees F. This criterion was adopted as a conservative value for the recovery program to maintain a positive margin to boiling. The analysis was directed at evaluating two concerns: (●) system effects of lowering the RCS level and (●) coolant temperatures with lowered water level.

- ***Analysis (Overview)***. The steps used in this evaluation included the following: (1) Estimate the decay heat generation rate. (2) Develop a heat balance model to predict the pre-draindown core heat removal rate and compare to the estimated core heat generation. (3) Evaluate the radial core temperature profile against the criterion for average and maximum in-core thermocouple temperature limits. (4) Predict in-core temperature before partial draindown of the RCS against a range of containment building ambient air temperatures. (5) Predict in-core temperature after partial draindown of the RCS against containment building ambient air temperatures.

[a] Editor's Note: The analysis report was summarized here. The reader should refer to the complete report for additional details.

- ***Models (Overview)***. Three models were used in this evaluation: (●) decay heat generation; (●) system heat capacity; and (●) heat transfer model.

  - *Decay Heat Model*. The decay heat model provided a conservative calculation of core power based on American National Standards Institute/American National Standard 5.1-1979, "Decay Heat Power in Light Water Reactors," [12] standard methodology. The decay heat generation rate was based on 4 years of operation; TMI-2 had about 88 days of full-power operation.

  - *System Heat Capacity Model*. The heat capacity model included those portions of the RCS consistent with the heat transfer model (i.e., only the reactor vessel and the water contained in the vessel). This produced a conservatively small system heat capacity that resulted in a fast RCS heatup.

  - *Heat Transfer Model*. The heat transfer model assumed that heat was transferred only through the reactor vessel walls, lower dome, closure head, and hot legs. To ensure conservative results, this model did not account for any heat transfer through the steam generators or cold legs. Thus, only the ambient air temperature in the containment building was needed to predict RCS bulk water temperatures. The containment building ambient temperatures used in the analysis were from the daily logs. Since only one containment building temperature was recorded, the ambient air temperature was assumed to be constant throughout the containment building. The analysis further assumed: (●) water in the containment building sump was 60 degrees F; (●) both RCS loops contributed to heat transfer; and (●) averaged temperatures for nodes between measured temperatures.

- ***Results (Overview)***. The end results were a set of three curves that presented guidelines for the predicted average in-core temperature after partial draindown of the RCS. These curves included: (●) in-core temperature after RCS draindown without supplemental RCS cooling for upper and lower bounds of containment building ambient air temperatures, (●) additional heat removal after RCS draindown needed to maintain core average temperature at or below 170 degrees F for upper and lower bounds of containment building ambient air temperatures, and (●) containment building air temperature required after RCS draindown to keep in-core temperatures below 170 degrees F. (Refer to Figures 8, 10, and 11 of the report.)

- ***Uncertainties***. The effects of the various conservativisms and uncertainties in the analyses could not be accurately quantified. Rather than place arbitrary uncertainty and conservatism in the analysis method, the recommendations based on these analyses reflected a qualitative judgment of the effect of these uncertainties. For example, conservatisms were introduced into the analyses through the elimination of all heat removal from the RCS from the reactor vessel head, control rod drive mechanisms, and cold-leg piping. Therefore, supplemental heat removal could not actually be required.

Other uncertainties included the following: (●) The decay heat generation rate was based on 4 years of operation; TMI-2 had about 88 days of full-power operation. (●) Overall heat transfer coefficients of RCS components were dominated by the air side convective film coefficients, which were not precisely known. (●) Average coolant temperature for a component was used rather than actual gradient temperatures. (●) Measurement uncertainty in the in-core and loop temperatures was not considered.

- ***Conclusion***. The guidelines (curves) indicated that under certain conditions, the heatup of the reactor coolant could ultimately approach the criterion of 170 degrees F if measures were not taken to improve heat removal. Because of uncertainties identified in the analysis, the report recommended that some options of improved heat removal should be provided in the procedure as a backup measure. Core heat rates indicated that a significant time period (days to weeks) would be available to implement the improved heat removal, if required.

  - *Maximum RCS Temperature*. The maximum thermocouple reading was expected to remain below 210 degrees F, thus maintaining the average thermocouple reading below 170 degrees F. This conclusion was based on the core radial thermocouple map (refer to Figure 6 of the report) and on the expected trend in maximum/average readings with decreasing decay heat production.

  - *Containment Temperature*. Containment building ambient air temperature was shown to significantly affect the cooling requirements in the partially drained configuration (following the guidelines in Figure 11 of the report) and was noted as an effective means of holding average in-core temperatures below 170 degrees F without requiring supplemental heat removal. Given the slow heatup rates, core coolant temperatures could be kept below the desired limit by altering the ambient air temperature, as required.

  - *Corrective Measures*. The use of airflow across the reactor coolant inside the vessel head was reported as an alternative means of providing supplemental heat removal. Such a flowpath could be established between the control rod drive mechanism and the reactor vessel head thermocouple penetrations. Maintaining a flow across the coolant in the partially drained configuration would provide direct convective and evaporative cooling. With the slow heatup rates involved, the adequacy of this cooling mechanism could be verified by monitoring the actual in-core thermocouple temperature response. Supplemental cooling could also be accomplished by adding water to the RCS by means of the decay heat removal system, mini-decay heat removal system, or the standby pressure control system. (The latter two systems were installed after the accident.)

6.2.2 Addendum to TMI-2 Decay Heat Removal Analysis of April 1982

(Babcock & Wilcox, Revision 1, December 1982)

The next step in the recovery process after the Quick Look inspection inside the reactor vessel required the further draindown of the reactor coolant system (RCS) to allow removal of the reactor vessel head. This report [b, 13] presented the results of an analysis performed to determine whether the decay heat loss to the containment building was sufficient to support the

[b] Editor's Note: This NUREG/KM chapter summarizes the analysis report. The reader should refer to the complete report for details of the analysis, which was enclosed in the licensee's safety evaluation report for the reactor vessel head lift.

RCS draindown to head removal level (321.5-foot elevation) without exceeding the temperature criterion. The report concluded that the draindown to reactor head removal level could be accomplished without exceeding the criterion of 170 degrees Fahrenheit (degrees F).

An additional analysis was performed to determine whether the reactor decay heat loss to the containment was sufficient to support the RCS draindown to the bottom of the reactor vessel nozzles (314-foot elevation) without exceeding the temperature criterion. The temperatures predicted in the original April 1982 report used conservative models for the December 1, 1982, and July 1, 1983, draindown dates and exceeded the criterion of 170 degrees F. This was a result of the large degree of conservatism in the decay heat generation, heat transfer, and heat capacity models.

The best estimate models were developed for draindown to the reactor vessel nozzle levels for two cases. The first case included the heat transfer areas and heat capacities of the hot legs and steam generators, and the second case did not include hot legs and steam generators. The reason for developing two best estimate models was the uncertainty of whether the steam generators would be an effective thermal coupling with the core when the cold legs were no longer full. Both best estimate models, however, yielded temperature predictions well below the temperature limit of 170 degrees F.

- ***Analysis (Overview)***. This evaluation included the following steps: (1) Benchmark the original April 1982 models against the RCS water and containment building ambient air temperature data following the draindown for the Quick Look inspection. (2) Evaluate the original models for conservatisms. (3) Predict temperatures for the further RCS draindown to reactor vessel head removal level using the conservative models developed for the April 1982 analysis with slight modifications. (4) Quantify the degree of conservatism in the results reported in the preceding step for best estimate models. (5) Refine original models to reduce conservatisms to produce best estimate models. (6) Analyze best estimate temperatures for RCS draindown to heat removal level. (7) Analyze best estimate and conservative temperatures for RCS draindown to the bottom of the reactor vessel nozzles.

- ***Models (Conservatisms)***. To quantify the degree of conservatism in the conservative models, best estimate models for decay heat generation, system heat capacity, and heat transfer were developed. The Quick Look temperature data were used for benchmarking best estimate type models. Once these models were developed, temperatures resulting from the RCS draindown to reactor vessel head removal level were calculated.

  - *Decay Heat Model*. The American National Standards Institute (ANSI) decay heat prediction method was believed to be very conservative in its treatment of the neutron absorption factor (G factor), which caused high decay heat predictions during the timeframe of interest. The TMI-2 decay heat analysis based on the LOR-2 computer code (the Babcock & Wilcox version of ORIGIN fuel burnup code) was estimated to provide a more realistic prediction or best estimate of the decay heat power levels. Figure 4 of the report showed a comparison of the LOR-2 and ANSI-based decay heat power levels. The decay heat power levels based on LOR-2 were used for best estimate purposes.

- *System Heat Capacity Model*. The system heat capacity was expanded significantly to match the measured Quick Look temperatures. Table 4 of the report showed the physical description of the best estimate system heat capacity. Minor core and reactor vessel internals and hot-leg piping contributions were added to the calculations. The major new contributors were the steam generators and the primary- and secondary-side water contained in them. Only 50 percent of the total available steam generator/water heat capacity was needed to reproduce the measured temperature rate. This magnitude of effective contribution to system heat capacity appeared credible; therefore, this capacity was assumed for best estimate purposes.

- *Heat Transfer Model*. The heat transfer model was expanded to remain consistent with the system heat capacity model. In addition to the reactor vessel, the new heat transfer model included all of the hot legs and the steam generators. Since the hot-leg-to-ambient and steam-generator-to-ambient temperature difference was not known, a factor was determined that could be applied to the core-to-ambient temperature difference in order to estimate the effective hot-leg-to-ambient or steam-generator-to-ambient temperature difference. This factor (0.27) balanced the heat transfer used to produce the measured terminal temperature at the end of the RCS heatup. The constant value coefficients developed in the April 1982 analysis were replaced by air film correlations dependent on temperature difference from the handbook of the American Society of Heating, Refrigerating and Air-Conditioning Engineers. Table 5 of the subject report presented the best estimate heat transfer model.

- *Benchmarking*. The Quick Look draindown temperatures calculated with these best estimate models were compared to the measured heatup temperatures (refer to Figure 5 of the report). The agreement between measured and calculated temperatures was reported to be excellent. These best estimate models were used to simulate the RCS draindown to reactor vessel head removal level.

• ***Applications Models***. The three best estimate models were modified for the following applications: (•) draindown to the head removal level; (•) draindown to the bottom of the reactor vessel nozzles with hot legs and steam generators modeled; and (•) draindown to the bottom of the reactor vessel nozzles without hot legs and steam generators.

- *Draindown to Head Removal Level (Best Estimate)*. Only a few modifications to the best estimate models were needed to reflect draindown to the head removal level. The system heat capacity was decreased both by the lowered water level on the primary side and the assumed complete draining of the steam generator secondary-side water. The total system heat capacity was reduced to 584,132 British thermal units per degree F. The only change to the heat transfer model was the assumption that no heat was transferred through the closure head dome. There were also no changes assumed for the containment building ambient temperatures.

- *Draindown to Bottom of Reactor Vessel Nozzles (Best Estimate)*. Modifications to the best estimate models were made to reflect draindown. The system heat capacity was again decreased by the lowered water level on the preliminary side and the assumed complete drainage of the steam generator secondary-side water. Uncertainty as to whether the steam generator heat sinks would be coupled with the core heat source when the cold legs were no longer full resulted in the development of two best estimate models. The first model included the heat transfer areas and heat capacities of the hot legs and steam generators, and the second did not include them. The two models were considered because of the uncertainty of whether the hot legs and steam generators contributed to heat transfer and heat capacity, which were deduced from heatup data following the draindown for the Quick Look inspection.

Two possible heat transfer mechanisms were considered. One was the convection of heated vapor up the hot legs to the steam generators, which would still function with the RCS water level at the bottom of the reactor vessel nozzles. The other was a stratified convective circulation through the cold legs to the steam generators, which would be interrupted by the reduced water level. Since the validity of each of the two possible heat transfer mechanisms was unknown, the two best estimate models were postulated.

- ***Results***. Both conservative and best estimate models of equilibrium temperatures and heatup rates were determined for draindown to the head removal level and to the bottom of the reactor vessel nozzles (hot and cold legs). The equilibrium temperatures and heatup rates calculated with the best estimate models were predictably lower than those calculated with the models from the April 1982 analysis.

  - *Head Removal Level*. For draindown to the head removal level (321.5-foot elevation), the conservative temperatures and heatup rates showed that RCS temperatures did not exceed the criterion of 170 degrees F after December 1, 1982. The best estimate temperatures and heatup rates were believed to be more representative of the expected RCS temperature response to the draindown to head removal level and were in the range of 110 to 120 degrees F.

  - *Vessel Nozzle Level*. The conservative temperatures and heatup rates for draindown to the bottom of the reactor vessel nozzles (314-foot elevation) exceeded the criterion of 170 degrees F for December 1, 1982, and July 1, 1983. The best estimate temperatures and heatup rates for this water level, however, were well below the criterion for all specified dates for the models both with and without hot-leg/steam generator heat transfer areas.

- ***Conclusion***. The revised report concluded that, based on the conservative models from the April 1982 analysis, the RCS draindown to the reactor vessel head removal level could be accomplished without exceeding the temperature criterion after December 1, 1982. The conservative models from the April 1982 analysis supported draindown to the bottom of the reactor vessel nozzles after January 1, 1984. Based on the best estimate models, however, RCS draindown to the bottom of the reactor vessel nozzles could be accomplished without exceeding the temperature criterion after December 1, 1982. To ensure adequate margin to boiling, the criterion was that RCS bulk water temperature did not exceed 170 degrees F.

6.3 Data Collection Activities

6.3.1 Axial Power Shaping Rod Insertion Test (NA)

6.3.2 Camera Insertion through Reactor Vessel Leadscrew Opening

- ***Purpose***. To insert a camera into the reactor vessel through a control rod drive mechanism leadscrew opening to provide the first visual assessment of conditions inside the reactor vessel. This activity was known as "Quick Look."

- ***Evaluation***: ***Decay Heat Removal***. [14] Appendix B to the licensee's safety evaluation report described the effects of the draindown of the reactor coolant system (RCS) on decay heat removal capabilities (refer to Section 2.1 in this chapter).

The results of the analysis showed the following: (●) Some flow of water through the RCS loops (hot legs and cold legs) indicated that the heat rejection mode included system components beyond the reactor vessel and head. (●) Lowering the reactor water level and isolating the system components would increase the reactor water equilibrium temperature. (●) The reactor temperature increase to achieve equilibrium conditions depended on the heat sink temperature (i.e., containment building temperature). (●) Anticipated heatup rate, after partial draindown to the 323.5-foot elevation, was expected to be less than 5 degrees Fahrenheit (degrees F) per day during the summer months in mid-1982 with a containment building temperature of 100 degrees F. The report concluded that sufficient time would be available to monitor the actual heatup rate and to determine if enhanced heat removal would be required to maintain the equilibrium peak temperature below about 190 degrees F. (●) Existing systems such as feed and bleed through letdown and the standby pressure control system, refill of the RCS, or mini-decay heat removal system could perform enhanced heat removal, if required.

This analysis was based on an RCS water level 1 foot above the plenum cover (the elevation of the plenum cover was 322.5 feet). For the Quick Look, the RCS water level would be lowered to between the 331-foot and 335-foot elevations. The analysis noted that the heatup rate should be less for the Quick Look examination because of the additional water in the RCS during the examination.

- ***NRC Review: Decay Heat Removal***. [15] The NRC's safety evaluation noted that the decay heat in the reactor core was about 45 kilowatts, and the average in-core reactor coolant temperature was limited by operating procedures to 170 degrees F. Decay heat was removed by losses to ambient air. The partial draindown of the RCS would reduce the effective area for convective heat removal to the air circulating in the containment building causing a potential rise in the average reactor coolant temperature in the pressure vessel by a few degrees during the Quick Look activities. However, the NRC evaluation expected this temperature rise to be minimal for the following reasons: (●) The Quick Look program would be conducted over a relatively short time period (about 1 week). Following completion of the Quick Look, the RCS would be pressurized and refilled to the water-solid condition. (Experience with decay heat removal would be factored into future examinations.) (●) The average in-core temperature would be about 100 degrees F; thus, a large margin would exist between the actual temperature and the procedure limit. (●) The rate of temperature rise in the pressure vessel would depend on the temperature of the containment building atmosphere. The ambient temperature in the building would be relatively cool, in the range of 70 to 75 degrees F, and the core heatup rate would be correspondingly low.

A conservative interpretation of the decay heat analysis curves in the licensee's safety evaluation indicated that the average in-core temperature would not rise more than 25 degrees F during the 1-week effort. Further, if unexpected temperature increases occurred, initiating feed and bleed to the RCS, refill of the RCS, or operation of the mini-decay heat removal system would provide adequate backup decay heat removal capability.

6.3.3 Reactor Vessel Underhead Characterization (Radiation Characterization)

- ***Purpose***. To characterize radiation under the reactor vessel head to ensure that adequate radiological protection measures would be taken to keep radiation exposures ALARA during head removal. To achieve this objective, measurements were necessary with the reactor vessel head still in place.

- ***Evaluation***: ***Decay Heat Removal***. (16) The licensee's safety evaluation noted that the performance of the underhead radiation characterization would require the water level in the reactor coolant system (RCS) to be lowered to about the 321.5-foot elevation, which was about 1 foot below the top of the plenum. At the 321.5-foot elevation, there was significantly less water volume in the RCS than had been maintained in the past. As a result of having less water volume in the RCS, the ability to continue decay heat removal and maintain the bulk RCS temperature within acceptable limits for the losses to ambient cooling mode was investigated.

The licensee submitted an analysis of decay heat removal with the RCS water level at the 321.5-foot elevation to the NRC as an appendix to the licensee's safety evaluation report (17) for Quick Look. An updated analysis (18) was performed with the RCS water level at the 321.5-foot elevation and at the bottom of the reactor vessel nozzles (314-foot elevation). The results of the new analysis showed that the expected rise in RCS temperature would be acceptable.

This updated analysis presented both a conservative analysis and a best estimate analysis. The conservative calculations were made with the models originally developed for the appendix to the Quick Look safety evaluation. This conservative analysis showed RCS bulk temperatures of 165 degrees Fahrenheit (degrees F) and 198 degrees F for an RCS water level at 321.5-foot and 314-foot elevations, respectively. These temperatures were based on the decay heat rate for December 1, 1982 (or 32 months following the accident). The analysis showed that the RCS bulk temperature decreased with time as the decay heat rate decreased for the same water level.

The best estimate models that were benchmarked to the temperatures measured following the partial draindown for the Quick Look video inspection were developed to predict the expected RCS bulk temperatures. These models resulted in RCS temperatures between 112 and 120 degrees F with the RCS water level at the 321.5-foot elevation and 149 to 152 degrees F with the RCS water level at the 314-foot elevation. These temperatures were based on the decay heat rates for December 1, 1982, and July 1, 1983, respectively.

Subsequent to Quick Look, a test instrument was inserted into the reactor vessel above the rubble bed to verify the temperatures being recorded by the in-core thermocouples. The results of this effort indicated that the in-core thermocouples were providing a reasonably accurate measurement of the water temperature in the core cavity and were transmitted (19) to the NRC. Since the best estimate models were benchmarked against the in-core thermocouple readings, there was reasonable assurance that the temperatures predicted by the best estimate models were accurate. The licensee's safety evaluation concluded that the bulk RCS temperature for the draindown conditions associated with underhead characterization were well within the acceptable values for the present losses to ambient cooling mode.

- ***NRC Review: Decay Heat Removal***. (20) The NRC's review of potential hazards associated with the proposed activity included decay heat removal, which had been analyzed previously in the agency's safety evaluation report (21) for the Quick Look program and verified to some degree during the initial underhead characterization study. Based on this previous work, the NRC concluded that there was a high level of confidence that these potential hazards would not present a problem during the proposed characterization study.

6.3.4 Reactor Vessel Underhead Characterization (Core Sampling)

- ***Purpose***. To obtain up to six specimens of the core debris using specially designed tools. These tools would be lowered into the reactor to extract samples of the loose debris and to load the samples into small, shielded casks for offsite shipment and analysis. The analysis of the samples would: (●) identify the composition of the particulate core debris; (●) determine particle size; (●) determine fission product content; (●) determine fission product leachability from the debris; (●) analyze the drying properties of the debris; and (●) determine whether pyrophoric materials existed in the core debris. This examination was a follow-on activity to the underhead characterization study.

- ***Background***. Three shipping casks [c] were considered (22) for transporting the core debris samples to the laboratory. The selected cask was the modified and recertified Model

[c] Editor's Note: While large shipping containers of radioactive materials may often be referred to as "shipping casks," the proper term for such containers, when loaded with contents and in their transportation configuration, is "package." See 10 CFR Part 71, "Packaging and Transportation of Radioactive Material," Section 71.4, "Definitions."

CNS 1-13C Type B shipping cask, with the sample in a Specification 2R container. An earlier version of this cask was used to ship spent submerged demineralizer system liners. [23]

- ***Evaluation***: ***Decay Heat Removal***. [24] The licensee's safety evaluation stated that the review of the activities associated with the debris sample program had not revealed any additional safety considerations in the underhead radiation characterization program. The debris sample program consisted of activities similar to those performed during the core probe into the debris bed, which was previously performed safely. Therefore, the licensee concluded that the debris sample program would not present any undue risk to the health and safety of the public.

- ***NRC Review***. [25] The NRC's review did not specifically examine this topic.

6.3.5 Core Stratification Sample Acquisition (NA)

6.3.6 Use of Metal Disintegration Machining System to Cut Reactor Vessel Lower Head Wall Samples

- ***Purpose***. To remove metallurgical samples from the inner surface of the lower reactor vessel head using the metal disintegration machining system. An international research program used these samples to study core melt and reactor vessel interactions. This program was conducted after the reactor vessel was defueled.

- ***Evaluation: Decay Heat Removal***. [26, 27] The licensee's safety evaluation concluded that the decay heat removal concerns were bounded by the assessment in the licensee's safety evaluation report [28] for bulk defueling. This report stated that ambient decay heat removal was adequate throughout defueling. Therefore, decay heat removal would be more than adequate during this post-defueling activity.

- ***NRC Review: Decay Heat Removal***. [29] The NRC's safety evaluation concluded that issues related to pyrophoricity, fire protection, decay heat, and release of radioactivity fell within the bounds of previous agency safety evaluation reports [30, 31, 32,] for defueling activities.

6.4 Pre-Defueling Preparations

6.4.1 Containment Building Decontamination and Dose Reduction Activities (NA)

6.4.2 Reactor Coolant System Refill (NA)

6.4.3 Reactor Vessel Head Removal Operations (NA)

6.4.3.1 Polar Crane Load Test

- ***Purpose***. To conduct load testing of the polar crane in preparation for reactor vessel head lift and removal. The polar crane load test required four major sequential activities: (1) relocation of the internals indexing fixture; (2) assembly of test load; (3) load test; and (4) disassembly of test load.

- ***Evaluation: Decay Heat Removal***. (33) The licensee's safety evaluation considered normal decay heat removal and a postulated loss in decay heat removal capability.

  - *Normal Decay Heat Removal*. The decay heat production rate diminished greatly from the time of the accident to the time when active heat removal systems were no longer required (and, in fact, had not been required for some time). The mode of decay heat removal up to this time was by natural losses to ambient air via the reactor coolant system with the main pumps idle. Previous analyses showed that the decay heat rate was so small that losses to ambient air could occur without undesirable consequences (such as boiling) with a level of coolant in the reactor vessel lowered to the elevation of the bottom of the cold-leg nozzles. Given this condition, decay heat removal could be maintained even with a postulated main loop cold-leg piping sheared off at the vessel nozzles. The planned activities of the load test would not include manipulation or use of any system associated with the maintenance of decay heat removal capability. Therefore, no credible unplanned occurrence could result in a loss of this capability.

  - *Postulated Loss of Decay Heat Removal*. Decay heat removal capability was ensured by maintaining water in the reactor vessel. Analysis showed that the water could be drained to the bottom of the cold-leg nozzles with no adverse consequences, such as boiling. The only way to drain the vessel below this level would be through damage to the in-core instrument tubes. Also, if damage to the reactor coolant system that caused leakage of reactor coolant were to occur, makeup capability existed at least through one loop, since damage to makeup injection lines in both D-rings [d] at the same time was not credible by physical separation. Damage to makeup system penetrations would not occur since they were located on the north side of the containment building away from the load paths.

- ***NRC Review: Decay Heat Removal***. (34) The NRC's safety evaluation considered nominal decay heat removal and a postulated loss in decay heat removal capability. The reactor had been shut down for about 4.5 years, and the decay heat generation had decayed to a level of about 24 kilowatts, roughly the heat generated by 25 household toasters. The removal of decay heat was occurring by purely passive means (losses to ambient air); therefore, the potential loss

[d] D-rings were shield enclosures around the steam generator compartments; they were so named because of their shape.

of an active means of removing decay heat from the core, as a result of a heavy load drop accident, was determined not to be a serious concern.

6.4.3.2 First-Pass Stud Detensioning for Head Removal (NA)

6.4.3.3 Reactor Vessel Head Removal Operations

- ***Purpose***. To prepare for reactor vessel head removal to perform the lift and transfer of the head and conduct the activities necessary to place and maintain the reactor coolant system (RCS) in a stable configuration for the next phase of the recovery effort.

- ***Evaluation: Decay Heat Removal***. [35] The licensee's safety evaluation considered the impact on decay heat removal during this activity. Head removal activities required that the RCS water level be lowered to about 1 foot below the plenum cover plate. At this level, there was less water in the RCS than had been maintained in the past, except during the underhead characterization program. As a result of having less water in the RCS, the ability to continue to adequately remove decay heat and maintain the bulk RCS temperature within procedural limits (i.e., 170 degrees Fahrenheit (degrees F)) for the losses to ambient cooling mode was investigated.

- ***Results.*** An analysis of decay heat removal ability with the RCS water level at 1 foot above the plenum cover plate) was submitted to the NRC as Appendix B to the safety evaluation report (SER) [36] for the Quick Look examination (refer to Section 2.1 in this chapter). The Quick Look analysis was updated [37] for the head removal with additional analyses of the RCS water level at 1 foot below the plenum cover plate and at the bottom of the reactor vessel nozzles. This update was provided as Attachment 1 to the head removal SER (refer to Section 2.2 in this chapter). The results showed that, even for the conservative analysis, the expected rise in RCS temperature would be acceptable to maintain the losses to ambient cooling mode for the normal draindown level at 1 foot below the plenum cover plate.

  - *Models*. The safety evaluation presented both a conservative analysis and a best estimate analysis. The conservative calculations were made with the models originally developed for the Quick Look safety evaluation. The Quick Look conservative analysis resulted in equilibrium RCS bulk temperatures of 158 degrees F and 183 degrees F for the RCS water level at 1 foot below the plenum cover plate and at the bottom of the reactor vessel nozzles, respectively. These temperatures were based on the decay heat rate for July 1, 1983. The analysis showed that for the same water level, the RCS bulk temperature decreased with time as the decay heat rate decreased. Given that decay heat power of the core continued to decrease, the equilibrium temperature 1 year later (during the planned head lift) was expected to be lower.

  - *Model Benchmarking*. The best estimate models, which were benchmarked to the temperatures measured following the partial draindown for the Quick Look video inspection, were developed to predict the expected RCS bulk temperatures. These models resulted in RCS temperatures of 120 degrees F and 151 degrees F at both water levels, based on the decay heat rate for July 1, 1983. Subsequent to the Quick Look inspection, a test instrument inserted into the reactor vessel above the rubble bed verified that temperatures being recorded by the in-core thermocouples provided a reasonably accurate measurement of the temperature of the water in the core cavity, and these results were transmitted [38] to the NRC. RCS draindown for underhead characterization was in August 1983. The average RCS thermocouple temperature was 90 degrees F in December 1983.

- *Conclusion*. Since test results indicated that the in-core thermocouples were providing reasonable measurements of the water temperature in the core cavity and the best estimate models were benchmarked against the in-core thermocouple readings, there was reasonable assurance that the temperatures predicted by the best estimate models were accurate. Therefore, the licensee's evaluation concluded that the bulk equilibrium RCS temperature for the draindown conditions associated with head removal were well within the acceptable values for the losses to ambient cooling mode.

- ***NRC Review: Decay Heat Removal***. [39] The NRC's safety evaluation noted that the decay heat removal, currently estimated to be about 17.0 kilowatts, continued to be adequately dissipated through losses to the containment building ambient air. The partial draindown of the RCS during head removal activities would reduce the effective area for convective heat transfer to the containment building atmosphere; therefore, an increase in average RCS temperature was anticipated. However, the loss-to-ambient cooling mode was expected to be sufficient to keep the average RCS temperature well below the procedural limit of 170 degrees F for the duration of head lift activities and beyond.

- *Analysis Assumptions and Results*. The licensee performed conservative analyses that indicated a substantial margin would exist between the predicted average RCS temperature and the procedural limit. The conservative assumptions used in the analysis included the following: (●) RCS draindown to a level 7.5 feet below the planned level for head lift with associated reduction in heat transfer surface area; (●) decay heat levels as of January 1, 1984, which was 7 months before the scheduled RCS draindown; and (●) initial temperature for containment building ambient air and RCS well above current measured temperatures. The bounding case that incorporated all of the above assumptions yielded an equilibrium RCS temperature of 151 degrees F; thus, this case provided a margin of 19 degrees F below the procedural limit of 170 degrees F.

- *Experimental Data*. The actual RCS equilibrium temperatures following draindown for head lift were expected to be closer to 115 degrees F, based on the temperature experience of the drained-down RCS during the 10-month underhead characterization study in 1983 and 1984. The RCS temperature never increased to higher than about 114 degrees F during the underhead characterization study, amounting to a temperature rise of 10 to 12 degrees F from ambient levels (i.e., the RCS temperatures before draindown).

- *Contingencies*. The licensee's analysis showed that it was highly unlikely that the average RCS temperatures would approach the procedural limit during head lift operations with decay heat being removed in the loss-to-ambient mode. However, in the event of an unexpected temperature increase, several backup heat removal systems were available. These systems, which included the mini-decay heat removal system and the normal decay heat removal system, were sized to handle decay heat loads well in excess of the present core decay heat rate. Therefore, adequate backup heat removal capability existed to support head lift activities. Section 3.G of the NRC's SER examined the consequences of a postulated reactor vessel head drop on decay heat removal capability during lifting operations.

6.4.4 Heavy Load Handling inside Containment

- ***Purpose***. To ensure that handling of heavy loads in the containment building and spent fuel pool "A" within the fuel handling building was in accordance with the safety requirements of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," [40] issued July 1980.

- ***Evaluation: Decay Heat Removal***. [41] The licensee's safety evaluation for load drop accidents included decay heat removal evaluation as required by NUREG-0612, to ensure safe-shutdown functions during postulated heavy load drop accidents. Reactor coolant would be maintained in the reactor system above the reactor vessel nozzles for decay heat removal. Decay heat was removed by heat losses to ambient air, which was demonstrated to be adequate to remove all decay heat [42] produced by the core material in the reactor vessel. Thus, no additional equipment was necessary to remove decay heat.

- ***NRC Review: Decay Heat***. [43] The NRC's safety evaluation noted that there would be no immediate effects if the spent fuel pool or fuel transfer canal were to drain. Given that the entire core had a decay heat of less than 12 kilowatts and heat generation in individual stored canisters would be less than 100 watts, the NRC's safety evaluation concluded that decay heat would not pose a problem.

6.4.5 Heavy Load Handling over the Reactor Vessel (NA)

6.4.6 Plenum Assembly Removal Preparatory Activities

- ***Purpose***. To confirm the adequacy of plenum removal equipment and techniques, the preparatory activities included: (●) video inspection of potential interference areas; (●) video inspection of the core void space; (●) video inspection of the axial power shaping rod (APSR) assemblies; (●) measurement of the loss-of-coolant accident restraint boss gaps; (●) measurement of the elevations of the APSR assemblies; (●) cleaning of the plenum and potential interference areas; (●) separation of unsupported fuel assembly end fittings; and (●) movement of the APSR assemblies.

- ***Evaluation: Decay Heat Removal***. [44] Editor's Note: The licensee's original safety evaluation report (SER) did not specifically address this topic. The revision to this SER was not located.

- ***NRC Review: Decay Heat Removal***. [45, 46] The NRC issued two SERs: the first SER covered the first five activities (see the above purpose), and the second SER covered the remaining three activities. The first SER was brief and did not specifically address this topic. However, the first safety evaluation stated that the first five activities were previously addressed in the NRC's SERs [47, 48] for the Quick Look video inspection of the reactor core and for the reactor vessel underhead characterization. The NRC concluded that the licensee's experience with the core and plenum video inspections and other in-vessel activities (e.g., radiation measurements, reactor coolant sampling) demonstrated that these were benign activities (i.e., their environmental impacts were very small) and posed little risk to the onsite workers or offsite public. The NRC further stated that the corresponding plenum inspection activities did not warrant further review.

The NRC's second SER for the remaining three activities noted that the decay heat in the reactor core continued to be removed adequately in the loss-to-ambient cooling mode. The reactor coolant system would remain in draindown, in a depressurized condition during the proposed activities. Consequently, the coolant temperature (about 97 degrees Fahrenheit) was expected to remain essentially unchanged by these activities. The NRC's SER [49] for the reactor vessel head removal described the existing safety margin and alternative decay heat removal methods that would also be available during the plenum removal preparatory activities. Additionally, heat removal in the loss-to-ambient mode would be enhanced by the recent installation of the containment building chiller system and by the operation of the internals indexing fixture processing system. The NRC concluded that the heat removal capability was adequate to accommodate the small amount of decay heat (17.0 kilowatts) in the core.

6.4.7 Plenum Assembly Removal (NA)

6.4.8 Makeup and Purification Demineralizer Resin Sampling (NA)

6.4.9 Makeup and Purification Demineralizer Cesium Elution (NA)

6.5 Defueling Tools and Systems

6.5.1 Internals Indexing Fixture Water Processing System

- ***Purpose***. To provide reactor coolant water processing capability during head removal until plenum removal. The internals indexing fixture (IIF) processing system was selected based on the limited reactor coolant processing capability in the drained-down condition and the desire to provide adequate water cleanup capacity to minimize radiation dose rates around the IIF.

- ***Evaluation: Decay Heat Removal***. [50] The licensee's safety evaluation noted that the reactor coolant system (RCS) level would be maintained well above the plenum cover plate elevation during operation of the IIF processing system. The bulk RCS temperature was previously maintained at less than 100 degrees Fahrenheit by the loss-to-ambient cooling mode with the water level below the plenum cover plate. Therefore, no reduction in decay heat removal capability was expected during IIF processing system operation. If the water level decreased and the low-level alarm trip failed, the RCS level could be lowered only to the pump suction point, about 2.5 feet above the plenum cover plate, located above the reactor vessel nozzles. Adequate decay heat removal capability was demonstrated for the RCS level lowered to the nozzles in the technical evaluation report [51] for the submerged demineralizer system. Therefore, the licensee's safety evaluation concluded that adequate decay heat removal capability would be available in the event of failure of level control and low-level trip.

- ***NRC Review***. [52] Editor's Note: The NRC's safety evaluation report did not specifically address this evaluation topic.

6.5.2 Defueling Water Cleanup

6.5.2.1 Defueling Water Cleanup System

- ***Purpose***. To remove organic carbon, radioactive ions, and particulate matter from the reactor vessel and fuel transfer canal/spent fuel pool "A" (FTC/SFP-A). The defueling water cleanup system (DWCS) was actually two independent subsystems (sometimes called trains): the reactor vessel cleanup system and the FTC/SFP-A cleanup system. The DWCS was totally contained within areas that had controlled ventilation and could be isolated. This limited the environmental impact of the system during normal system operations, shutdown, or postulated accident conditions.

- ***Evaluation: Decay Heat Removal***. [53] Decay heat was removed by heat loss to ambient air inside the containment building. The licensee's safety evaluation noted that no change in this mode of operation was required to operate the DWCS. The large, exposed surface of the open reactor vessel and the fuel transfer canal would significantly enhance the removal of decay heat.

- ***NRC Review***. [54] Editor's Note: The NRC's safety evaluation report did not specifically address this evaluation topic.

6.5.2.2 Cross-Connect to Reactor Vessel Cleanup System (NA)

6.5.2.3 Temporary Reactor Vessel Filtration System (NA)

6.5.2.4 Filter-Aid Feed System and Use of Diatomaceous Earth as Feed Material (NA)

6.5.2.5 Use of Coagulants (NA)

6.5.2.6 Filter Canister Media Modification (NA)

6.5.2.7 Addition of a Biocide to the Reactor Coolant System (NA)

6.5.3 Defueling Canisters and Operations (NA)

6.5.3.1 Defueling Canisters: Filter, Knockout, and Fuel (NA)

6.5.3.2 Replacement of Loaded Fuel Canister Head Gaskets (NA)

6.5.3.3 Use of Debris Containers for Removing End Fittings (NA)

6.5.3.4 Fuel Canister Storage Racks

- ***Purpose***. To provide storage for the three different types of canisters (fuel, filter, and knockout) filled with debris material from the reactor vessel. Storage for 263 canisters was available in the racks, located in spent fuel pool "A" and in the deep end of the fuel transfer canal.

- ***Evaluation: Decay Heat Removal***. [55] The licensee's safety evaluation noted that the loaded defueling canisters, when standing in the fuel canister storage racks, were cooled by the fuel pool water. The decay heat produced by the fuel in each canister was about 60 watts (based on a decay heat load of 15 kilowatts being distributed among 250 canisters), and the surface area of the defueling canister exposed to the fuel pool water was about 46 square feet. The canister support plate contained the necessary openings for adequate cooling flow around the canisters.

- ***NRC Review***. [56, 57] Editor's Note: The NRC's safety evaluation report did not specifically address this evaluation topic.

6.5.3.5 Canister Handling and Preparation for Shipment (NA)

6.5.3.6 Canister Dewatering System (NA)

6.5.3.7 Use of Nonborated Water in Canister Loading Decontamination System (NA)

6.5.4 Testing of Core Region Defueling Techniques (NA)

6.5.5 Fines/Debris Vacuum System

- ***Purpose***. To modify the fines/debris vacuum system using a knockout canister and a filter canister in series. Modifications included: (●) use of a vacuum nozzle to allow larger debris particles to be vacuumed into the knockout canisters; (●) use of mechanical probes and water jets on the end of the vacuum nozzle to loosen the packed rubble; (●) use of a larger vacuum tool to allow debris removal from the lower head; and (●) temporary use of the vacuum system without a filter canister. The safety evaluation report [58, 59] for early defueling had previously approved the initial use of the fines/debris vacuum system.

- ***Evaluation: Decay Heat Removal (Draindown)***. [60] The licensee's safety evaluation noted that the use of the modified water jet nozzle was assessed for the potential safety concern raised by a lowered water level in the reactor vessel, involving a postulated hose rupture or inadvertent operation of the submersible pump. The extent of water level decrease was bounded by the location of the pump suction. The pump was located within the internals indexing fixture that precluded the lowering of the reactor vessel water level near the core. A previous safety analysis had addressed a draindown event below the reactor vessel flange.

- ***NRC Review***. [61] Editor's Note: The NRC's safety evaluation report did not specifically address this evaluation topic.

6.5.6 Hydraulic Shredder (NA)

6.5.7 Plasma Arc Torch

6.5.7.1 Use of Plasma Arc Torch to Cut Upper End Fittings

- ***Purpose***. Use of a plasma arc torch to cut the upper end fittings inside the reactor vessel to allow the fittings to be placed directly in fuel canisters for transfer from the reactor vessel to the fuel canal. This was the first use of the plasma arc torch inside the reactor vessel.

- ***Evaluation: Decay Heat Removal***. [62] The licensee's safety evaluation stated that the calculations for the energy input from operating the plasma arc indicated an increase in the reactor coolant system water temperature no more than 0.25 degree Fahrenheit per hour.

- ***NRC Review***. [63] Editor's Note: The NRC's safety evaluation report did not specifically address this evaluation topic.

6.5.7.2 Use of Plasma Arc Torch to Cut Lower Core Support Assembly (NA)

6.5.7.3 Use of Plasma Arc Torch to Cut Baffle Plates and Core Support Shield (NA)

6.5.7.4 Use of Air as Secondary Gas for Plasma Arc Torch (NA)

6.5.8 Use of Core Bore Machine for Dismantling Lower Core Support Assembly (NA)

6.5.9 Sediment Transfer and Processing Operations (NA)

6.5.10 Pressurizer Spray Line Defueling System (NA)

6.5.11 Decontamination Using Ultrahigh Pressure Water Flush (NA)

6.6 Evaluations for Defueling Operations

6.6.1 Preliminary Defueling

- ***Purpose***. To allow movement of debris within the reactor vessel, to allow installation of defueling equipment, and to identify core debris samples for later removal and analysis by INEL. The movement of debris within the vessel was prepared for early defueling and required some rearrangement of debris in the reactor vessel before the actual removal of fuel. These activities included the loading of small pieces of core debris into debris baskets but did not include actual loading of fuel debris into fuel canisters.

- ***Evaluation: Decay Heat Removal***. (64) Editor's Note: The licensee's safety evaluation report did not specifically address this topic.

- ***NRC Review: Decay Heat Removal***. (65) The NRC's safety evaluation concluded that decay heat from the damaged reactor core continued to be adequately removed from the reactor coolant through the loss-to-ambient cooling mode. In the NRC's safety evaluation report (66) for the reactor pressure vessel head lift, the agency approved the licensee's analysis that demonstrated that decay heat removal via the loss-to-ambient mode was sufficient to maintain the reactor coolant system (RCS) temperature well below the procedural limit of 170 degrees Fahrenheit. The assumptions used in that analysis were more conservative than the present conditions because the proposed activities would be performed at a decreased decay heat level and with the reactor vessel water level at an elevation of about 327.5 feet, as opposed to the level of only 314 feet assumed in the analysis. Temperatures in the reactor vessel averaged about 85 degrees Fahrenheit and were not expected to change significantly. Therefore, the NRC found that the previous analysis was bounding and that the loss-to-ambient mode of decay heat removal would be adequate during the proposed debris movement activities. The RCS temperature would be monitored in accordance with the recovery technical specifications and the recovery operations plan (technical specifications surveillance schedule) during the proposed activities. In the unlikely event of RCS leakage, sufficient makeup capability would be available to maintain decay heat removal in the loss-to-ambient mode.

6.6.2 Early Defueling

- ***Purpose***. To remove end fittings, structural materials, and related loose debris that would not involve removal of significant amounts of fuel material; to remove intact segments of fuel rods and other pieces of core debris; and to remove loose fuel fines (particles) by vacuuming operations.

- ***Evaluation: Decay Heat Removal***. [67] Editor's Note: The licensee's safety evaluation report for early defueling was practically identical to its subsequent safety evaluation report [68, 69] for bulk defueling; therefore, this section does not include the evaluation text. Please refer to the next section for details.

- ***NRC Review: Decay Heat Removal***. [70] The NRC's safety evaluation noted that the decay heat in the damaged reactor core (about 12 kilowatts) continued to be removed in the loss-to-ambient mode. Defueling activities were not expected to significantly increase the current reactor coolant system temperature of 85 degrees Fahrenheit; however, the reactor coolant temperature would be monitored in accordance with the technical specifications and associated surveillance requirements as documented in the recovery operations plan (technical specifications surveillance schedule). The conclusions of a previous NRC safety evaluation report [71] for preliminary defueling operations with respect to decay heat removal were applicable throughout early defueling activities, and the loss-to-ambient mode would adequately dissipate the small amount of decay heat generated.

6.6.3 Storage of Upper End Fittings in an Array of 55-Gallon Drums (NA)

6.6.4 Defueling (Also Known as "Bulk" Defueling)

- ***Purpose***. To develop defueling tools and to conduct activities necessary to remove the remaining fuel and structural debris located in the original core volume. An additional activity was to address vacuuming of debris from the lower head of the vessel, a primary activity associated with lower head defueling.

- ***Evaluation: Decay Heat Removal***. [72, 73] The licensee's safety evaluation noted that during defueling activities, the water level in the reactor coolant system (RCS) would be at an elevation about 5 feet above the vessel flange. An analysis of decay heat removal ability with the RCS water level at an elevation 13 feet lower was previously performed in the licensee's safety evaluation report [74] for removal of the reactor vessel head. The results of this best estimate analysis showed that loss-to-ambient cooling would maintain the RCS bulk temperature at less than 170 degrees Fahrenheit during defueling activities. The video system lighting and other defueling equipment would add additional heat to the reactor vessel water; however, these heat sources were not considered a safety problem, as the lights or other equipment could be turned off to eliminate heat input if the water temperature increased to unacceptable levels. The RCS water temperature would be monitored during defueling in accordance with the recovery technical specifications and the recovery operations plan (technical specifications surveillance requirements).

- ***NRC Review: Decay Heat Removal***. [75] The NRC's safety evaluation determined that decay heat removal was adequately addressed in previous agency safety evaluations and that the agency's earlier conclusions were applicable to the proposed activities.

Editor's Note: Refer to the NRC's safety evaluation report [76] for early defueling.

6.6.5 Use of Core Bore Machine for Bulk Defueling (NA)

6.6.6 Lower Core Support Assembly Defueling

- ***Purpose***. To dismantle and defuel the lower core support assembly (LCSA) and to partially defuel the lower reactor vessel head. Structural material removed from the LCSA included the: (●) lower grid rib assembly; (●) lower grid forging; (●) distribution plate; and (●) in-core guide support plate. The lowest elliptical flow distributor and the gusseted in-core guide tubes would not be removed under this proposed activity. The use of the core bore machine, plasma arc cutting equipment, cavitating water jet, robot manipulators, automatic cutting equipment system, and other previously approved tools and equipment was included in this activity and associated safety evaluations.

- ***Evaluation: Decay Heat Removal***. [77] The licensee's safety evaluation stated that decay heat removal concerns during LCSA defueling were generally bounded by the licensee's safety evaluation report [78] for bulk defueling. The maximum power requirements for the plasma arc torch were 1000 amps at 200 volts direct current. Operation of the torch underwater would provide a significant heat source; however, continuous operation was not probable because of the need to reposition the torch. Even if the torch were to operate continuously for 1 hour, this heat source would raise the reactor coolant system temperature only about 2 degrees Fahrenheit. The reactor coolant system temperature would be monitored to preclude an uncontrolled water temperature increase.

- ***NRC Review***. [79] Editor's Note: The NRC's safety evaluation report did not specifically address this evaluation topic.

6.6.7 Completion of Lower Core Support Assembly and Lower Head Defueling

- ***Purpose***. To remove the elliptical flow distributor and gusseted in-core guide tubes of the lower core support assembly and to defuel the reactor vessel lower head.

- ***Evaluation: Decay Heat Removal***. [80] The licensee's safety evaluation concluded that decay heat removal concerns during the defueling of the lower core support assembly and the reactor vessel lower head were generally bounded by the evaluation in the licensee's safety evaluation report [81] for bulk defueling. The maximum power requirements for the plasma arc torch were 1000 amps at 200 volts direct current. Operation of the torch underwater would provide a significant heat source; however, continuous operation was not probable because of the need to reposition the torch. Even if the torch were to operate continuously for 1 hour, the reactor coolant temperature would rise only about 2 degrees Fahrenheit. The reactor coolant temperature would be monitored to preclude an unlikely uncontrolled water temperature increase.

- ***NRC Review***. [82] Editor's Note: The NRC's safety evaluation report did not specifically address this evaluation topic.

6.6.8 Upper Core Support Assembly Defueling

- ***Purpose***. To cut and move the baffle plates and to defuel the upper core support assembly. This evaluation addressed the following activities: (●) cutting the baffle plates for later removal; (●) removing bolts from the baffle plates; (●) removing the baffle plates; and (●) removing core debris from the baffle plates and core former plates.

- ***Evaluation: Decay Heat Removal***. [83] The licensee's safety evaluation stated that decay heat removal concerns during upper core support assembly defueling were generally bounded by the evaluation in the licensee's safety evaluation report [84] for bulk defueling. The maximum power requirements for the plasma arc torch were 1000 amps at 200 volts direct current. Operation of the torch underwater provided a significant heat source; however, continuous operation was not probable because of the need to reposition the torch. Even if the torch were to operate continuously for 1 hour, the reactor coolant temperature would rise only about 2 degrees Fahrenheit. Reactor coolant temperature would be monitored to preclude an unlikely uncontrolled water temperature increase. Experience from prior cutting activities confirmed that water temperature was not measurably affected by operation of the plasma arc torch.

- ***NRC Review***. [85] Editor's Note: The NRC's safety evaluation report did not specifically address this evaluation topic.

6.7 Evaluations for Waste Management (NA)

6.7.1 EPICOR II (NA)

6.7.2 Submerged Demineralizer System (NA)

6.7.2.1 Submerged Demineralizer System Operations (NA)

6.7.2.2 Submerged Demineralizer System Liner Recombiner and Vacuum Outgassing System (NA)

6.8 Endnotes

Reference citations are exact filenames of documents on the Digital Versatile Discs (DVDs) to NUREG/KM-0001, Supplement 1, [86] "Three Mile Island Accident of 1979 Knowledge

Management Digest: Recovery and Cleanup," issued June 2016, DVD document filenames start with a full date (YYY-MM-DD) or end with a partial date (YYYY-DD). Citations for references that are not on a DVD begin with the author organization or name(s), with the document title in quotation marks. Contents of the DVDs are available at the Idaho National Laboratory Web site (https://tmi2kml.inl.gov).

[1] (1979-04-12) GPU Safety Analysis, Transition to Natural Circulation
[2] (1979-04-16) NRC (Internal), Updated Status of Proposals for Achieving Cold Shutdown for TMI-2
[3] NUREG-0557, Evaluation of Long-Term Post-Accident Core Cooling of TMI-2 (1979-05)
[4] (1979-08-17) GPU Response to NRC, Natural Circulation Stability (re 07-26-1979)
[5] (1980-01-04) GPU Response to NRC, Natural Circulation (re 11-05-1979)
[6] (1980-04-29) GPU Response to NRC, Mini Decay Heat Removal System Overpressure from OTSG Burp (re 04-16-1980)
[7] (1980-04-16) NRC, Mini Decay Heat Removal Overpressurization from Steam Generator Burp
[8] (1980-04-29) GPU Response to NRC, Mini Decay Heat Removal System Overpressure from OTSG Burp (re 04-16-1980)
[9] (1980-11-14) Order Amendment (also ROP-4)
[10] USNRC, "Three Mile Island Accident of 1979 Knowledge Management Digest: Recovery and Cleanup," NUREG/KM-0001, Supplement 1, June 2016 [Available at nrc.gov]
[11] (1982-07-06) GPU Safety Evaluation, Insertion Camera Through Reactor Vessel Leadscrew Opening, Rev. 2
[12] American National Standards Institute/American Nuclear Society, ANSI/ANS 5.1–1978, "Decay Heat Power in Light Water Reactors," ANS, LaGrange Park, IL.
[13] (1984-03-09) GPU Safety Evaluation, Head Removal, Rev. 5
[14] (1982-07-06) GPU Safety Evaluation, Insertion Camera Through Reactor Vessel Leadscrew Opening, Rev. 2
[15] (1982-07-13) NRC Review, Control Rod Drive Mechanism Quick Look Camera Inspection (re 07-06-1982) (2)
[16] (1983-05-19) GPU Safety Analysis, Radiation Characterization Under Reactor Vessel Head (No Polar Crane), Rev. 0
[17] (1982-07-06) GPU Safety Evaluation, Insertion Camera Through Reactor Vessel Leadscrew Opening, Rev. 2
[18] (1982-12-01) GPU, Addendum to TMI-2 Decay Heat Removal Report of April 1982, Rev. 1
[19] (1982-12-30) GPU, Results of Test Measurements of RCS Water Temperature
[20] (1983-07-13) NRC Review, Reactor Vessel Underhead Characterization (re Various Letters)
[21] (1982-07-13) NRC Review, Control Rod Drive Mechanism Quick Look Camera Inspection (re 07-06-1982)
[22] (1983-07-20) GPU Safety Evaluation, Underhead Characterization, Core Sampling Addendum
[23] USNRC, "Three Mile Island Accident of 1979 Knowledge Management Digest: Recovery and Cleanup," NUREG/KM-0001, Supplement 1, June 2016 [Available at nrc.gov]
[24] (1983-07-20) GPU Safety Evaluation, Underhead Characterization, Core Sampling Addendum
[25] (1983-08-19) NRC Safety Evaluation, Addendum to the Underhead Characterization Study (re 07-30-1983)
[26] (1989-08-18) GPU Safety Evaluation, Remove Metallurgical Samples from Reactor Vessel SER

[27] (1989-10-20) GPU Safety Evaluation, Remove Metallurgical Samples from Vessel, Rev. 1 (re 08-18-1989) (effective pages)
[28] 1986-05-15) GPU Safety Evaluation, Defueling SER, Rev. 10
[29] (1989-11-28) NRC Safety Evaluation, Reactor Vessel Lower Head Metallurgical Sampling (re 08-18-1989)
[30] (1985-10-29) NRC Safety Evaluation, Commencement of TMI-2 Preliminary Defueling Operations (re 10-24-1985)
[31] (1985-11-12) NRC Safety Evaluation, Early Defueling of the TMI-2 Reactor Vessel (10-10-1985)
[32] (1986-07-24) NRC Review, Core Region Bulk Defueling Operations Limited to the Core Region (re 05-15, 07-23-1986)
[33] (1983-02-18) GPU Safety Evaluation, Polar Crane Load Test
[34] (1983-11-18) NRC Safety Evaluation, Reactor Building Polar Crane Load Test
[35] (1984-03-09) GPU Safety Evaluation, Head Removal, Rev. 5
[36] (1982-07-06) GPU Safety Evaluation, Insertion Camera Through Reactor Vessel Leadscrew Opening, Rev. 2
[37] (1982-12-01) GPU, Addendum to TMI-2 Decay Heat Removal Report of April 1982, Rev. 1
[38] (1982-12-30) GPU, Results of Test Measurements of RCS Water Temperature
[39] (1984-07-17) NRC Safety Evaluation, Reactor Pressure Vessel Head Lift (re -3-09-1984)
[40] USNRC, "Control of Heavy Loads at Nuclear Power Plants," NUREG-0612, July 1980 [Available at nrc.gov]
[41] (1986-06-02) GPU Safety Evaluation, Heavy Load Handling Inside Containment, Rev. 3
[42] (1983-03-15) GPU Response to NRC, Polar Crane SER
[43] (1986-06-02) GPU Safety Evaluation, Heavy Load Handling Inside Containment, Rev. 3
[44] (1984-06-18) GPU Safety Evaluation, Preparatory Activities for Plenum Assembly Removal
[45] (1984-07-31) NRC Review, Preparatory Activities for Plenum Assembly Removal SER (re 06-18-1984)
[46] (1984-09-14) NRC Safety Evaluation, Plenum Assembly Removal (re various letters)
[47] (1982-07-13) NRC Review, Control Rod Drive Mechanism Quick Look Camera Inspection (re 07-06-1982)
[48] (1983-07-13) NRC Review, Reactor Vessel Underhead Characterization (re Various Letters)
[49] (1984-07-17) NRC Safety Evaluation, Reactor Pressure Vessel Head Lift (re -3-09-1984)
[50] (1984-05-31) GPU Safety Evaluation, Internals Indexing Fixture Processing System, Rev. 1
[51] (1982-11-03) GPU Technical Evaluation, RCS Processing with RCS in a Partial Drained Condition
[52] (1984-07-24) NRC Safety Evaluation, Internals Index Fixture Water Processing System (re 05-31, 07-16-1984)
[53] (1989-05-12) GPU Technical Evaluation, Defueling Water Cleanup System, Rev. 12
[54] (1985-08-06) NRC Safety Evaluation, Defueling Water Cleanup System (re various letters)
[55] (1986-05-29) GPU Technical Evaluation, Fuel Canister Storage Racks, Rev. 1
[56] (1985-05-03) NRC Review, Fuel Canister Storage Racks TER (re 02-27-1985)
[57] (1986-09-16) NRC Review, Fuel Canister Storage Racks TER (re various letters)
[58] (1985-10-10) GPU Safety Evaluation, Early Defueling of the TMI-2 Reactor Vessel, Rev. 4
[59] (1985-11-12) NRC Safety Evaluation, Early Defueling of the TMI-2 Reactor Vessel (10-10-1985)
[60] (1986-01-28) GPU, Fines-Debris Vacuum System Modifications (re various letters)
[61] (1986-02-21) NRC Review, Modification of Fines and Debris Vacuum System (re various letters)
[62] (1986-08-27) GPU, Plan to Use Plasma Arc Torch to Cut Upper End Fittings (re 08-18-1986)

[63] (1987-08-20) NRC Review, Use of Plasma Arc Torch to Cut Upper End Fittings (re various letters)
[64] (1985-10-24) GPU, Commencement of Preliminary Defueling Operations
[65] (1985-10-29) NRC Safety Evaluation, Commencement of TMI-2 Preliminary Defueling Operations (re 10-24-1985)
[66] (1984-07-17) NRC Safety Evaluation, Reactor Pressure Vessel Head Lift (re -3-09-1984)
[67] (1985-10-10) GPU Safety Evaluation, Early Defueling of the TMI-2 Reactor Vessel, Rev. 4
[68] (1986-05-15) GPU Safety Evaluation, Defueling SER, Rev. 10
[69] (1986-05-20) GPU Safety Evaluation, Defueling SER, Rev. 10 (Correction)
[70] (1985-11-12) NRC Safety Evaluation, Early Defueling of the TMI-2 Reactor Vessel (10-10-1985)
[71] (1985-10-29) NRC Safety Evaluation, Commencement of TMI-2 Preliminary Defueling Operations (re 10-24-1985)
[72] (1986-05-15) GPU Safety Evaluation, Defueling SER, Rev. 10
[73] (1986-05-20) GPU Safety Evaluation, Defueling SER, Rev. 10 (Correction)
[74] (1984-03-09) GPU Safety Evaluation, Head Removal, Rev. 5
[75] (1986-07-24) NRC Review, Core Region Bulk Defueling Operations Limited to the Core Region (re 05-15, 07-23-1986)
[76] (1985-11-12) NRC Safety Evaluation, Early Defueling of the TMI-2 Reactor Vessel (10-10-1985)
[77] (1988-01-18) GPU Safety Evaluation, Lower Core Support Assembly Defueling, Rev. 2
[78] (1986-05-15) GPU Safety Evaluation, Defueling SER, Rev. 10
[79] (1988-04-01) NRC Safety Evaluation, Lower Core Support Assembly Defueling (re 01-18-1988)
[80] (1988-06-06) GPU Safety Evaluation, Completion of Lower Core Support and Lower Head Defueling, Rev. 0
[81] (1986-05-15) GPU Safety Evaluation, Defueling SER, Rev. 10
[82] (1988-12-01) NRC Safety Evaluation, Lower Core Support and Lower Head Defueling (re 06-06-1988)
[83] (1988-09-15) GPU Safety Evaluation, Completion of Upper Support Core Assembly Defueling, Rev. 0
[84] (1986-05-15) GPU Safety Evaluation, Defueling SER, Rev. 10
[85] (1989-04-04) NRC Safety Evaluation, Completion of Upper Core Support Assembly Defueling (re 9-15-1988)
[86] USNRC, "Three Mile Island Accident of 1979 Knowledge Management Digest: Recovery and Cleanup," NUREG/KM-0001, Supplement 1, June 2016 [Available at nrc.gov]

7 LOAD DROP SAFETY EVALUATIONS

TABLE OF CONTENTS

Note: “NA” (not applicable) means the licensee and the NRC determined that the safety topic was not important for the activity.

7.1 Introduction

7.1.1 Background

The broad safety topic of load drops essentially covered the entire fuel cycle, ending with storage or disposal. This chapter focuses on the safety evaluations of handling heavy loads and postulated load drops during postaccident TMI-2 cleanup.

For the TMI-2 activities, as well as those at all commercial nuclear power plants, the issue of heavy load drops was a subject of interest to the NRC and the licensee. The need for the lifting of loads greater than loads that could be hand carried required the use of various hoists and cranes. The hoists and cranes used for handling heavy loads inside the containment building included: (●) main hoist of the polar crane (rated 174 tons) used to lift the reactor vessel missile shields, head, and plenum; (●) auxiliary hoist of the polar crane (rated 5 tons); (●) containment building service crane; (●) defueling work platform jib cranes used for handling heavy-duty defueling tools; (●) defueling canister-lifting mechanism within the fuel transfer cask used to lift the canister in and out of the water; and (●) canister handling bridge used to lift the fuel transfer cask from the work platform to the deep end of the fuel transfer canal.

The fuel handling building hoists and cranes included: (●) main hoist of the fuel handling building crane used to transfer submerged demineralizer system vessels and their shipping casks; (●) auxiliary hoist of the fuel handling building crane; (●) mini hot cell jib crane [a] ;(●) canister lifting mechanism within the fuel transfer cask used to lift the canister in and out of the water; and (●) canister handling bridge used to lift the fuel transfer cask from the work platform to the deep end of the fuel transfer canal.

An accidental load drop could impact nuclear fuel or safety-related equipment with the potential for excessive offsite releases, inadvertent criticality, loss of water inventory in the reactor or spent fuel pool, or loss of safe-shutdown equipment. In 1980, the NRC issued to all licensees of operating reactors Generic Letter 80-113, [1] "Control of Heavy Loads," which required them to review controls for handling heavy loads to determine the extent to which the guidelines of NUREG-0612, [2] "Control of Heavy Loads at Nuclear Power Plants," were satisfied in the plant and to identify necessary changes and modifications.

The safety evaluation of each activity at TMI-2 that required the handling of heavy loads included a NUREG-0612 evaluation of postulated heavy load drops. The evaluation included a definition of load handling areas and demonstration that the effects of load drops in these areas would not reduce the margin of safety being maintained or create the potential for a criticality event.

The analysis of a heavy load drop in the containment building assumed that postulated load drops would result in the local failure of floors. Evaluations were done to ensure that the postulated failure would not result in draining the reactor vessel below the 314-foot elevation

[a] The mini hot cell was a small, shielded transfer cask used to remove and install the shield plugs from the top of the seven canister holding cavities in the shipping cask.

(bottom of the hot-leg outlet nozzle) or disabling all makeup paths to the reactor pressure vessel.

Two generic safety evaluations were submitted, approved, and revised for heavy load handling in the containment building and heavy load handling over and inside the reactor vessel. The safety evaluation for each cleanup activity that was evaluated for load drops usually referred to one of the two generic or previously issued SERs. This chapter summarizes the licensee's and the NRC's safety evaluations associated with the handling of heavy loads.

7.1.2 Chapter Contents

This chapter presents load drop safety evaluations of the postaccident TMI-2 and defueling operations. It describes the many studies performed to help the reader understand the thinking of the analysts at the time, the expectations and the reality, uncertainties in the data, and measurement and mitigation methods. It also presents a high-level timeline of cleanup activities.

The evaluations presented in this chapter ensured that all activities involving the movement of heavy loads inside the containment building or directly over the reactor vessel were addressed and consequences evaluated; controls were maintained in accordance with the requirements of the plant's license, technical specifications, procedures, and applicable regulatory requirements; and adequate contingencies were developed for normal and accident conditions.

Key activities of concern with load drops included: (●) reactor vessel underhead characterization (sample cask removal); (●) core (bore) stratification sample acquisition; (●) polar crane load test; (●) reactor vessel head removal; (●) plenum assembly removal; (●) movement of heavy loads in the containment building and over the reactor vessel (load drop); (●) defueling water cleanup system filter movements; (●) defueling canister movements (storage and shipment); (●) fuel canister storage rack impacts from load drop; (●) heavy defueling tool movements; (●) defueling operations such as core region defueling, lower core support structure removal and defueling, core former baffle plates removals; and (●) others.

Additional evaluations of the impacts of load drops can be found in the chapters of this NUREG/KM supplement on criticality and boron dilution (Chapter 3), decay heat removal (Chapter 6), reactor vessel integrity (Chapter 8), occupational exposure (Chapter 9), radiological release (Chapter 10), and vital equipment protection (Chapter 11).

Section 2 summarizes the key studies that were used to support safety evaluations. The following sections present the safety evaluation for each applicable cleanup activity or system. Section 8 lists endnotes showing references cited throughout this chapter.

7.2 Key Studies

7.2.1 Control of Heavy Loads at Nuclear Power Plants

(USNRC, NUREG-0612, July 1980)

This report [b,3] summarized work performed by the NRC in the resolution of Generic Technical Activity A-36, "Control of Heavy Loads Near Spent Fuel." This activity [c] was one of the generic technical subjects designated as "unresolved safety issues" pursuant to Section 210 of the Energy Reorganization Act of 1974. The report described the technical studies and evaluations performed by the NRC, the agency's guidelines based on these studies, and the agency's plans for implementing its technical guidelines.

Section 2 of that report discussed the potential for an accidental load drop that could impact nuclear fuel or safety-related equipment with the potential for excessive offsite releases, inadvertent criticality, loss of water inventory in the reactor (or spent fuel pool), or loss of safe-shutdown equipment.

The analysis of potential consequences of a heavy load drop onto spent fuel assemblies contained in Section 2 was based primarily on the methods and assumptions used for fuel handling accidents as shown in the NRC Standard Review Plan, Section 15.7.4, [3] "Radiological Consequences of Fuel Handling Accidents," and Regulatory Guide 1.25, [d] "Assumptions Used for Evaluating the Potential Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors."

Section 5 described various alternative approaches that provided acceptable measures for the control of heavy loads. The objectives of these guidelines were to ensure that either (1) the potential for a load drop was extremely small, or (2) for each area addressed, the following evaluation criteria were satisfied:

> *Criterion I*: Releases of radioactive material that could result from damage to spent fuel, based on calculations involving an accidental drop of a postulated heavy load, produced

---

[b] Editor's Note: In October 2005, the NRC issued Regulatory Issue Summary (RIS) 2005-25, "Clarification of NRC Guidelines for Control of Heavy Loads," as a result of recommendations developed through Generic Issue 186, "Potential Risk and Consequence of Heavy Load Drops in Nuclear Power Plants," and findings from the NRC inspection program. This RIS reemphasized the need to follow NUREG-0612 guidance, which addressed good practices for crane operations and load movements. Attachment 1 to this RIS described the application of insights gained from operating experience and inspection to the guidelines of NUREG-0612. The attachment also clarified the guidelines where operating experience or inspection results indicated further explanation was necessary.

[c] Editor's Note: Generic Technical Activity A-36 is described in NUREG-0933, "Resolution of Generic Safety Issues" (available at nrc.gov).

[d] Editor's Note: Regulatory Guide (RG) 1.25 was withdrawn in 2016. The guidance was updated and incorporated into RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Plants, and RG 1.195, "Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors." The information in RG 1.183 (issued July 2000) provides guidance for new and existing light-water reactor (LWR) plants that have adopted the alternative source term, and RG 1.195 (issued May 2003) provides guidance for those LWR plants that have not adopted the alternative source term.

doses that were well within 10 CFR Part 100, "Reactor Site Criteria," limits of 300 rem thyroid and 25 rem whole body (analyses should show that doses were equal to or less than 25 percent of 10 CFR Part 100 limits).

Criterion II: Damage to fuel and fuel storage racks, based on calculations involving an accidental drop of a postulated heavy load, did not result in a configuration of the fuel such that effective neutron multiplication was larger than 0.95.

Criterion III: Damage to the reactor vessel or the spent fuel pool based on calculations of damage following an accidental dropping of a postulated heavy load was limited so as not to result in water leakage that could uncover the fuel (makeup water provided to overcome leakage should be from borated source of adequate concentration if the water being lost was borated).

Criterion IV: Damage to equipment in redundant or dual safe-shutdown paths, based on calculations assuming an accidental drop of a postulated heavy load, would be limited so as not to result in loss of required safe-shutdown functions.

Section 5 also provided an overall philosophy for a defense-in-depth approach for controlling the handling of heavy loads. This philosophy encompassed the intent to prevent, as well as to mitigate, the consequences of postulated accidental load drops. This defense-in-depth approach included: (1) providing sufficient operator training, handling system design, load handling instructions, and equipment inspection to ensure reliable operation of the handling system; (2) defining safe load travel paths through procedures and operator training so that heavy loads were not carried over or near irradiated fuel or safe-shutdown equipment; and (3) providing mechanical stops or electrical interlocks to prevent movement of heavy loads over irradiated fuel or in proximity to equipment associated with redundant shutdown paths. The guidance also provided alternative measures that could be taken to compensate for deficiencies in approaches (2) and (3).

The guidelines explained how the above defense-in-depth approach could be satisfied for various plant areas, such as spent fuel pool areas, the containment building, and other areas, where safe-shutdown equipment could be damaged. Guidance included: (●) identification of safe load paths; (●) use of load handling procedures; (●) training of crane operators; (●) use of slings and special lifting devices; and (●) periodic inspection and maintenance for the crane. Alternatives to this guidance included: (●) use of a single-failure-proof handling system; (●) use of mechanical stops or electrical interlocks to keep heavy loads away from fuel or safe-shutdown equipment, or (●) analysis of the consequences of postulated heavy load drops to show that these were within acceptable limits.

Appendix A to the report provided guidance for the analysis of alternatives in Section 5.

Certain alternatives required an analysis of postulated load drops and evaluation of potential consequences to ensure that the evaluation criteria (I–IV) were met for such an event.

The guidance identified considerations and assumptions that should be used in analyzing the potential consequences of a drop of the reactor vessel head assembly, spent fuel shipping cask, or load drops. Guidance was also provided for performing criticality calculations.

7.2.2 Safety Evaluation Report for Heavy Load Handling inside Containment
(GPU Nuclear, November 1, 1984, as revised)

This safety evaluation report (SER) [(4)] provided a NUREG-0612 evaluation of postulated heavy load drops, including a definition of load handling areas and demonstration that the effects of load drops in these areas would not reduce the margin of safety being maintained or create the potential for a criticality event within the containment.

This SER addressed the handling of heavy loads within the containment, identification of load handling areas, and any necessary restrictions on handling these loads. The areas above the in-core instrument seal plate, the reactor vessel, and the northwest corner of the "A" D-ring [(e)] were identified as exclusion areas where heavy loads were not to be handled without the specific approval of the NRC, in accordance with the recovery technical specification.

In addition to the three exclusion areas, the deep end of the fuel transfer canal was an exclusion area when either of the following two conditions existed: (●) fuel-filled canisters were present in the deep end of the fuel transfer canal; or (●) fuel-filled canisters were present in the fuel handling building fuel pool and one or both of the fuel transfer tubes were open.

On a case-by-case basis before the load handling operation, the NRC would evaluate and approve loads to be handled over the three exclusion areas or over the deep end of the fuel transfer canal when either of the above two conditions existed. Additionally, the handling of canisters filled with fuel was outside the scope of this SER and would be treated in separate SERs. Since this SER did not address specific loads and load handling operations, offsite releases were addressed only generically in this SER. Rather than addressing specific load paths, this SER addressed an entire area (e.g., D-rings, hatch area, fuel transfer canal, or floor slab) as the area subject to the load drop. This SER also addressed reactivity control and radiological consequences of a heavy load drop into the reactor vessel.

The results presented in this SER were based on evaluations of design drawings and calculations that determined the structural response of and local damage to floor slabs and hatch covers.

This SER was revised at least three times and was most often cited in other safety evaluations of cleanup activities. This NUREG/KM chapter provides details of this SER.

---

[e] D-rings were shield enclosures around the steam generator compartments; they were so named because of their shape.

7.2.3 Safety Evaluation Report for Heavy Load Handling over Reactor Vessel
(GPU Nuclear, April 19, 1985, as revised)

The purpose of this safety evaluation report (SER) [5] was to demonstrate that heavy load handling activities directly above and within the reactor vessel through to the completion of reactor fuel removal could be performed without presenting undue risk to the health and safety of the public. This SER was the controlling document for all heavy load handling activities occurring above or in the reactor vessel, including plenum lift.

The scope of this SER included the handling of heavy loads (loads greater than 2400 pounds including rigging weight) in the vicinity of the reactor vessel. This included any load handling activity that could result in a heavy load drop onto or into the vessel either directly or indirectly (by causing the collapse of structures or equipment installed over the vessel). This SER addressed all such load handling activities through to the completion of reactor vessel fuel removal activities but excluded removal of the core support assembly.

This SER addressed the potential impact of heavy load handling activities on the integrity of the reactor coolant system; it did not address the potential damage to the item dropped or the consequences of that damage (e.g., damage to a dropped defueling canister and the consequences of canister damage were not addressed). This SER also addressed reactivity control and radiological consequences of a heavy load drop into the reactor vessel.

The SER for heavy load handling in the containment building and the SER for plenum lift and transfer addressed heavy load handling activities outside the area over the reactor vessel.

This SER was revised at least three times and was most often cited in other safety evaluations related to defueling activities. This NUREG/KM chapter provides details of this SER.

7.3 Data Collection Activities

7.3.1 Axial Power Shaping Rod Insertion Test (NA)

7.3.2 Camera Insertion through Reactor Vessel Leadscrew Opening (NA)

7.3.3 Reactor Vessel Underhead Characterization (Radiation Characterization) (NA)

7.3.4 Reactor Vessel Underhead Characterization (Core Sampling)

- ***Purpose***. To obtain up to six specimens of the core debris using specially designed tools. These tools would be lowered into the reactor to extract samples of the loose debris and to load the samples into small, shielded casks for offsite shipment and analysis. The analysis of the samples would: (●) identify the composition of the particulate core debris; (●) determine particle size; (●) determine fission product content; (●) determine fission product leachability from the debris; (●) analyze the drying properties of the debris; and (●) determine whether pyrophoric materials existed in the core debris. This examination was a follow-on activity to the underhead characterization study.

- ***Background***. Three shipping casks [f] were considered [6] for transporting the core debris samples to the laboratory. The selected cask was the modified and recertified Model CNS 1-13C Type B shipping cask, with the sample in a Specification 2R container. An earlier version of this cask was used to ship spent submerged demineralizer system liners. [7]

- ***Evaluation: Load Drop (Sample Handling)*** [8] The licensee's safety evaluation noted that after the specimens were retrieved from the debris bed, they would be left in their respective casks on top of the control rod drive mechanism service structure or individually moved to the refueling deck floor level inside the containment building (347-foot elevation) by means of the 5-ton hoist. After all specimens were obtained, the six casks would be moved to the 305-foot elevation (entry level) at the entrance to the airlock. The casks and other hardware would be transported by means of the containment building crane (if missile shields over the reactor vessel were removed) or a combination of the 5-ton crane attachment, the missile shield trolley and hoist, and a two-person carrying team.

If the missile shields were moved, the debris sample casks would be transferred from the control rod drive mechanism service structure manually by a two-person carrying team or by rigging from the 5-ton hoist. The most difficult traverse in moving the cask to the 347-foot evaluation (operating level) would be to and from the catwalk above the fuel pool if the 5-ton crane could not be used. During this traverse, the cask could fall from the catwalk to either the service structure or the 347-foot elevation. In either case, the cask might drop about 10 feet.

A cask drop test was performed with the cask loaded with simulated particulate core debris from a height of 10 feet onto a concrete surface. The cask was in a polybag. After the cask was dropped, no debris was observable in the polybag. Consequently, the evaluation assumed that the cask would not release any significant radioactive material if the cask were dropped in the containment building.

For movement over the shallow end of the refueling canal, the floor slab could withstand a 250-pound load drop, provided that the dropped object had a contact area with equivalent diameter greater than 1 inch and that the lift height was restricted to 110 feet. For movement over the 347-foot elevation, the concrete floor slab could withstand a 250-pound load drop, provided that the dropped object had an equivalent diameter greater than 1 inch and that the lift height was restricted to 85 feet. For movement over the 305-foot elevation (entry level), the 3-foot-thick concrete slab west of the hatch could withstand a 250-pound load drop, provided that the dropped object had a contact area with an equivalent diameter greater than 1 inch and that the lift height was restricted to 120 feet.

The postulated drop of a cask would lie within these boundaries because of local deformation of both the dropped object and the concrete slabs. Further, a qualitative evaluation of the relative energies of the calculated enveloping cases, with respect to the energy available from a drop of

[f] Editor's Note: While large shipping containers of radioactive materials may often be referred to as "shipping casks," the proper term for such containers, when loaded with contents and in their transportation configuration, is "package." See 10 CFR Part 71, "Packaging and Transportation of Radioactive Material," Section 71.4, "Definitions."

the cask onto the concrete floor slabs, led to the conclusion that the integrated damage potential would be well within the calculated bounds for structural failure. Therefore, a postulated drop of a cask would not pose an unacceptable risk.

- ***NRC Review***. [9] The NRC's safety evaluation report did not specifically address this topic.

7.3.5 Core Stratification Sample Acquisition

- ***Purpose***. To perform the activities associated with: (●) installation, operation, and removal of the core boring machine; (●) acquisition of the core samples; (●) transfer of the samples from the machine to the defueling canisters; and (●) viewing of the lower vessel region through the bored holes.

- ***Evaluation: Load Drop***. [10] The licensee's safety evaluation stated that load handling activities during equipment installation, operation, and removal would be performed in accordance with the safety evaluation report (SER) [11] for heavy load handling in the containment building and with the SER [12] for heavy load handling over the reactor vessel. The analyses presented in these two SERs demonstrated that any potential drop accidents associated with the core sample acquisition activities would not impact the health and safety of the public.

During the installation and removal of the core boring equipment, some loads to be handled were identified to exceed the height and weight limitations presented in the SER for heavy load handling over the reactor vessel. An analysis was performed to bound all heavy load handling during the installation and removal activities associated with the core drilling equipment. The maximum load and lift height mentioned in the SER included a 5000-pound load raised to a plant elevation of 339 feet. The analysis determined that this lift height was allowable if the following conditions were met: (●) The load was maintained within a limit of 3 feet 6 inches from either side of the defueling work platform support structure centerline. (●) The T-slot tool positioner, tool rack, jib cranes, and transfer shield-shield collar were not on the platform during this load handling activity. (●) The core bore drill rig, flush water tank, and hydraulic control assembly were installed before this load handling activity.

A load less than 2000 pounds that would be handled above the defueling work platform would be handled in accordance with the following equation: [$H = (17500/W) + 331.5$], where H was the maximum plant elevation to which the load could be raised (in feet), W was the weight of the lifted load including the weight of the rigging that was rigidly attached to the load (in pounds), and H was the maximum elevation of the lowest rigid point of the suspended load.

Any additional load handling activities that were identified to exceed the limitations of the two SERs mentioned above would be evaluated on a case-by-case basis.

- ***NRC Review***. (13) The NRC's SER did not specifically address this topic.

7.3.6 Use of Metal Disintegration Machining System to Cut Reactor Vessel Lower Head Wall Samples

- ***Purpose***. To remove metallurgical samples from the inner surface of the lower reactor vessel head using the metal disintegration machining system. An international research program used these samples to study core melt and reactor vessel interactions. This program was conducted after the reactor vessel was defueled.

- ***Evaluation: Load Drop***. (14, 15) The licensee's safety evaluation stated that the previous safety evaluation reports (16, 17) bounded the issues concerning load drops in the reactor vessel until the time when the samples were removed from the reactor vessel. All lifting and handling conformed to the TMI-2 lifting and handling program for samples, containers, and tools. In addition, lifting and handling loads were expected to be less than loads that were experienced during lower core support structure disassembly and removal. For sample locations away from in-core nozzles, calculations indicated that the remaining 2 inches of reactor vessel wall thickness would withstand postulated load drops.

- ***NRC Review: Load Drop***. (18) The NRC's safety evaluation noted that the licensee and the NRC had previously evaluated a wide range of activities that included load drops that could cause leakage associated with in-core instrument penetrations. These evaluations (19, 20) were associated with reactor vessel lower head defueling and previous defueling activities. Leakage through the annular gap between an instrument tube and the reactor vessel wall could produce a leak of 0.4 gallon per minute for each penetration. This leakage would result from the case in which an in-core instrument penetration and its weld were sheared off, but the instrument tube remained in the hole in the reactor vessel wall.

Another set of analyses evaluated the case of an additional unspecified mechanism forcing the instrument tube out of the vessel wall. This would result in a 1-inch-diameter hole and a leak of 120 gallons per minute. The licensee had safety systems to make up this potential leakage. These systems included gravity feed from the borated water storage tank and forced circulation via the containment building recirculation pumps. The cavity under the reactor vessel contained borated water to preclude criticality in the event that any fuel was flushed down with the leaking water.

7.4 Pre-Defueling Preparations

7.4.1 Containment Building Decontamination and Dose Reduction Activities

Purpose. To conduct decontamination and dose reduction activities in the containment building at elevation levels 305-feet (entry level) and above. Planned activities included: (●) flushing containment building surfaces with deborated water; (●) scrubbing selected components of the polar crane; (●) conducting hands-on decontamination of vertical surfaces; (●) decontaminating the air coolers; (●) flushing the elevator pit; (●) cleaning floor drains; (●) shielding the seismic gap and penetration; (●) decontaminating and shielding hot spots; (●) decontaminating missile shielding; (●) shielding reactor vessel head surface structure; (●) removing concrete and paint; (●) decontaminating cable trays; (●) decontaminating equipment; (●) remote flushing the containment building basement; and (●) testing remote decontamination technology.

- ***Evaluation: Load Drop***. [(21)] The licensee's safety evaluation stated that decontamination and dose reduction activities could necessitate the movement of heavy loads at various locations in the containment building. The licensee anticipated that these loads could include shielding and decontamination equipment. These loads would be handled according to the safety evaluation report for handling heavy loads in the containment building, or they would be evaluated on a case-by-case basis and be subject to NRC approval.

- ***NRC Review***. [(22)] The NRC's safety evaluation report did not specifically address this topic.

7.4.2 Reactor Coolant System Refill (NA)

7.4.3 Reactor Vessel Head Removal Operations

7.4.3.1 Polar Crane Load Test

- ***Purpose***. To conduct load testing of the polar crane in preparation for reactor vessel head lift and removal. The polar crane load test required four major sequential activities: (1) relocation of the internals indexing fixture; (2) assembly of test load; (3) load test; and (4) disassembly of test load.

- ***Evaluation: Load Drop (Prevention)***. [(23)] The licensee's safety evaluation considered many risk reduction features in the crane design and rigging and administrative controls. These risk reduction measures, as described below, taken in conjunction with the refurbished condition of the crane as described in the safety evaluation, led to the conclusion that the probability of a load drop was so small that it approached the incredible drop probability of the so-called single-failure-proof cranes.

  - *Risk Reduction Considerations*. The evaluation noted that the TMI-2 polar crane was not a single-failure-proof crane, as defined in NUREG-0554, "Single-Failure-Proof Cranes for Nuclear Power Plants," [(24)] issued May 1979. However, the evaluation approached the particular case of a missile shield drop from a more mechanistic standpoint and concluded that the probability was extremely small because of the following factors: (●) The polar crane factor of safety was greater than 10 compared to the original design rating and about 5 compared to the requalification rating. (●) Missile shield blocks were moved sequentially starting with the one farthest from the load test frame using the intervening blocks to protect the reactor. (●) Shield block lift rigging incorporated significant conservatism (for example, the missile shield block lifting pad eyes had a safety factor of 3 to the ultimate breaking point based on only two cables supporting the load (instead of four), and the cable lift angle with the vertical would not exceed 30 degrees. (●) The amount of time when these loads were lifted in the vicinity of the reactor vessel would be minimized.

In the case of the test load, the safety evaluation made similar arguments about the conservatism of the crane and rigging capacities. However, since the test load was about 5 times greater than the load of a missile shield, the following additional conservatisms were introduced: (●) The test load area was carefully reviewed and selected on the basis that a minimum amount of equipment was located directly beneath it in comparison to other areas of the containment. (●) The lifting time of the test load would be minimized to the greatest extent compatible with crane requalification.

- *Procedure Controls*. Procedural conservatisms were introduced. For example, the initial loads to be lifted by the polar crane were lighter than later loads. The 6-ton internals indexing fixture (IIF) was lifted before the movement of a missile shield. Each time a load to be lifted was heavier than any previously lifted load, the procedure required a new load to be lifted in steps to ensure that, should a failure occur, consequences would be minimized. The procedure required that the new load be lifted initially only a small distance and held in place to verify that no problems were encountered, and then the lift would be completed. An example of this was the initial lift of a reactor vessel missile shield. The missile shield was lifted only a small distance and held in place while still on the guide studs. A load drop of the missile shield would have no unacceptable consequence as it would merely settle back into place on the D-rings across the refueling canal.

- *Movement Control*. No reliance was placed on the installation and use of electrical interlocks or mechanical stops to keep the load in its prescribed load path. The movement of the load was controlled by the test director, who was equipped with a voice actuated headset, as were other workers in the containment building who were associated with the load test. The test director could communicate with a person stationed by the main power supply breaker in the auxiliary and fuel handling building. In case of an emergency, the test director could have the main power supply interrupted. This would freeze the crane in the position it was in when the power was cut off.

- ***Evaluation: Load Drop (Regulatory Requirements)***. [25] The licensee's safety evaluation provided a detailed analysis that examined the potential consequences of load drops in the vicinity of important equipment. These analyses encompassed all equipment in the load path down to the individual valves and instrument lines. Heavy load drop analysis was performed in accordance with the regulatory requirements on the control of heavy loads that mandated licensees to address the guidelines of NUREG-0612. These guidelines required the following: (●) In the vicinity of the reactor core, demonstrate that adequate measures were taken to ensure that either the likelihood of a load drop that could damage spent fuel was extremely small or the estimated consequences of such a drop would not exceed the limits set by the evaluation Criteria I through III of NUREG-0612, Section 5.1. (●) In the vicinity of equipment or components required for safe reactor shutdown (subcriticality) and decay heat removal, demonstrate that either the likelihood of a load drop that could prevent safe reactor shutdown or prohibit continued decay heat removal was extremely small or that damage to such equipment would be limited so as not to result in the loss of these safety-related functions (Criterion IV).

The NUREG-0612 guidance described various alternative approaches that provided acceptable measures for the control of heavy loads. The objectives of these guidelines ensured that either: (1) the potential for a load drop was extremely small, or (2) for each area addressed, the following evaluation criteria were satisfied:

> *Criterion I*: Releases of radioactive material that could result from damage to spent fuel, based on calculations involving an accidental drop of a postulated heavy load, produced doses that were well within the 10 CFR Part 100 limits of 300 rem thyroid and 25 rem whole body (analyses should show that doses were equal to or less than 25 percent of 10 CFR Part 100 limits).
>
> *Criterion II*: Damage to fuel and fuel storage racks, based on calculations involving an accidental drop of a postulated heavy load, did not result in a configuration of the fuel such that effective neutron multiplication (k_{eff}) was larger than 0.95.
>
> *Criterion III*: Damage to the reactor vessel or the spent fuel pool, based on calculations involving an accidental drop of a postulated heavy load, was limited so as not to result in water leakage that could uncover the fuel (makeup water provided to overcome leakage should be from a borated source of adequate concentration if the water being lost was borated).
>
> *Criterion IV*: Damage to equipment in redundant or dual safe-shutdown paths, based on calculations involving an accidental drop of a postulated heavy load, would be limited so as not to result in loss of required safe-shutdown functions.

Each evaluation within a critical area in the containment building included the identification of loads, targets, and load/target interactions. The effects of a heavy load drop in areas where heavy loads were expected to be moved were analyzed to ascertain the worst credible consequence. The consequences of the loss for each component within the designated areas were determined and presented in the evaluation. The evaluations addressed estimated consequences of such a drop to ensure that they did not exceed the limits set by the evaluation criteria of NUREG-0612.

- ***Evaluation: Load Drop (Core Vicinity)***. [26] The safety evaluation of a heavy load drop in the vicinity of the reactor core included the identification of loads, targets, and load/target interactions, as well as the evaluation of the worst credible consequence. The evaluation showed that Criteria I though III of NUREG-0612 were met.

  - *Identification of Loads*. For the performance of the polar crane load test, the loads to be moved in the vicinity of the reactor core were the reactor vessel missile shields. These missile shields were constructed in the shape of oblong blocks of concrete and rebar, which weighed about 40 tons each. The safety evaluation report (SER) provided figures that showed the lift paths.

- *Load/Target Interaction*. In the event that a shield block was to fall onto the reactor head and service structure, damage to the control rod drive mechanism motor tubes would cause leakage of reactor coolant into the containment building. Since penetration of the 8-inch-thick steel closure head was not credible, the maximum leakage would occur if the reactor coolant system (RCS) drained to the top of the level of the reactor vessel closure head. A shield block striking the reactor vessel closure head could also bring about some physical redistribution of loose core debris within the RCS. The evaluation noted that much of the kinetic energy of the falling shield block would be absorbed in physical deflection of the tall service structure and control rod drive mechanism apparatus above the head and that instantaneous impact directly on the head would not occur.

- *Evaluation: Release of Radioactivity (Criterion I)*. The impact of a missile shield block dropping onto the reactor vessel head and service structure could cause leakage of reactor coolant through the control rod drive mechanism motor tubes into the containment building. This liquid would remain in the containment building; thus, the containment building would act as a physical barrier and prevent any liquid releases from escaping to the environment. Likewise, any gaseous releases caused by this postulated drop would be physically contained since the containment building integrity would be set and maintained throughout the load test. Containment building integrity would be further ensured since there was no longer any energy source capable of producing a driving pressure that could transport this activity across the containment building boundary. Any gaseous activity released in the containment building would be directed through the high-efficiency particulate air (HEPA) filters and containment building purge exhaust system. Gaseous activity would eventually be released in a controlled manner; therefore, the release would not exceed the limits established in Criterion 1. Further, any releases that could occur in spite of the factors presented above would be only a small fraction of the calculated release presented for a loss-of-coolant accident in Chapter 15 of the preaccident TMI-2 final safety analysis report, thus meeting Criterion I.

- *Evaluation: Criticality (Criterion II)*. The licensee's evaluation considered criticality in the reactor vessel and in the steam generators.

  - *Reactor Vessel*. The precise configuration of the fuel before head removal was unknown at the time of this analysis. Therefore, the exact k_{eff} that would result from the potential redistribution of the fuel due to the impact of a missile shield block on the reactor vessel head could not be calculated. Despite the inability to calculate the exact k_{eff}, bounding analyses [27, 28] were performed for the 100-percent fuel damage case. These analyses concluded that the fuel debris would not be critical when the debris was in its most reactive condition. The analyses accounted for the effects of structural material and a reactor coolant boron concentration of 3000 parts per million (ppm). In view of these results and because the concentration of boron in the reactor coolant was over 3500 ppm, the evaluation concluded that a criticality was precluded.

  - *Steam Generator*. The evaluation determined that a sequential series of low-probability events would need to occur before criticality in the steam generators could be viewed as a legitimate safety issue. The following sequence of low-probability events would have to occur in sequence, each conditioned on the occurrence of all the prior low-probability events, before a criticality in the steam generator could occur: (1) A missile shield must be dropped above one of the D-rings. (2) The missile shield must travel far enough into the D-rings to impact a reactor coolant pump or cold-leg piping. Note that there were massive structural beams crossing the D-ring above the reactor coolant pump elevation where the reactor coolant pumps were vertically supported. (3) The missile shield must impact the reactor coolant pump or other structure in such a way as to rupture the pump suction line at a point well below the secondary-side water level. (4) An amount of fuel sufficient to raise criticality concerns must be transferred to the steam generator during the accident, migrate into the steam generator tubes, and lodge in the tubes. (5) This fuel must be in a high-density, close-pack critical configuration inside the tubes with borated water drained from the tubes and the tubes surrounded by unborated secondary water.

Even with the application of conservative probabilities to each event in the required sequence, the evaluation concluded that the probability of criticality occurrence was below any reasonable threshold for safety concern.

- *Evaluation: Fuel Uncovery (Criterion III)*. The maximum leakage resulting from a drop of a missile shield block onto the reactor head and service structure would be drainage of the RCS to the level of the top of the reactor vessel closure head. Drainage to this level would not uncover the fuel; therefore, Criterion III would be met. In addition, at least one makeup train capable of delivering water with a 3500-ppm boron concentration to the reactor vessel would be available.

- ***Evaluation: Load Drop (Vicinity of Important Equipment)***. [29] The licensee's safety evaluation of load drops in the vicinity of safe-shutdown and decay heat removal equipment (or other equipment specifically important at TMI-2 because of unique site considerations) included the identification of loads, targets, and load/target interactions, as well as the evaluation of the worst credible consequence. The evaluation showed that Criterion IV of NUREG-0612 was met.

- *Identification of Loads*. The polar crane test loads that would be moved in the vicinity of important equipment included: (●) the IIF; (●) the missile shields, including the 32-ton pressurizer shield; and (●) the load test assembly composed of the missile shields stacked on a storage frame. The indexing fixture was not specifically addressed in the analyses because the results of a drop of this item were enveloped and bounded by the missile shield loads. The assembled test load was specifically addressed.

- *Identification of Targets*. The SER listed the systems and components, including valves and instrumentation, that were considered essential functions. The selection criteria included: (●) equipment within the reactor coolant pressure boundary required for decay heat removal and reactivity control; (●) equipment required to be operable by the recovery technical specifications; and (●) equipment required by plant procedures, which were approved in accordance with recovery technical specifications.

Major fluid systems that were examined for possible targets included: (●) reactor coolant; (●) makeup and purification; (●) decay heat removal; (●) mini-decay heat removal; (●) standby reactor coolant pressure control system; (●) core flood; (●) decay heat closed cooling water; (●) containment building spray; (●) chemical addition; (●) nuclear services closed cooling water; (●) feedwater and condensate; (●) main steam; (●) demineralized service water; (●) intermediate closed cooling water; (●) nuclear services river water; (●) containment building ventilation; (●) containment building purge; (●) fire protection; and (●) steam generator secondary-side vent and drains.

- *Load/Target Interactions*. The effects of a heavy load drop in areas where heavy loads were expected to be moved were analyzed to ascertain the worst credible consequence. The evaluation assumed that the top and middle floors would locally collapse when impacted by the dropped load. The evaluation included a matrix of the impact areas and load/equipment combinations. Components were not considered to be functional after a heavy load drop, which was assumed to occur directly over the components. (The availability of an unaffected alternative was ascertained in view of why the component was required. A component was considered to be an alternative only if it performed the same safe-shutdown function as the component subjected to a heavy load drop.) The location of each piece of essential equipment was determined from the drawings available at the time. The consequences of the loss of each component within the designated areas were further evaluated.

- *Evaluation: Damage to Equipment (Criterion IV)*. Criterion IV of NUREG-0612 defined the required safe-shutdown functions as those needed to: (●) maintain the reactor coolant pressure boundary; (●) reach and maintain subcriticality; (●) remove decay heat; and (●) maintain the integrity of components whose failures could result in excessive offsite releases. The required functions that applied to TMI-2 in the cooling mode and core configuration at the time of the load tests included: (●) capability to maintain subcriticality; (●) decay heat removal; and (●) capability to maintain the integrity of components whose failures could result in excessive offsite releases.

  - *Criticality (Reactor Core)*. Because of the configuration of TMI-2, the only credible mechanisms by which criticality control could be compromised were determined to be deboration of the water in the RCS. Systems within load impact areas that contained unborated water were investigated and found to fail in such a way as to drain their contents onto the containment building floor and not into the RCS. For example, the boron concentration of the RCS could be reduced by gross leakage from the unborated secondary side into the primary side as a result of a postulated load drop. Damage to the steam generators severe enough to cause such leakage would cause damage to the outer surface of the steam generator, allowing the unborated water to drain to the containment sump. Further, systems capable of injecting highly borated water into the RCS were available, and it was not feasible with one load drop to reduce the functional capability of these systems to a point that boron injection could not function.

  - *Criticality (Steam Generator)*. Before a criticality in the steam generator could occur, the following sequence of low-probability events would have to occur in sequence in the steam generators, each conditioned on the occurrence of all the prior low-probability events: (1) A missile shield must be dropped above one of the D-rings. (2) The missile shield must travel far enough into the D-rings to impact a reactor coolant pump or cold-leg piping. (Note: There were massive structural beams crossing the D-ring above the reactor coolant pump elevation where the reactor coolant pumps were vertically supported.) (3) The missile shield must impact the reactor coolant pump or other structure in such a way as to rupture the pump suction line at a point well below the secondary-side water level. (4) An amount of fuel sufficient to raise criticality concerns would have to be transferred to the steam generator and the steam generator tubes during the TMI-2 accident. (5) Fuel must be in a high-density, close-pack configuration within the tubes in a manner that would allow criticality if the borated water were drained from the tubes while the fuel was surrounded by unborated secondary water.

– *Criticality (Containment Building Sump)*. The only potential of a criticality in the containment building sump would be from the drop of a heavy load onto systems that could provide a source of unborated water to the sump. To eliminate the potential of a criticality due to a boron dilution accident, the water supply to these systems would be isolated while the load test was being performed. (Note: The operable fire protection system was nominally isolated.) An evaluation showed that a significant quantity of unborated water would be required to lower the sump water concentration from current values to a level below the 1700-ppm value, which was specified as the reasonable point to avoid a sump reactivity problem.

  Further, several low-probability events would be required for a potential sump criticality, regardless of the amount of unborated water delivered to the sump. First, a sufficient amount of fuel to create a critical mass would have had to wash into the sump during the TMI-2 accident. Second, this fuel must be in a configuration that could induce criticality if a global boron dilution of the sump water were to occur. A qualitative assessment concluded that this issue could be eliminated as a legitimate safety concern based on the combination of the: (●) administrative controls in place at the time of the load test that would limit the amount of unborated water that could be delivered to the sump; and (●) low probability of simultaneous occurrence of the initial conditions for fuel deposition in the sump that could lead to a criticality problem.

– *Decay Heat Removal*. Decay heat removal capability was ensured by maintaining water in the reactor vessel. Analysis showed that the water could be drained to the bottom of the cold-leg nozzles with no adverse consequences, such as boiling. The only way to drain the vessel below this level would be through damage to the in-core instrument tubes. In addition, if damage to the RCS causing leakage of reactor coolant were to occur, makeup capability existed at least through one loop since a decay heat removal system suction line (drop line) in both D-rings simultaneously was not credible by physical separation. Makeup system penetrations would not be damaged since they were located on the north side of the containment building away from the load paths.

- *Pressure Boundary Integrity*. The reactor coolant pressure boundary needed to be maintained only insofar as reactor coolant had to be maintained in the reactor coolant system for decay heat removal and reactivity control. Both evaluations were discussed above.

- *Damaged In-core Instrument Lines*. A special consideration was evaluated regarding the potential consequences of a load drop damaging the in-core instrument lines. The reactor vessel lower head was penetrated by in-core instrument lines. These lines ran from beneath the bottom head of the vessel through a tunnel in the containment building base mat to terminations at the seal table area. The routing of these lines paralleled a line drawn between the center of the reactor vessel and the center of the seal table. The width of the area occupied by these lines was smaller than the diameter of the seal table. The seal area was not located within the load path. However, a portion of the in-core instrument lines was physically located below the area of the load path for the reactor missile shield blocks. This portion of the lines was separated vertically from load impact surfaces by concrete and steel structures of such massive proportion as to render load penetration impossible.

  The only remaining scenario in which a dropped load could damage an in-core instrument line included the following sequence: (1) The missile shield must be dropped into the refueling canal. (2) The shield block must be oriented and reconfigured itself to fit between the reactor vessel and primary shield wall (present dimensions precluded such an event). (3) The shield block must travel down to the elevation of the reactor vessel skirt, disintegrate into pieces small enough to fit through holes in the reactor vessel skirt, and travel horizontally far enough and with sufficient remaining energy to damage the stainless-steel in-core lines. This scenario was judged not credible based on the improbability of these three occurrences, especially the second, which violated the physical laws of nature.

- *Offsite Radiological Release*. The containment building pressure boundary prevented an offsite release. The containment building integrity, as required by the recovery technical specifications, was established during the load test. All containment penetrations that could be damaged in the event of a load drop were isolated outside the containment building.

- ***NRC Review: Load Drop***. [30] The NRC reviewed the entire load test sequence and considered the potential for accidents in relation to the required lifts and pathways selected for lift movement. The NRC's safety evaluation included a review of the heavy load drop analysis provided by the licensee to address the guidance in NUREG-0612. The guidance in NUREG-0612 was developed to address the concerns related to the dropping of heavy loads in certain locations in the plant and impacting stored spent fuel or fuel in the core, equipment required to achieve safe shutdown, or equipment to remove decay heat from the core. While these were valid concerns at normal operating plants, in its SER, the NRC indicated that they were of less concern at TMI-2 for the reasons discussed below.

- *Safe Shutdown*. The TMI-2 facility was already in a safe-shutdown condition; therefore, there were no concerns about a potential drop impacting the capability to achieve safe shutdown.

- *Decay Heat Removal*. The reactor had been shut down for about 4.5 years, and the decay heat generation had decayed to a level of about 24 kilowatts, roughly the heat generated by 25 household toasters. Decay heat was being removed by purely passive means (losses to ambient air in the containment building); therefore, the potential loss of an active means of removing decay heat from the core, as a result of a heavy load drop accident, was determined not to be a serious concern.

- *Radiological Release*. No spent fuel was stored in the refueling canal; thus, there was no potential for a drop accident to impact exposed fuel assemblies outside the reactor vessel. However, there was the potential, even though the probability was very low, for dropping a missile shield on the reactor vessel and service structure and rearranging the physical distribution of the fuel debris in the reactor vessel. Considering the potential for radioactive releases from such an event, the NRC noted that virtually all of the noble gases and iodine radionuclides had already been released from the damaged fuel assemblies in the core or had decayed to insignificant levels. Thus, there was no potential for a large release of volatile gaseous radionuclides from a drop accident. Furthermore, any generation of airborne particulate activity would be confined inside the containment building and filtered by the building ventilation system HEPA filters before release. Therefore, any potential releases would result in doses well within the limits of 10 CFR Part 100.

- *Criticality*. With regard to the potential for criticality in the core from an impact-induced fuel debris rearrangement, a number of criticality analyses were previously performed as part of the NRC's SER [31] for the axial power shaping rod insertion tests that postulated fuel redistribution. The NRC considered the crane load test to be bounded by the previous analyses.

- *Load Drop Potential*. The NRC considered the potential for a load drop to be extremely low for a number of reasons. First, the crane was originally rated for 500 tons and was refurbished with parts (e.g., brake pads) sized for the 500-ton rating. However, the maximum load to be handled in the vicinity of the reactor vessel was a single 40-ton missile shield. The cleanup required only a crane capable of lifting the 163-ton reactor vessel head and service structure. In addition, the requalification test load (about 212 tons) was less than half the related capability of the original design. Second, the environmental conditions during and following the accident were not severe enough to affect the structural integrity of the crane and related components (e.g., wire rope and tripod assembly). A detailed inspection of the crane components, including critical welds, verified the condition of the exposed elements. Finally, the crane had a demonstrated history of significant lifts, including previous lifts of the reactor vessel head and service structure and the 152-ton pressurizer.

- *Risk Reduction*. Given the potential for a severe accident related to load testing and the low probability of such an event, the licensee planned the load test to minimize the risk associated with the activity. Risk reduction measures included the following: (●) The load test sequence was structured to requalify the crane in a progressive series of steps, beginning with the 6-ton IIF, proceeding to a 40-ton missile shield, and then the 212-ton requalification test for reactor vessel head lift. (●) Each load, regardless of size, was initially lifted only a short distance and held in place while still on the guide studs to further minimize the potential for a drop on the reactor vessel head. (●) In assembling the test load, the missile shield located farthest from the test frame was moved first for subsequent transport over the remaining shields, which served to protect the reactor vessel and other equipment below in the event of a drop. (●) In general, the lifting time for all lifts was minimized to the extent necessary to complete a movement to satisfy a test. (●) Where possible, all load pathways were selected to avoid the vicinity of the reactor vessel and with consideration for the piping and components located on the elevations below the operating floor. (●) Rotation of the bridge was bounded by procedure to the azimuthal sector required to conduct the tests, and markers were placed on the containment building wall to identify the limits for bridge travel. Placement of markers was consistent with the guidance in NUREG-0612. (●) Other procedural precautions for the test included stationing an individual near the crane's main power supply breaker located in the auxiliary and fuel handling building. This individual would communicate directly with the command center. If necessary, the test director could disconnect the main breaker automatically set the brake on the main hoist.

- *Load Drop Consequences*. The NRC considered the consequences of a missile shield drop on the reactor vessel head and service structure. The worst case credible event would be the fracturing of one of the pipes (e.g., core flood tank inlet) penetrating the reactor vessel, which would result in draining a portion of the reactor coolant. However, even in this case, the reactor coolant would drain down only to the level of the pipe inlet nozzle, which was still above the core. Thus, the core would remain covered. The lost reactor coolant would collect in the containment building and would not pose significant radiological risks for several reasons. Because the reactor coolant was at ambient temperature, there was no driving force to evaporate the coolant and disperse the entrained radioactivity. The gross radionuclide concentration in the reactor coolant was less than 10 microcuries per milliliter, and there were no significant iodine radionuclides or dissolved noble gases in the reactor coolant.

7.4.3.2 First-Pass Stud Detensioning for Head Removal (NA)

7.4.3.3 Reactor Vessel Head Removal Operations

- ***Purpose***. To prepare for reactor vessel head removal to perform the lift and transfer of the head and conduct the activities necessary to place and maintain the reactor coolant system (RCS) in a stable configuration for the next phase of the recovery effort.

- ***Evaluation: Load Drop (General)***. [32] The licensee's safety evaluation considered the consequences of a heavy load drop during this activity. NRC Generic Letter 80-113 on the control of heavy loads required licensees to address the guidelines of NUREG-0612.

    - *Background*. Generic Letter 80-113 required information to sufficiently demonstrate that adequate measures would be taken to ensure that in the vicinity of the reactor core, either the likelihood of a load drop, which might damage spent fuel, was extremely small, or that the estimated consequences of such a drop would not exceed the limits set by the evaluation Criteria I–III of NUREG-0612, Section 5.1. The letter also required information to sufficiently demonstrate that in the vicinity of equipment or components required for safe reactor shutdown and decay heat removal, either the likelihood of a load drop, which might prevent safe reactor shutdown or prohibit continued decay heat removal, was extremely small, or that damage to such equipment would be limited so as not to result in the loss of these safety-related functions (Criterion IV).

    - *Identification of Heavy Loads*. Heavy loads that would be handled during the reactor vessel head assembly removal evolution included: (●) lift rigging; (●) vessel closure head; (●) control rod drive mechanism motor tube assemblies; (●) service structure; and (●) attached shielding and its support frame. Following the removal of the reactor vessel head assembly, other heavy loads that would be installed included: (●) internals indexing fixture; (●) fixture cover; and (●) cover shielding plates, if required. Of these heavy loads and any others that could be handled within the head load path during the head removal evolutions, the lifting of the 170-ton head assembly was bounding. Therefore, the consequences of a postulated drop of any other load would be less than those of a postulated drop of the head. Within the scope of this safety evaluation, load drops after the installation of the internals indexing fixture were bounded by the head drop scenarios.

    - *Approach*. To address the NUREG-0612 guidelines, the drop of the reactor vessel head was evaluated to ensure that: (●) the likelihood and consequences of a drop on the reactor vessel were acceptably small; and (●) safety-related functions were not lost due to a head drop.

      Approaching the case of a reactor vessel head drop from a more mechanistic standpoint, the evaluation concluded that the probability of dropping the head on safety-related equipment was small considering the following: (●) The polar crane factor of safety was greater than 2.5 compared to the original design rating. (●) The amount of time when the load would be lifted in the vicinity of the reactor vessel would be minimized. (●) The method for lifting the reactor vessel head was previously demonstrated, not only at TMI-2 but at other operating pressurized-water reactors as a part of normal refueling procedure. (●) Before the reactor vessel head lift, the polar crane would be tested with loads exceeding those of the reactor vessel head assembly. (●) Conservatisms were considered and introduced in the head lift procedure. For example, the lifting of the closure head would be stopped if the head tilt indication exceeded allowable tolerances or if the load lifting force exceeded the specified maximum value. Also, the procedure required that the closure head be lifted initially only a small distance and be held in place for inspection to verify that no problems were encountered before proceeding with the lift.

In addition to the above considerations, detailed analyses were conducted to examine the potential consequences of a head drop in the vicinity of safety-related equipment. These analyses encompassed all equipment in the path of the head from the reactor vessel to the head storage stand.

- ***Evaluation: Load Drop (Reactor Core)***. [33] The licensee's safety evaluation considered a heavy load drop on or near the reactor core. The evaluation identified loads, targets, and the load/target interactions, as well as the resulting worst credible consequence.

  - *Identification of Loads*. The reactor vessel closure head with the attached control rod drive mechanisms and the service structure had a combined weight of less than 170 tons. After the head lift prerequisites were completed, the polar crane would be used to first lift the head several inches above the reactor vessel flange and then to a height that cleared the guide studs and the control rod guide tubes. The head would then be moved to the south end of the refueling canal in order to clear the reactor vessel before proceeding with any further vertical lift. The exact lifting heights and path coordinates would be specified in the detailed instructions for accomplishing this task.

  - *Load/Target Interactions*. Attachment 3 of the safety evaluation report (SER) analyzed the effects of the reactor vessel head drop onto the vessel itself. These analyses showed that the reactor vessel and components could withstand the impact of the vessel head and support structure if they were dropped on the vessel flange from a height of 56.1 inches or less. The head lift, removal procedures, and detailed instructions for accomplishing this task specified that until the head cleared the vessel, the head would be lifted no higher than this maximum height. A drop of the head onto the reactor vessel could bring about physical redistribution of fuel material within the RCS. Analyses showed that subcriticality of the core would be maintained following a postulated heavy load drop (refer to Section 4.9.1.3 of Attachment 4 of this SER).

  - *Evaluation: Radiological Release (Criterion I)*. Criterion I of NUREG-0612 stated that, based on calculations involving the accidental drop of a postulated heavy load, releases of radioactive material resulting from potential damage to spent fuel would produce doses that were well within the 10 CFR Part 100 limits of 300 rem thyroid and 25 rem whole body (analyses should show that doses were equal to or less than 25 percent of 10 CFR Part 100 limits).

    The impact of the vessel head and service structure dropping onto the vessel could cause a release of airborne radioactivity into the containment building. An uncontrolled release of this activity to the environment would be prevented by the containment building integrity during head removal. For gaseous release (krypton-85), calculations showed that, even for the worst credible release, the offsite doses would be a small fraction of the 10 CFR Part 100 limits. Particulate releases would be controlled by the containment building boundary and, if released, would be processed through high-efficiency particulate air (HEPA) filters so as not to exceed 10 CFR Part 100 limits. Containment building integrity was ensured since there was no longer an energy source capable of producing a driving pressure that could transport the activity across the containment building boundary.

- *Evaluation: Criticality (Criterion II)*. [g] Criterion II of NUREG-0612 stated that damage to fuel and fuel storage racks, based on calculations involving the accidental drop of a postulated heavy load, does not result in a configuration of the fuel such that the effective neutron multiplication (k_{eff}) is larger than 0.95.

  The precise configuration of the fuel was unknown at the time of the planned head lift activity; therefore, the exact k_{eff} resulting from the potential redistribution of the fuel due to the impact of the vessel head and support structure on the reactor vessel could not be calculated. However, conservative calculations using various models of the damaged core were performed to ensure that the core would remain subcritical with sufficient poisoning of the system.

  - *Conservative Model*. The criticality analyses of the TMI-2 reactor to support the recovery activities through head removal had modeled the core assuming 50-percent cladding failure in all fuel rods. However, a model with 50-percent damage could be nonconservative if additional fuel disruptions occurred as a result of a heavy load drop accident. The heavy load drop model was conservative for criticality analyses because it assumed 12-percent additional fuel damage and an optimization of all parameters affecting core reactivity. The degree of damage (62 percent) was the maximum credible amount of cladding failure. The model assumed that the fuel was collapsed to the most reactive configuration (i.e., that all damaged Batch 3 (highest enrichment) fuel was sandwiched between the undamaged fuel on the bottom and the remaining damaged (Batch 1 and 2) fuel on the top). This separation of the damaged Batch 3 fuel from the other damaged fuel produced a higher reactivity than any homogenized mixture of all the damaged fuel. The analyses indicated that, in this conservative model, the core would remain subcritical (k_{eff} = 0.988) with a boron concentration of 3500 parts per million.

  - *Realistic Model*. A more realistic case was also analyzed. This case still assumed segregation of the damaged Batch 3 fuel, which was in a layer on top of the damaged Batch 1 and 2 fuel (i.e., the peripheral fuel assemblies collapsed on the existing rubble bed). Instead of optimizing the particle size and arrangement as was done for the conservative case, analysts used the more reasonable assumption of random particle size and distribution. The effects of structural materials were also considered, and the boron concentration was assumed to be 3700 parts per million, which was the boron concentration currently present in the RCS. For the more realistic case, the value of k_{eff} was less than 0.944. Attachment 4 of this SER provided the report on the heavy load drop criticality analyses.

[g] Editor's Note: An additional criticality evaluation was reported in the SER. Refer to NUREG/KM Chapter 3 on criticality evaluations.

- *Evaluation: Fuel Uncovery (Criterion III)*. NUREG-0612, Criterion III, stated that damage to the reactor vessel or the spent fuel pool based on calculations of damage following the accidental drop of a postulated heavy load was limited so as not to result in water leakage that could uncover the fuel (if the water being lost was borated, makeup water provided to overcome leakage should be from a borated source of adequate concentration).

  The analysis for a postulated head drop on the vessel showed that the reactor vessel could withstand the impact of the head and the support structure. Therefore, there would be no water leakage from the RCS, and Criterion III would be met.

- ***Evaluation: Load Drop (Safe-Shutdown Equipment)***. [34] The licensee's safety evaluation considered a heavy load drop on or near safe-shutdown equipment. The evaluation identified loads, targets, and load/target interactions, as well as the resulting worst credible consequence.

- *Identification of Loads*. For the head lift and transfer, the load to be moved in the vicinity of safe-shutdown and decay heat removal equipment (or other equipment important at TMI-2 because of unique site considerations) was the head rigging, the reactor vessel closure head with the attached control rod drive mechanisms, the service structure, and any required shielding. The combined weight of these items was less than 170 tons.

- *Identification of Targets*. The SER contained a list of essential systems and components, including valves and instrumentation. The selection criteria included: (●) equipment within the reactor coolant pressure boundary required for decay heat removal and reactivity control; (●) equipment required to be operable by the recovery technical specification; and (●) equipment required by plant procedures, which were approved in accordance with the recovery technical specification.

  The major fluid systems examined as possibly presenting important targets included: (●) reactor coolant; (●) makeup and purification; (●) decay heat removal; (●) mini-decay heat removal; (●) standby reactor coolant pressure control; (●) core flood; (●) decay heat closed cooling water; (●) containment building spray; (●) chemical addition; (●) nuclear services closed cooling water; (●) feedwater and condensate; (●) main steam; (●) demineralized service water; (●) intermediate closed cooling water; (●) nuclear services river water; (●) containment building ventilation; (●) containment building purge; (●) fire protection; and (●) steam generator secondary-side vent and drains.

- *Load/Target Interactions*. The effects of a heavy load drop in areas where the reactor vessel head and the service structure were expected to be moved were bounded by the drop safety analyses [35] for the polar crane load test. This SER (for head lift) provided floor diagrams that showed potential drop areas. To ascertain the worst credible consequences of a load drop, the evaluation determined that the top and middle level floors would collapse locally when impacted. The SER presented the impact areas and load equipment combinations in a matrix format. Components were not considered to be functional after a heavy load drop, which was assumed to occur directly over the components. The availability of an unaffected alternative was ascertained in view of why the component was required. A component was considered to be an alternative only if it performed the same safe-shutdown function as the component subjected to a heavy load drop. The location of each piece of essential equipment was determined from the latest available drawings. The SER summarized the consequences of the loss of each component lying within the designated areas.

- *Evaluation: Damage to Equipment (Criterion IV)*. NUREG-0612, Criterion IV, stated that damage to equipment in redundant or dual safe-shutdown paths, based on calculations assuming the accidental drop of a postulated heavy load, would be limited so as not to result in loss of required safe-shutdown functions. Criterion IV defined “required safe-shutdown functions” as those required to: (●) maintain the reactor coolant pressure boundary; (●) reach and maintain subcriticality; (●) remove decay heat; and (●) maintain the integrity of components whose failures could result in excessive offsite releases.

  The required safe-shutdown functions that applied to the TMI-2 reactor in its current cooling mode and core configuration included: (●) capability to maintain subcriticality; (●) capability to maintain decay heat removal; and (●) capability to maintain the integrity of components whose failures could result in excessive offsite releases. The reactor coolant pressure boundary needed to be maintained only insofar as reactor coolant must be kept in the RCS for decay heat removal and reactivity control.

- *Evaluation: TMI-2 Safe-Shutdown Systems*. Required safe-shutdown functions unique to TMI-2 included the following:

  – *Subcriticality*. SER Section 4.9.1.3 addressed the capability to maintain subcriticality in the core due to load impact on the reactor itself. Since a physical redistribution of fuel due to a load drop was not expected to cause criticality, the only credible mechanism by which criticality control could be compromised was the boron dilution of the RCS water. Systems within load impact areas that contained unborated water were investigated and found to fail in such a way as to drain their contents onto the containment building floor and not into the RCS. Further, systems capable of injecting highly borated water into the RCS were available, and it was not feasible to reduce the functional capability of these systems to such a point that boron injection could not be done with one load drop.

  – *Decay Heat Removal*. Decay heat removal capability was ensured by maintaining water in the reactor vessel (refer to Section 4.2 of the SER). Even though best estimate analysis showed that equilibrium RCS temperature was acceptable with the RCS drained to the bottom of the vessel nozzles, if leakage of reactor coolant were to occur because of damage to the RCS, borated water makeup capability would exist through at least one coolant loop. Makeup system penetrations would not be damaged since they were located on the north side of the building away from the load paths.

  – *Radiological Release*. Offsite releases were prevented by the containment building boundary. The containment building integrity would be maintained during the actual head lift.

- *Evaluation: Accident Scenarios*. Additionally, a few accident scenarios were postulated as having some finite although very small occurrence potential. These included: (●) criticality in the containment building sump; and (●) impact-induced failure of in-core instrument piping. Accident scenarios included the following:

  – *Subcriticality (Sump)*. In considering subcriticality in the containment building sump, an evaluation indicated that the only point of potential concern would be the drop of a heavy load onto the systems that could provide a source of unborated water to the sump. Potential sources of unborated water in the containment building included: (●) fire protection system; (●) demineralized water system; (●) nuclear services closed cooling water system; (●) intermediate closed cooling water system; (●) normal cooling water system; (●) nuclear services river water system; (●) main steam and feedwater systems; and (●) decontamination water.

    The SER stated that this problem could be addressed in two ways. First, to limit the amount of unborated water available for leakage to the sump from these systems, the water supply to these systems would be isolated while the head lift was being performed. (Note: The operable fire protection system was normally isolated.) An evaluation showed that a significant quantity of unborated water would be required to lower the sump water concentration from current values to a level below the value of 1700 parts per million, which was specified as adequate for avoiding sump criticality.

    Second, several low-probability events needed to occur before valid concerns about sump criticality could arise, regardless of the amount of unborated water delivered to the sump. First, an amount of fuel sufficient to create a critical mass needed to have washed into the sump during the TMI-2 accident. Then, this fuel needed to be in a configuration that could induce criticality if a global boron dilution of the sump were to occur.

    A qualitative assessment indicated that the administrative controls effectively eliminated criticality as a safety concern. These controls limited the amount of unborated water that could be delivered to the sump. Combined with the low probability of simultaneous occurrence of the initial conditions for fuel deposition in the sump that could lead to a criticality problem, these controls made criticality a low-probability event.

  – *In-core Instrument Line Damage*. The other special consideration was the potential consequences of a load drop damaging the in-core instrument lines. The reactor vessel lower head was penetrated by in-core instrument lines. These lines ran from beneath the very bottom of the vessel through a tunnel in the base mat of the containment building to terminations at the seal table area. The routing of these lines paralleled a line drawn between the center of the reactor vessel and the center of the seal table. The width of the area occupied by these lines was slightly smaller than the diameter of the seal table.

    The analysis of structural effects of the head drop on the vessel showed that this accident would not cause a shearing of the instrument lines. Neither the instrumentation lines nor the seal table was located within the load path for the reactor vessel head and support structure once the head was removed from the vessel area. Therefore, a load drop that could damage the in-core instrumentation lines was considered improbable.

- ***NRC Review: Load Drop***. [36] On December 22, 1980, the NRC issued a generic letter [37] on the control of heavy loads to all licensees of operating plants. In this letter, the NRC requested that each licensee review the controls for the handling of heavy loads to determine the extent that the requirements of NUREG-0612 were met. Also contained in the generic letter was a request for additional information on the control of heavy loads: (●) Section 2.1 of the letter asked for information on the requirements for overhead handling systems. (●) Section 2.2 requested a response to specific requirements for overhead handling systems operating in the vicinity of fuel storage racks. (●) Section 2.3 stated specific requirements of overhead handling systems operating in the containment. (●) Section 2.4 stated specific requirements for overhead handling systems operating in plant areas containing equipment required for reactor shutdown, core decay heat removal, or spent fuel cool cooling. Responses to Sections 2.1 and 2. 2 were forwarded by the licensee and addressed by the NRC in previous correspondence. The licensee's head lift SER [38] discussed Sections 2.3 and 2.4.

  - *Load Handling in the Containment Building*. NUREG-0612, Section 5.1.3, provided guidance for the design and operation of load handling systems in the vicinity of the reactor core. The licensee was required to demonstrate that adequate measures were taken to ensure that, in the vicinity of the core, either the likelihood of a drop that might damage spent fuel was extremely small, or that the estimated consequence of such a drop would not exceed the limits set by the evaluation criteria of NUREG-0612, Section 5.1, Criteria I through III.

    - *Radiological Release (Criterion I)*. The licensee's SER stated that the impact of the reactor vessel head and service structure dropping onto the vessel could cause a release of gaseous radioactivity into the containment building environment. An uncontrolled release of this activity to the environment was precluded by containment building integrity during head removal. The worst case gaseous release (krypton-85) that would result in doses was a small fraction of the 10 CFR Part 100 limits. Although little airborne particulate material would be expected from a load drop, any material generated would be within the containment building boundary. Any of the airborne material that did not settle would be processed through HEPA filters before release to the environment. The NRC concluded that airborne particulates would have no impact on the doses resulting from postulated krypton-85 releases. Therefore, the NRC determined that the requirements of Criterion I were met.

    - *Criticality (Criterion II)*. Criterion II required that damage to fuel and fuel storage racks based on calculations involving accidental drop of a postulated heavy load did not result in a configuration of the fuel where the k_{eff} was larger than 0.95. Accordingly, the NRC considered the significance of a postulated head drop on the reactor pressure vessel that resulted in fuel reconfiguration. Criticality analysis for postulated core configurations with 3500 parts per million boron in the RCS yielded realistic k_{eff} values that were less than 0.90. Further, the licensee raised the boron concentration in the RCS to 5000 parts per million to ensure safe shutdown for any hypothetical core configuration. The NRC concluded that the requirements of Criterion II were met.

– *RCS Integrity (Criterion III)*. Criterion III required that damage to the reactor vessel or the spent fuel pool, based on calculations of damage following an accidental drop of a postulated heavy load, was limited so as not to result in water leakage that could uncover the fuel (makeup water provided to overcome leakage should be from a borated source of adequate concentration if the water being lost was borated).

  All heavy load movements in the head removal program were made in the containment building, and there were no movements that could impact the spent fuel pool. Heavy loads that would be handled during the heat removal evolution included the reactor vessel head assembly, which comprised the (●) lift ring; (●) vessel closure head; (●) control rod drive mechanism motor tube assemblies; (●) service structure; and (●) attached shielding with its support frames. Other heavy loads included the (●) internals indexing feature; the fixture cover; and (●) the cover shielding plates. The lifting of any combination of loads was limited to a maximum weight equal to the 170-ton rating of the containment building polar crane.

  While near the reactor vessel, the head lift height would be monitored and controlled. Given that all the reactor vessel studs were successfully removed, the head lift height for required clearance would be about 33 inches. Cover placement on the underside of the head following the initial lift could require an additional 12 inches of clearance over the reactor vessel. Thus, the head lift height in the vicinity of the reactor vessel would be no higher than 45 inches.

  The licensee evaluated the effects of a load drop on the reactor vessel in Attachment 3 of its SER [(39)] on the head lift. In this analysis, the licensee stated that depending on the components that were attached to the head during the lift, the assembly could weigh from 158 to 170 tons. Each weight had a corresponding maximum lift height, equal to the maximum vertical distance that the load could be dropped without a breach of RCS integrity (i.e., failure of the reactor vessel or the attached in-core instrument tubes). For reference, the licensee had analyzed the case in which the service structure shielding blankets were in place and all studs were removed. However, all studs had been removed in preparation for head lift. The weight under these conditions was conservatively assumed to be 174 tons. This assumed weight was about 10 tons more than the actual weight of the head and attached shielding (about 163 tons). The maximum lift height, assuming a worst case point load drop (structure tilts when dropped and hits the vessel or plenum at an angle), was calculated to be 56.1 inches. The NRC structural engineering branch reviewed the licensee's load drop calculations and confirmed the results.

  The NRC concluded that measures taken to limit head lift height in the vicinity of the reactor vessel were adequate to mitigate the consequences of an accident. Therefore, there was adequate protection against uncovering the fuel, and Criterion III was satisfied.

- *Impact of a Load Drop on Safe-Shutdown Functions*. Criterion IV of NUREG-0612 required the presentation in a matrix table of all heavy loads and potential impact areas where safety-related equipment might be damaged. The licensee provided that matrix in its SER. NUREG-0612 also required the identification of the load and impact area combinations that could be eliminated because of separation and redundancy of safety-related equipment. For load/target combinations that could impact safety-related equipment, the licensee was required to state the basis for determining that load drops would not affect the ability to perform a safety-related function (e.g., reactor shutdown, core decay heat removal, and containment building integrity).

  The required safe-shutdown functions that applied to TMI-2 at the time of the head lift activities included: (●) capability to maintain subcriticality; (●) capability to maintain decay heat removal; and (●) capability to maintain the integrity of components whose failure could result in excessive offsite release. The SER discussed the ability to maintain subcriticality and to minimize offsite release. Decay heat removal capability would be maintained because of the: (●) head lift height restrictions; (●) passive loss-to-ambient cooling mode (which depended only on water being retained in the RCS); and (●) the various options available for introducing borated water into the RCS. The NRC concluded that the requirements of NUREG-0612 were satisfied.

7.4.4 Heavy Load Handling inside Containment

- ***Purpose***. To ensure that handling of heavy loads in the containment building and spent fuel pool "A" (SFP-A) within the fuel handling building was in accordance with the safety requirements of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," [40] issued July 1980.

- ***Evaluation: Load Drop (Loads and Targets)***. [41] The containment building load drop analyses assumed that postulated load drops would result in the local failure of floors. The licensee's safety evaluation considered heavy load drops in the containment building to ensure that the postulated failures could not result in draining the reactor vessel below the bottom of the reactor vessel hot leg (314-foot elevation), disabling all makeup paths to the reactor vessel, or draining the fuel transfer canal (FTC). The safety evaluation report (SER) [42] for load handling over the reactor vessel addressed heavy load drops that could potentially drain the reactor vessel below the 314-foot level (bottom of the hot-leg outlet nozzle). Load drop analyses for load drops in SFP-A assumed that postulated load drops could result in local damage to the fuel canister storage racks (FCSRs), the fuel pool liner plate, or both.

- *Identification of Loads*. Loads handled inside the containment building were expected to range up to a maximum of 25 tons, excluding the plenum; however, this SER addressed all loads up to the 170-ton rated capacity of the main hook of the polar crane. In addition, this

SER addressed all loads that could be handled inside SFP-A up to and including the design defueling canister weight of 3355 pounds.

- *Identification of Targets*. The target for a postulated load drop was considered to be the floor and equipment in the region directly below the suspended load. Specific target areas would be identified in both the containment building and SFP-A. These target areas would be differentiated based on their ability to withstand a specific load impact. The load handling areas in the containment building included: (●) the reactor vessel; (●) the FTC deep end (with and without loaded defueling canisters); (●) the FTC shallow end; (●) the northwest "A" D-ring and seal table; and (●) all other areas inside the containment. The load handling areas in the fuel handling building (with and without defueling canisters) included SFP-A.

- *Load/Target Interactions (inside Containment)*. The SER provided floor plans of the containment building and fuel handling building, with allowed load handling areas identified. The classifications of various load handling areas were based on the following evaluations:

  - *Reactor Vessel*. All loads to be handled over the reactor vessel were discussed and evaluated in detail in the licensee's SER [43] for load handling over the reactor vessel.

  - *FTC Deep End (without Defueling Canisters)*. The handling of loads over the deep end of the FTC without filled canisters in the FTC presented no plant safety concerns. A drop in this area would not affect the stability of the core, drain or reduce the water level in the reactor coolant system, or impact the availability of makeup. In addition, a drop in this area would not prevent access to the containment building.

  - *FTC Deep End (with Defueling Canisters)*. The handling of loads over the FCSRs in the deep end of the FTC, when canisters were in the racks, would be restricted such that the potential energy would not be greater than a suspended fuel canister. The equation [H = (37,000/W) + (322)] would be used to determine the maximum plant elevation (H, maximum plant elevation in feet) to which a given weight (W, in pounds and not greater than 3355 pounds as stated in the technical evaluation report [44] for FCSRs) could be raised over the FCSRs in the containment.

  - *FTC Shallow End*. The analysis of load drops occurring in the FTC shallow end assumed that objects fell from their lift height unimpeded to the floor of the FTC and impacted at a point. This resulted in the greatest transmission of potential impact energy directly to the FTC floor, as no impact energy absorbed by the collapse of platforms or equipment was assumed.

  - *FTC North End*. The shallow end of the FTC, north of the reactor vessel, was classified as a restricted lift area. A load drop in this area could result in damage to the floor at the 322-foot elevation and could possibly impact the availability of normal makeup to the reactor vessel or damage the in-core tubes. These damages could result in draining the reactor vessel.

Load handling in this area without lift height restrictions could create a potential for local damage, such as spalling of concrete from the bottom of the floor slab, which could in turn impact the in-core instrument cable chase. To preclude any spalling that might occur, load/lift height limits were established. Table 3.5-1 of the SER presents these limits, which would be used for load handling in the north half of the shallow end of the FTC.

In the low-probability event that excess dam leakage or a complete loss of the dam function occurred, the water level would decrease in the deep end of the FTC and in SFP-A. Water shielding over both the plenum assembly and the canisters would be reduced; however, the canal could be completely flooded to increase the water level and reduce the radiation exposure levels.

- *FTC South End*. The shallow end of the FTC south of the reactor vessel was classified as an unrestricted lift area, based on the reviews performed for: (●) the SER [(45)] for removal of the reactor vessel head; (●) the SER [(46)] for the polar crane test; and (●) a review of loads that would be handled over this end of the FTC. This review examined the potential for failure of the floor at the 322-foot elevation, its impact on the availability of makeup to the reactor vessel, and damage to the in-core tubes. These damages could result in draining the reactor vessel. Based on this review, loads could be handled in these areas without presenting the potential for draining the reactor vessel or impacting the availability of makeup to the reactor vessel.

- *Northwest "A" D-Ring and Seal Table*. This area was an exclusion area as it was identified as a location in the containment building where a load drop could impact the in-core tubes and potentially drain the reactor vessel.

- *Other Areas in the Containment*. This area was classified as an unrestricted lift area if all unborated water sources were isolated. This classification was based on the review performed for: (●) the SER [(47)] for removal of the reactor vessel head and (●) the SER [(48)] for the polar crane test, which demonstrated that load drops in these areas could not result in draining the reactor vessel, impacting the availability of makeup to the reactor vessel, or creating an inadvertent criticality. Criticality was prevented by the isolation of nonborated water sources.

- o *Load/Target Interactions (SFP-A)*. The SER provided floor plans of the containment building and SFP-A with allowed load handling areas identified. The classifications of various load handling areas were based on the following evaluations:

  - *SFP-A without Defueling Canisters*. The handling of loads over SFP-A without loaded defueling canisters in the fuel pool presented no plant safety concerns. Such a drop would not affect the stability of the core, drain or reduce the water level in the reactor coolant system, impact the availability of makeup, or create the potential for an inadvertent criticality event. Therefore, the unrestricted lift area classification for this area was appropriate.

– *SFP-A with Defueling Canisters*. The handling of loads over the FCSRs in the fuel pool when loaded canisters were in the racks would be restricted such that the potential energy would not be greater than a suspended fuel canister. The equation [$H = (37{,}000/W) + (321)$] would be used to determine the maximum plant elevation (H, maximum plant elevation in feet) where a given weight (W, in pounds and not greater than 3355 pounds as stated in the technical evaluation report [(49)] for FCSRs) could be raised over the FCSRs in the containment.

- ***Evaluation: Load Drop (NUREG-0612 Criteria)***. [(50)] The licensee's safety evaluation examined the results of load drops postulated in its SER against the four criteria given in NUREG-0612.

o *Radiological Release (Criterion I)*. Any releases of radioactivity caused by the load drops addressed in the SER would be in the containment building or in the fuel handling building. These buildings would act as a physical barrier and prevent any liquid releases from escaping to the environment. Likewise, HEPA air filters would remove any additional particulates that could become airborne so as not to exceed the limits established in Criterion I. A bounding analysis was performed that assumed an instantaneous total release of the unaccounted for krypton-85 inventory from the reactor core. The amount released was assumed to be 31,300 curies of krypton-85 with the resulting dose estimated to be 9.7 millirem to the whole body for an individual located at the nearest site boundary and 1.8 millirem to the whole body for an individual located at the low-population zone boundary. The meteorological dispersion factors used were 6.1×10^{-4} second per cubic meter (sec/m^3) at the site boundary and 1.1×10^{-4} sec/m^3 at the low-population zone boundary (taken from the preaccident TMI-2 final safety analysis report).

An additional analysis was performed in the licensee's SER [(51)] for bulk defueling of the reactor vessel to determine the maximum offsite dose due to any airborne particulates that could pass through HEPA air filters following the drop of a defueling canister. This analysis used conservative assumptions and calculated a critical organ (teenager's bone) dose of 2.96 rem, which was less than 4 percent of the 75 rem acceptance criteria and 25 percent of the 10 CFR Part 100 dose guidelines. The bone dose was presented since the bone was determined to be the critical organ based on comparisons of dose conversion factors for several organs, including the lung, kidney, liver, and gastrointestinal tract, for the distribution of radionuclides available for release.

o *Criticality (Criterion II)*. The dropping of heavy loads on the FCSRs without defueling canisters loaded with fuel debris (in either SFP-A or the FTC) posed no safety concern as there was no opportunity for a criticality event, radiation release, or uncovering of fuel. The handling of heavy loads over the FCSRs with loaded or partially loaded canisters present would be maintained within the limits set in the SER. This would ensure that the FCSR was not damaged to such an extent as to cause a return to criticality. Load handling over the reactor vessel and the associated safety issues were discussed in the SER [(52)] on that topic.

As in previous load handling SERs, the isolation of nonborated water sources during the handling of heavy loads to prevent the addition of nonborated water to the containment building sump was necessary. As stated in a letter from the NRC, [53] isolation of the containment building chilled water system to prevent a sump criticality event was no longer required. The systems identified in a previous SER [54] on the polar crane test were potential sources of unborated water inside containment.

As an alternative, adherence to the load weight and height guidelines in the SER would ensure that a dropped load would not fail the floor slab. Therefore, the unborated water systems located beneath the floor slab where a load was being carried need not be isolated. Any unborated water systems that could be directly impacted by a load drop within the area of a particular load handling activity would be isolated until completion of that activity.

During any load handling activity with load weight/height in excess of the guidelines in the tables in the SER, all unborated water sources in the containment building would be isolated unless it could be demonstrated that there was sufficient physical separation between the load handling area and specific systems to ensure there would be no system failure in the event of a load drop.

- *Reactor Vessel/SFP-A Integrity (Criterion III)*. As loads would not be handled over the in-core instrument tubes, the load drops postulated in the SER could not drain the reactor vessel below the bottom of the reactor vessel hot leg (314-foot elevation). Drainage to this level would not uncover the fuel. Makeup could be provided by the makeup system through redundant pathways to the reactor vessel.

  The dropping of a heavy load in the deep end of the FTC or in SFP-A could result in local damage to the stainless-steel liner plate. The extent of this damage would be determined by the shape and weight of the dropped load. Damage could range from denting to perforation of the liner plate. The perforation of the liner plate could result in water being lost from SFP-A/FTC; this water would be collected by the liner leakage collection system and directed to the auxiliary building sump for SFP-A leakage or containment building sump for FTC leakage. The borated water storage tank would provide necessary makeup. The catastrophic failure of the slab in the deep end of the FTC was not considered credible because of the existence of a concrete support wall located at the center of the slab.

  The technical evaluation report for defueling canisters (refer to Section 5.3.1 in NUREG/KM Chapter 3) described an analysis to determine the potential for criticality to occur in SFP-A/FTC because of a catastrophic failure of the liner causing SFP-A/FTC to be drained of water. This analysis determined that a criticality event would not occur.

- *Damage to Equipment (Criterion IV)*. The licensee evaluated the results of load drops postulated in the SER against Criterion IV of NUREG-0612. Criterion IV referred to “required safe-shutdown functions” as those necessary to: (●) maintain the reactor coolant pressure boundary (not applicable for this revision of the SER because the reactor vessel was open);

(●) remove decay heat; (●) maintain subcriticality; and (●) maintain the integrity of components whose failures could result in excessive offsite releases.

– *Decay Heat Removal* Control. Reactor coolant would be maintained in the reactor coolant system above the reactor vessel nozzles for decay heat removal and reactivity control. Subcriticality would be maintained as described in the criticality evaluation (Criterion II). Decay heat was removed by heat losses to ambient air in the containment building, which was demonstrated in an earlier licensee evaluation [55] to be adequate to remove all decay heat produced by the core material in the reactor vessel. Thus, no additional equipment was necessary to remove decay heat.

– *Criticality Control*. Reactivity would continue to be controlled if the level of borated water in the reactor coolant system and SFP-A/FTC was maintained. Thus, dropping a heavy load would affect reactivity control only if the load drop resulted in breaking in-core instrument tubes since the breaking of the in-core instrument tubes would drain the reactor vessel below the 314-foot elevation. However, for the load drops postulated in the SER, in-core instrument tubes would not break because there were no in-core instrument tubes outside of the load handling exclusion areas.

– *Component Integrity*. For the load drops postulated in the SER, in-core instrument tubes would not break because there were no in-core instrument tubes outside of the load handling exclusion areas.

– *Conclusion*. The licensee concluded that safe shutdown would be maintained for load handling and load drop accidents postulated in the SER.

- ***NRC Review: Load Drop (Fuel Canister)***. [56,57] Several accident scenarios that involved a dropped fuel canister were considered by the licensee and evaluated by the NRC. Both safety evaluations were based on the criteria of NUREG-0612.

o *Radiological Release (Criterion I)*. The licensee assumed that a canister drop would release the entire unaccounted for krypton-85 inventory (about 31,000 curies) of the core with 0.12 weight percent of the contents of a canister as particulate matter. The resultant offsite dose consequence was less than 4 percent of the acceptance criteria. The licensee's analysis presented a very conservative case from both a probabilistic and consequence standpoint. The drop was assumed to occur over a dry location; the canisters would be retained by two separate, diverse mechanisms when lifted over dry areas. The drop assumed a dry powder was in the canisters when they would be drained but still wet with surface water.

o *Criticality (Criterion II)*. The licensee and the NRC both considered cases in which a canister was dropped on another canister either in the FTC or SFP-A. The entire contents of the upper canister were assumed to spill and to form the worst case geometry, including highest enrichment fuels surrounding the lower canister. No credit was taken for zirconium cladding material, poison (criticality control) materials incorporated in the canister, and the structural materials of the canister. The NRC's criticality evaluation report [58] for a loaded canister dropping its load onto another canister concluded that a considerable shutdown margin (i.e., effective neutron multiplication less than 0.95) would exist. The licensee also examined the case of an infinite array in a drained fuel pool condition and concluded that subcriticality would be maintained with an effective neutron multiplication less than 0.964. The infinite array scenario required a series of dozens of consecutively dropped canisters. The NRC considered this scenario not to be credible.

- *Reactor Vessel/SFP-A Integrity (Criterion III)*. The NRC provided its review of the licensee's load drop and leakage analysis for the reactor vessel in a previous SER [59] for heavy load handling over the reactor vessel. If SFP-A or the FTC drained, there would be no immediate effects. The entire core had a decay heat of less than 12 kilowatts, and heat generation in individual stored canisters would be less than 100 watts, posing no problem. The NRC evaluated the potential for gas generation in the canisters in a previous SER [60] for the defueling canisters and found it acceptable. Thus, the canisters would remain stable for long periods of time in a drained pool.

- *Important Equipment (Criterion IV)*. The NRC reviewed the results of load drops postulated in the SER against Criterion IV of NUREG-0612. The activities addressed would not result in loads over essential safe-shutdown equipment. In the current mode, forced cooling of the reactor core was not required. A dropped load could potentially cause leakage through the in-core instrument tubes. The consequence of this accident and the licensee's ability and method to mitigate the consequence were evaluated in a previous NRC SER [61] for heavy load handling over the reactor vessel and found acceptable.

- ***NRC Review: Load Drop (Reactor Vessel Integrity)***. [62,63] The NRC's safety evaluation considered postulated load drops in accordance with Criterion III of NUREG-0612. The NRC reviewed the licensee's load drop and leakage analysis for the reactor vessel in a previous SER [64] for heavy load handling over the reactor vessel. If SFP-A or the FTC drained, there would be no immediate effects.

- ***NRC Review: Load Drop (Impacts to Vital Equipment)***. [65,66] The NRC's safety evaluation considered postulated load drops in accordance with Criterion IV of NUREG-0612. The activities addressed would not result in loads over essential safe-shutdown equipment. In the current mode, forced cooling of the reactor core was not required. A dropped load could potentially cause leakage through the in-core instrument tubes. The consequence of this accident and the licensee's ability and method to mitigate the consequence were evaluated in a previous NRC SER [67] for heavy load handling over the reactor vessel and found acceptable.

7.4.5 Heavy Load Handling over the Reactor Vessel

- ***Purpose***. To permit the handling of heavy loads (greater than 2400 pounds including rigging weight) in the vicinity of the reactor vessel. This included any load handling activity that could result in a heavy load drop onto or into the vessel either directly or indirectly (as the result of the collapse of structures/equipment installed over the vessel).

- ***Evaluation: Load Drop (Accidents)***. [68] The licensee's safety evaluation considered loads dropped over or within the reactor vessel that have the potential to indirectly or directly damage the in-core instrument tubes, which penetrated the vessel lower head. Since these instrument tubes were part of the reactor coolant system (RCS) boundary, damage to the tubes or to the penetration of the tube at the lower head could result in loss of water from the RCS. The exact nature of the damage and the resulting reduction in reactor vessel water level were difficult to determine. If the loss of water could be shown to be within the makeup capability to the vessel, then the core would remain covered during postulated drop accidents. No other credible RCS failure modes could lower the water level below the elevation of the bottom of the reactor vessel outlet nozzles (314-foot elevation).

- ***Evaluation: Load Drop (Prevention)***. [69] The licensee's safety evaluation stated that the potential for a load drop accident into the reactor vessel was minimized by careful control of load handling activities and equipment. Heavy loads were controlled by the following measures: (●) Load handling equipment was conservatively designed and tested as described in Section 2.0 of the safety evaluation report (SER). For example, a drop of the plenum assembly (PA) by failure of the polar crane or the tripod was extremely unlikely since the PA and attendant rigging and attachments weighed about 73 tons. This weight was less than one-half of the current rating of the polar crane and was less than one-sixth of the crane's original design capacity rating of 500 tons. A drop of the PA by failure of the pendants was also extremely unlikely based on their factors of safety, including: (●) Load handling activities were performed in accordance with approved procedures. (●) Loads were not carried over the reactor vessel without first evaluating alternative load paths. (●) Each specific load handling activity was controlled by a unit work instruction or procedure, which was reviewed by a responsible technical reviewer in accordance with TMI-2 procedures. (●) Load lifting and handling activities would be performed by personnel who were trained and qualified for these activities as described In Section 2.2 of the SER.

- ***Evaluation: Load Drop (Reactor Vessel Integrity)***. [70] The licensee's safety evaluation stated that the only postulated failure mechanism that could potentially lower the water level in the reactor vessel below the bottom of the coolant pipe nozzles was damage to the in-core instrument tubes at the point where they penetrated the lower vessel head. There were 52 such penetrations distributed in the lower head. The analyses of load drop impacts on reactor vessel integrity included: (●) review of leakage experience; (●) identification of loads; (●) identification of targets; (●) evaluation of load/target interactions; and (●) evaluation of postulated failure mechanisms.

  - *Leakage Experience*. The SER noted that at present there was no measurable leakage from the RCS. Some loads had already been exerted on the top of the core debris bed during recovery activities performed previously (e.g., sample probes and dropped partial fuel assemblies during plenum jacking and end fitting removal) without causing leakage from the vessel. In addition, reactor vessel lower head video inspections did not show damage to the visible in-core instrument nozzles and the lower vessel head. In addition, the video inspections showed no evidence of bridging by debris buildup of the gap between the visible in-core instrument guide tubes and in-core instrument nozzles. In addition, from March 1979 until the removal of the vessel head in July 1984, the RCS was subjected to various internal pressures including greater than 1000 pounds per square inch gauge (psig) immediately following the accident; 300 psig ± 60 psig from May 1979 to April 1980; 90 psig ± 10 psig from April 1980 to April 1984; and 50 psig until June 15, 1984, when depressurization occurred.

- *Identification of Loads*. Load handling over the reactor vessel included the following heavy loads:

  - *Reactor Vessel Head Removal*. The SER (71) for the removal of the reactor vessel head presented an analysis of a postulated drop of the reactor vessel head onto the vessel flange and onto the PA. This analysis showed that the structural integrity of the reactor vessel and its support skirt was not compromised. The resulting reactor vessel displacements did not cause stresses on the attached piping, including the in-core instrument tubes, to exceed their faulted condition stress limits given in Section III of the American Society of Mechanical Engineers *Boiler and Pressure Vessel Code*, 1974 Edition, thus precluding failure of attached piping.

  - *PA Removal*. A separate analysis was performed for a drop of the PA 7.5 feet through air and an additional 14.8 feet in water. The analyzed plenum load weight was 73 tons, which included the plenum, rigging, and jacks. For conservatism, this analysis assumed the plenum would fall unimpeded through the internals indexing fixture (IIF), though this was not considered credible because of the clearances involved and the existence of indexing keys. The analysis considered the buoyancy and frictional drag forces afforded by the water. The resulting kinetic energy of the plenum at impact was about 1.2 million foot-pounds. The kinetic energy of the dropped reactor vessel head at impact was about 1.7 million foot-pounds, or 42 percent greater than the kinetic energy of the falling plenum. Based on analysis, the maximum allowable lift height of the plenum was 22.3 feet. However, it was expected that the actual 11-foot height would be well below this height.

  - *Other Loads*. In the event the load was dropped, the lift heights for loads, other than the PA to be handled over the reactor vessel, would be restricted to ensure that the impact energy transmitted to the reactor vessel or any internal reactor vessel components would be less than or equal to the impact energy resulting from a reactor vessel head drop, as analyzed in the SER for the removal of the reactor vessel head. The lift height and weight restrictions would differ depending on whether the IIF platform, PA, and defueling work platform were installed or removed. The restrictions were presented as four cases that covered all currently identified configurations through to the end of vessel defueling.

- *PA in Vessel on Jacks and IIF Platform Installed (Case 1)*. When the PA was supported by jacks and the IIF platform installed, any load that would be dropped over the reactor vessel was assumed to first impact the IIF platform, which would cause the platform to collapse onto the raised plenum. Further, the analysis assumed that the dropped load and collapsed IIF platform caused the jacks to fail and the plenum to fall to its pre-raised position. The total impact energy transmitted to the core support assembly from the IIF platform and the plenum was calculated to be about 330,255 foot-pounds. This would leave about 126,745 foot-pounds of impact energy available for the dropped load to remain within the bounds of the reactor vessel head drop analysis. The following expression conservatively defined the allowable lift height, H (feet), for a load of weight, W (pounds): [H = (126,745/W) + 322.5]. The term H was the maximum plant elevation in feet to which the load could be raised, and W was the weight in pounds of the lifted load including the weight of rigging, which was rigidly attached to the load.

- *PA in Vessel on Jacks and IIF Platform Removed (Case 2)*. When the PA was supported by jacks and the IIF platform was removed, this case assumed that a dropped load caused the jacks to fail and the plenum to fall to its pre-raised position. The total impact energy transmitted to the core support assembly from the plenum was calculated to be about 58,000 foot-pounds. This would leave about 399,000 foot-pounds of impact energy available for the dropped load in order to remain within the bounds of the reactor vessel head drop analysis. The following expression conservatively defined the allowable lift height, H (feet), for a load of weight, W (pounds): [H = (399,000/W) + 322.5]. The term H was the maximum plant elevation in feet to which the load could be raised, and W was the weight in pounds of the lifted load including the weight of rigging, which was rigidly attached to the load.

- *PA Removed and Defueling Work Platform Not Installed (Case 3)*. When the PA was removed from the reactor vessel and the defueling work platform was not yet installed, the following expression conservatively defined the allowable lift height, H (feet), for a load of weight, W (pounds): [H = (457,000/W) + 322.5]. The term H was the maximum plant elevation in feet to which the load could be raised, and W was the weight in pounds of the lifted load including the weight of rigging, which was rigidly attached to the load. In addition, no load would be raised above the 405-foot elevation.

- *PA Removed and Defueling Work Platform Installed (Case 4)*. An analysis was performed to calculate load handling limits for load handling over the defueling work platform. This analysis examined the entire platform to determine the weakest point, and impact loads were applied at that point. The load and lift height limits calculated in this analysis precluded the collapse of the defueling work platform. The table of results of the analysis that was presented in the SER (refer to page 11 of the SER) showed a range of 2000 to 50,000 pounds with an allowable plant elevation range of 340.3 to 331.8 feet.

- *Identification of Targets*. The only postulated failure mechanism that could potentially lower the water level in the reactor vessel below the bottom of the coolant pipe nozzles was damage to the in-core instrument tubes.

  - *Instrument Tube Integrity outside Vessel*. The in-core instrument tubes outside of the vessel were 0.5-inch Schedule 80, 304L stainless-steel pipes, each welded to a 0.75-inch Schedule 160 Inconel 600 nozzle penetrating the vessel wall. The SER [(72)] for the measurement of uranium fuel material in the lower reactor vessel region using gamma profiling of the in-core detectors demonstrated the integrity of the in-core instrument tubes outside of the reactor vessel. This evaluation concluded that these tubes were not adversely affected by the March 28, 1979, accident or by the environment (both internal and external) to which the tubes were exposed since the accident. Based on the reactor vessel integrity analyses described in the SER for load drop over the reactor vessel and the evaluation presented in the fuel measurement SER, a failure of in-core tubes outside of the reactor vessel for the maximum load to be handled over the vessel was not considered credible as a consequence of a load drop onto the vessel flange or PA.

  - *Instrument Tube Integrity inside Vessel*. Each in-core instrument tube terminated inside the vessel at the in-core instrument nozzle. Above each in-core instrument nozzle, separated by a vertical gap of several inches was the in-core instrument guide tube, which was attached to the lower core support assembly. The only physical connection between the in-core instrument nozzle and the in-core instrument guide tube was the in-core instrument detector assembly, which consisted of a cluster of nine detectors within an Inconel sheath with a wall thickness of 0.021 inch.

- *Load/Target Interactions*. A load drop could interact with in-core instrument guide tubes by impact on the upper core support assembly or on the lower core support assembly (LCSA).

  - *Upper Core Support Assembly*. The entire core support assembly was supported from a ledge in the interior wall of the reactor vessel just below the reactor vessel flange. A heavy load drop into the vessel onto the PA or onto the core debris bed following removal of the plenum (and including a drop of the plenum back into the vessel) could cause a downward deflection of the LCSA. The impact energy of the single heaviest load to be handled over the vessel (i.e., the plenum) would be less than the impact energy of the head drop analysis presented in the SER for the removal of the reactor vessel head. The resulting LCSA deflection calculated for the head drop was about 0.5 inch, considerably less than the normal vertical clearance (gap) between the in-core instrument nozzle and the in-core instrument guide tube.

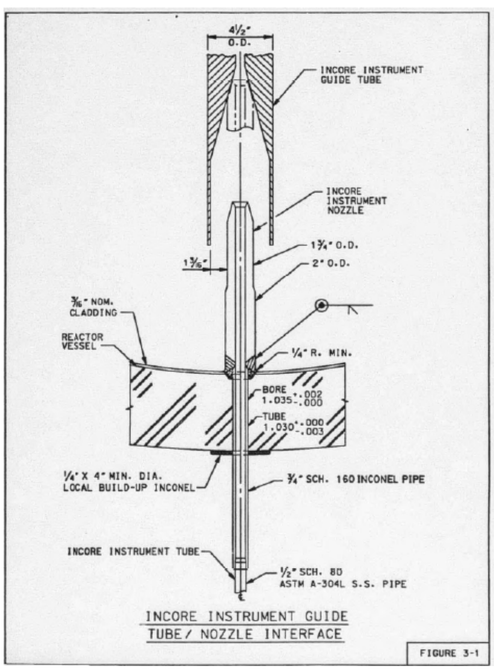
  - *LCSA*. The downward deflection of the LCSA could exert a force on the in-core instrument nozzles by one or more of the following three mechanisms: (●) transmission of force by the in-core instrument detector assembly; (●) bridging of the gap between the guide tubes and the nozzles by fuel debris; and (●) guide tube impact directly onto the nozzles.

- *Transmission of Force*. Since the in-core instrument assembly was contained within a 0.021-inch sheath, the cables and sheath would buckle before they could transmit a damaging force to the nozzles.

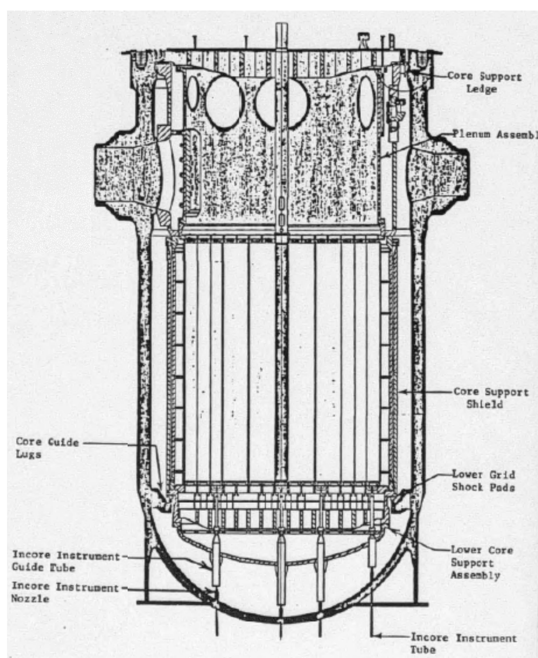
- *Bridging the Gap*. Above each in-core instrument nozzle, separated by a vertical gap of several inches was the in-core instrument guide tube, which was attached to the LCSA. Although there was a possibility that fuel debris bridging could have occurred between the nozzles and the guide tubes, bridging was highly unlikely because of the umbrella configuration (refer to Figure 3-1 of the SER).

- *Guide Tube Impact*. The nozzles could be damaged if local deflection of the LCSA, due to a point load drop onto the debris bed after the removal of the plenum, exceeded the vertical clearance between the guide tubes and the nozzles. The nozzle was shown in phantom in Figure 3-1 of the SER at the point where the nozzle would sit within the guide tube and the guide tube would exert its full downward force on the nozzle. If the load impact was great enough to cause a gross failure of the LCSA, the downward movement of the assembly could be restrained by the core guide lugs. These lugs were 0.833 inch below the lower grid shock pads when the LCSA was in its normal position (refer to Figure 3-2 of the SER). If the shock pads failed, most or all of the in-core instrument nozzles could be damaged.

  After the plenum was removed and during the early stages of defueling, the upper layer of the core debris bed would provide some protection for the in-core tube nozzles by absorbing some of the load drop energy. This layer consisted of gravel-



SER Figure 3-1. Incore instrument guide tube and nozzle interface.



SER Figure 3-2. Lower core support assembly showing core guide lugs and lower grid shock plate.

like material, which would transmit less energy to the LCSA than would be transmitted by a solid debris bed.

- *Postulated Failure Mechanism*. If a heavy load drop was postulated, and if the resulting downward deflection of the LCSA was sufficient to cause damage to the in-core instrument nozzles, the worst anticipated failure mechanism was the shearing off of the nozzle at the inside vessel wall. The 0.75-inch Schedule 160 portion of the instrument tube that penetrated the vessel wall was welded directly to the vessel wall. The in-core instrument nozzle was welded separately to the vessel wall and the 0.75-inch pipe. Consequently, failure of the nozzle was unlikely to fail the 0.75-inch pipe to vessel weld that provided the penetration seal. For conservatism, analysis assumed that this weld failed as a result of the postulated load drop accident.

  Failure of the tube-to-vessel-wall weld would not result in the tubes being forced out of the lower head by the head of water in the vessel. The tubes consisted of Schedule 80 stainless-steel pipe and were supported at the floor below the vessel. Considering manufacturing tolerance, the maximum clearance between the outside diameter of the tube and the inside diameter of the bore in the vessel wall was 0.010 inch. There was insufficient flexibility in the tubes to allow them to drop the 5.5 inches required to fall free to the bottom of the vessel head.

  As noted previously, in-core tube failure outside of the vessel was not considered credible. Consequently, the only credible leakage path from the vessel following a heavy load drop was through the annulus around the tube penetrations through the vessel wall.

- ***Evaluation: Load Drop (Mitigating Features)***. [73] The licensee's safety evaluation provided an overview of mitigating features that could identify leakage and refill the reactor vessel with borated water using installed and planned backup injection capabilities.

- *Level Indication*. In the event of a heavy load drop into the reactor vessel, an RCS leak would be detected by one or more of the level monitoring systems provided for the system. Since the two-level monitoring systems were installed outside of the reactor vessel (one connected to an RCS hot leg and the other to an RCS cold leg), they were not subject to damage from a load drop accident in the vessel. Low RCS level alarms were provided in the control room.

- *Leak Rate*. An analysis of the maximum leakage rate possible through the in-core tube lower head penetrations, which assumed a constant water level at the 327.5-foot elevation in the IIF and a 0.010-inch annular gap (refer to Figure 3-1 of the SER), demonstrated that the resulting flow from failure of all 52 penetrations would be less than 20 gallons per minute (gpm) (less than 0.40 gpm per penetration). This analysis assumed the maximum static head of water in the vessel before the load drop accident and the worst case manufacturing tolerances for all 52 penetrations. The analysis took no credit for the tube-to-vessel weld at the inside wall or the local Inconel buildup on the outside of the vessel.

- *Borated Water Sources*. In the unlikely event of a reactor vessel leak, the core debris bed would remain covered with borated water, by means of makeup from the borated water storage tank (BWST) and recirculation from the basement through the decay heat removal (DHR) system or a recirculation system to be installed for this purpose.

  As described in a previous safety evaluation [74] for a recovery technical specification change request (No. 46), the BWST, with a capacity of about 460,000 gallons of makeup water for the RCS, was located outside the containment building. This tank was the primary source of makeup water to the RCS in the event of a leak. The licensee planned to maintain a minimum of 390,000 gallons of water in the BWST except as permitted in accordance with approved procedures pursuant to technical specifications. However, before the plenum lift, the BWST inventory could be temporarily reduced to 310,000 gallons to flood the deep end of the fuel transfer canal. Consequently, for conservatism, 310,000 gallons was assumed as the BWST inventory.

- *Injection Capability*. Two methods were available to transfer water from the BWST to the RCS: (●) installed pumps, and (●) gravity flow. If electrical power was available and pumps were used, the entire quantity of water in the BWST was available for makeup. If electrical power was not available, about 220,000 gallons of the 310,000 gallons minimum inventory in the BWST were available as makeup by gravity flow through the reactor core flood nozzle at the 317.5-foot elevation. Analyses showed that substantially more flow capacity was available by gravity flow than the 20-gpm leak.

- *Refill Time*. The water level in the IIF during recovery operations was planned to be at the 327.6-foot elevation, but future operations could possibly lower the level to the 321.6-foot elevation (just below the reactor vessel flange) to facilitate fuel handling operations. The reactor water level would always be at the 321.5-foot elevation or higher before any postulated in-core leak. Under these conditions, an in-core leak of 20 gpm would take over 14 hours for the water level to drop to the 314-foot elevation. This would provide ample time to detect the leak and to initiate pumped or gravity flow from the BWST to the RCS even in the event of a postulated 5-hour loss of electrical power.

  Once gravity makeup to the RCS was initiated, about 220,000 gallons of water would be available in the BWST to provide sufficient water to make up for the 20-gpm leak rate for about 7.6 days. After restoration of offsite power, the use of the pumps would permit adding the remaining 90,000 gallons of BWST water, providing a total of about 10.7 days of water storage for RCS makeup. Recirculating water from the containment building basement by the DHR system could provide makeup to the reactor vessel. Each DHR system pump had a capacity of 3000 gpm at the rated head and would be operated, as required, to maintain the required water level in the reactor vessel.

- *Boron Concentration*. Before the initiation of recirculation flow from the containment building basement to the RCS, basement sump water would be measured to ensure that the sump water was borated to the minimum allowable boric acid concentration in the RCS. Since the basement boron concentration could be less than the minimum concentration, methods of attaining the proper boron concentration were investigated and would be implemented before initiating recirculation to the RCS. If the containment building basement contained 70,000 gallons of unborated water, there were two alternative methods available to ensure that the basement was sufficiently borated before the initiation of recirculation flow. These alternatives, which were described in the recovery technical specification change mentioned above, demonstrated that the boron concentration in the RCS would be maintained above 4350 parts per million (ppm) following a load drop into the reactor vessel.

- *Alternate Injection*. The licensee procured a portable, dedicated pumping system to provide recirculation to the RCS from the containment building basement. The system, which included a minimum of two submersible pumps and required hoses and valves, would replace the DHR system for recirculation following a load drop into the vessel. This alternate system would permit the licensee to remove the DHR from the technical specifications in accordance with recovery technical specification change request No. 46. The system would be procured, demonstrated to be operable, and placed in dedicated, onsite storage. In the event of a load drop into the vessel resulting in an RCS leak, the submersible pumps would be lowered into the containment building basement (with hoses attached) and connected to an installed electrical power receptacle. The discharge end of the hose(s) would be placed directly into the IIF. The complete system installation could be accomplished well within the period when makeup from the BWST was provided.

- ***Evaluation: Load Drop (NUREG-0612)***. (75) The licensee included postulated load drops in its safety evaluation against the four criteria in NUREG-0612, based on calculations involving accidental dropping of a postulated heavy load.

- *Radiological Release (Criterion I)*. Criterion I of NUREG-0610 stated that releases of radioactive material that could result from damage to spent fuel would produce doses that were well within 10 CFR Part 100 limits of 300 rem thyroid and 25 rem whole body (analyses should show that doses were equal to or less than 25 percent of 10 CFR Part 100 limits).

  Any activity releases caused by the load drops addressed in this SER would be released within the containment. The containment building would act as a physical barrier and prevent any liquid releases from escaping to the environment. Likewise, any additional particulates that could become airborne would be removed by HEPA filters to avoid exceeding the limits established in Criterion I. The analysis for krypton-85 release (refer to Section 3.3.2 of the SER) showed that even when using worst case assumptions (instantaneous total release with no containment), the maximum whole-body dose was 9.7 millirem compared to a limit of 6250 millirem (25 percent of 25 rem).

- *Criticality (Criterion II)*. Criterion II stated that damage to fuel and fuel storage racks, did not result in a configuration of the fuel such that effective neutron multiplication (k_{eff}) was larger than 0.95.

  The SER (76) for the criticality in the RCS conservatively demonstrated that, with the RCS maintained at a boron concentration of at least 4350 ppm, no reconfiguration of the fuel debris in the reactor vessel could cause criticality, including a reconfiguration resulting from a heavy load drop. That SER stated that k_{eff} was less than 0.99 at a boron concentration of 4350 ppm and k_{eff} should be less than 0.97 at 5000 ppm. These were conservatively calculated k_{eff}, whereas the actual k_{eff} was much lower. Therefore, these values met the intent of the criterion and were adequate for the TMI-2 recovery period. The RCS was normally maintained at a boron concentration of 5050 ppm ± 100 ppm. If a heavy load drop caused leakage from the vessel, the makeup and recirculation systems were available to prevent the RCS boron concentration from dropping below 4350 ppm (refer to Section 3.1.3 of the SER on mitigating features). Consequently, the load drops postulated in this SER could not cause a criticality event.

- *Reactor Vessel/Spent Fuel Pool "A" Integrity (Criterion III)*. Criterion III stated that damage to the reactor vessel or the spent fuel pool was limited in order to avoid water leakage that could uncover the fuel. If the water being lost was borated, makeup water provided to overcome leakage should also be from a borated source of adequate concentration.

  As described in the section on mitigating features (refer to Section 3.1.3 of the SER), sufficient makeup and recirculation capability was provided in the event of the maximum postulated water leakage from the vessel. These features would ensure that the fuel debris bed would remain covered and adequate boration would be maintained. Load handling activities that could potentially damage the integrity of the vessel could not also damage the makeup flowpath.

- *Damage to Equipment (Criterion IV)*. Criterion IV stated that damage to equipment in redundant or dual safe-shutdown paths would be limited to prevent loss of required safe-shutdown functions.

- *TMI-2 Safe-Shutdown Functions*. The required safe-shutdown functions that applied to the TMI-2 reactor in its current cooling mode and core configuration included: (•) capability to maintain subcriticality, (•) capability to maintain decay heat removal, and (•) capability to maintain the integrity of components whose failures could result in excessive offsite releases. The reactor coolant pressure boundary needed to be maintained only insofar as reactor coolant must be maintained in the RCS for DHR and reactivity control.

- *Results*. Reactor coolant would be maintained in the RCS above the reactor vessel nozzles for DHR and reactivity control. The SER concluded that subcriticality would be maintained (refer to Section 5.1.2 of the SER). As determined in the SER for the removal of the reactor vessel head, the decay heat was removed by heat losses to ambient air inside the containment building, which was previously demonstrated to be adequate in removing all decay heat produced by the core material in the reactor vessel, as long as the water level in the reactor vessel was at the 314-foot elevation as a minimum. The SER stated that the water level in the reactor vessel would be maintained above the 314-foot elevation (refer to Section 3.1.3 of the SER). Therefore, the ability to adequately remove the decay heat by the losses to ambient mode of cooling would be maintained. As such, no additional equipment was necessary to remove decay heat. The SER indicated that offsite releases were well within acceptable limits (refer to Section 5.1.1 of the SER).

- *Conclusion*. The SER concluded that safe shutdown would be maintained for load handling and load drop accidents postulated in the SER. In addition to the safe-shutdown functions, the RCS water level would be maintained to provide personnel shielding.

- ***NRC Review: Load Drop (Reactor Vessel)***. (77) The NRC reviewed the licensee's SER (78) for heavy load handling specifically over the reactor vessel. Later SERs combined various load handling activities in the containment building. The NRC's safety evaluation considered the following:

  - *Worst Case*. The worst case potential accident resulting from a load drop onto the reactor vessel would be the breach of the in-core instrument tubes, which penetrated the bottom of the vessel. These tubes were previously addressed in the licensee's SER (79) for the removal of the reactor vessel head, which the NRC reviewed. (80) However, when considering the video of lower core damage, there was a higher probability that some damage did occur to the in-core instrumentation tube area. Therefore, the licensee postulated the breakage of all 52 tubes, resulting in a 0.010-inch circular annular gap for each tube. This leakage pathway was based on the licensee's conclusion that none of the failure mechanisms would cause the 0.75-inch Inconel tube to fall out of the vessel (refer to Figure 1 of the NRC's SER). [h] After several discussions with the licensee and the NRC's independent review, the agency concurred with this conclusion. Therefore, with 52 in-core tubes leaking, the total leak rate was about 20 gallons per minute (52 tubes x 0.40 gallons per minute per tube).

  - *Mitigation Systems*. Systems available to prevent the uncovering of the core included: (●) the DHR system in recirculation mode; (●) greater than 20-gallon-per-minute portable recirculation system (available for plenum lift to mix borated water in the containment building sump and transfer water from the sump back to the reactor vessel); and (●) gravity flow from the BWST to the reactor vessel (only a temporary solution).

    - *Decay Heat Removal System*. The DHR system had a rated flow capacity of 3000 gallons per minute and was, therefore, more than adequate to take suction from the containment building sump (the drainage point for RCS leakage) and return the water to the vessel.

    - *Portable Boration System*. The portable boration system had a minimum recirculation capacity of 20 gallons per minute and could be put into operation within 24 hours. This system would also take suction from the containment building sump for recirculation and mixing and discharge back into the vessel. The capability to sample the sump before initiating recirculation would also be available for plenum lift, thereby ensuring that water with boron concentrations below 4350 ppm would not be pumped into the core. This

[h] Figure 1 in the NRC's SER is the same as Figure 3-1 in the licensee's SER.

boron concentration and the resulting k_{eff} was addressed in the NRC's amendment [81] to the recovery technical specification for an increase in the minimum required boron concentration.

– *BWST*. The BWST gravity feed had a guaranteed available inventory of 220,000 gallons of the BWST maximum estimated inventory of 310,000 gallons. The licensee's analyses showed that a substantially greater flow rate was available by gravity flow than the postulated leak rate of 20 gallons per minute.

- *Response Time*. Assuming that 20 gallons per minute was required, a recirculation capability would not be needed for about 183 hours or 7.6 days. This was sufficient time to decide when the recirculation system would be used. Further, assuming that no actions were taken, the RCS level would remain above the reactor vessel flange, and an in-core leak of 20 gallons per minute would take in excess of 14 hours before water dropped to the top of the reactor core. This leak rate would provide ample time to detect the leak and take action.

- *Radiological Release*. A bounding analysis performed by the licensee assumed an instantaneous total release of the presently unaccounted for 31,300 curies of krypton-85. This gas was assumed to be trapped in the grain boundaries of fuel pellets and fuel rods. The resulting dose was 9.7 millirem to the whole body for an individual located at the nearest site boundary and 1.8 millirem to the whole body for an individual located at the low population zone boundary. If a sump recirculation mode was necessary, any airborne particulate activity would be collected by HEPA filters in the ventilation systems for the containment building. The above doses were compared to the 6.25-rem limit in 10 CFR Part 100.

- *Load Drop (Work Platform)*. The NRC reviewed a proposed matrix that correlated weight and allowable lift heights for an object. If the object was to drop, the defueling work platform would not fail. This platform would be installed after the plenum lift. Since the use of load/height matrix would preclude failure of the platform for these analyzed load drop accidents, the NRC concurred with its implementation.

7.4.6 Plenum Assembly Removal Preparatory Activities

- ***Purpose***. To confirm the adequacy of plenum removal equipment and techniques, the preparatory activities included: (●) video inspection of potential interference areas; (●) video inspection of the core void space; (●) video inspection of the axial power shaping rod (APSR) assemblies; (●) measurement of the loss-of-coolant accident restraint boss gaps; (●) measurement of the elevations of the APSR assemblies; (●) cleaning of the plenum and potential interference areas; (●) separation of unsupported fuel assembly end fittings; and (●) movement of the APSR assemblies.

- ***Evaluation: Load Drop***. [82] To ensure that the plant remained in a safe condition, the licensee's safety evaluation considered all planned activities for this task with regard to keeping the reactor core stable. Planned inspection activities consisted of video examination, measurements, and manipulations within the plenum assembly and the core cavity. Stabilization of the core would be assured since the measures to prevent a boron dilution event were prerequisites for inspection activities.

- *Tool/Fuel Assembly End Fitting Drop*. Manipulations within the plenum assembly and the core cavity would be performed with extreme care and within given constraints because of their potential to disturb the core. The dislodging of fuel assembly end fittings would be performed provided that: (•) core inspection revealed that the end fitting was unsupported (no full-length rods existed) and (•) all unsupported end fittings from Batch 1 and Batch 2 fuel were dislodged before dislodging any unsupported end fittings from Batch 3 fuel.

  Implementation of these two operational constraints would ensure that the potential reactivity consequences would be negligible, compared to worst case core configurations analyzed for a postulated head drop accident presented as an attachment to the previous safety evaluation report (SER) [83] for the removal of the reactor vessel head. Core topography revealed, and video inspection would verify, that the APSR assemblies were unsupported and could therefore be driven into the core region with expectations that core disruptions would not be more severe than the dislodging of unsupported end fittings. If the APSR assemblies could not be driven into the core, then attempts would be made to withdraw them into the plenum assembly, provided that they were observed to have no full-length rods. Otherwise, their vertical travel would be limited to 5 inches. The inadvertent drop of tooling was not expected to disrupt the core to an extent greater than the dislodging of unsupported end fittings.

- *Shield Plate Drop*. Since accessibility to the plenum was obtained by removing an internals indexing fixture (IIF) platform shield plate, the consequences of dropping a shield plate were evaluated. The shield plates, which varied in size (6 to 25 square feet) and in weight (910 to 3600 pounds, assuming 1 inch of lead shielding), would be removed one at a time.

  - *Shield Plate Movement*. The removed plates were planned to be staged either on the IIF platform or in the refueling canal. The handling and moving of the plates would follow procedures or work instructions. If the plates were staged in the refueling canal, load paths would be defined in the applicable procedure, work instruction, or work package. These load paths would be administratively controlled to ensure that a postulated drop would not compromise plant safety or the structural integrity of the refueling canal floor. The consequences of a postulated drop of an IIF shield plate above the reactor cavity impacting the IIF platform were evaluated as described below.

  - *Drop Evaluation/Results*. The IIF platform, which totally covered the reactor vessel, could collapse if a dropped IIF shield plate impacted the IIF platform. The platform cover rested about 7.2 feet above the plenum cover plate. Assuming 1 inch of lead shielding, the IIF platform weighed about 19 tons. The total impact energy onto the plenum cover, assuming collapse of the IIF platform, was conservatively determined to be about 276,000 foot-pounds. This was conservative because most of the weight was in the lead shielding, which was located below the IIF platform cover (giving a shorter drop distance). In addition, mechanistically, this collapse of the IIF platform would not impact the plenum with a single sudden impact but rather with a sequence of lower energy impacts, which would result in a less severe total loading on the plenum. The previous SER [84] for the removal of the reactor vessel head presented the analysis of a postulated drop of the reactor vessel head assembly onto the plenum. The impact energy afforded by the dropped reactor vessel head assembly was calculated to be about 457,000 foot-pounds. The head drop bounded the IIF platform collapse for the following reasons: (●) the total impact energy due to collapse of the IIF platform was less than that due to the head drop, and (●) the impact area of the collapsed IIF platform was much greater than the point load drop considered in the head drop analysis resulting in a more distributed loading on the plenum.

The results of the previous postulated head drop analysis included the following: (●) With a boron concentration of 3500 parts per million, the core remained subcritical even for the worst case reconfiguration of the fuel. (●) The structural integrity of the reactor vessel and its support skirt was not compromised. (●) The resulting reactor vessel displacements did not cause stresses on attached piping to exceed their faulted condition stress limits given in Section III of the American Society of Mechanical Engineers *Boiler and Pressure Vessel Code*, 1974 Edition, precluding failure of attached piping.

- ***NRC Review: Load Drop***. [85, 86] The NRC issued two SERs; the first covered the first five activities (see the purpose of this section), and the second covered the remaining three activities. The first SER was brief and did not specifically address this topic, although it did state that the first five activities were previously addressed in the NRC's SERs [87, 88] for Quick Look video inspection of the reactor core and for the reactor vessel underhead characterization. The NRC concluded that the licensee's experience with conducting core and plenum video inspections and other in-vessel activities (e.g., radiation measurements, reactor coolant sampling) demonstrated that these were benign activities (i.e., their environmental impacts were very small) that posed little risk to the onsite workers or offsite public. The NRC further stated that the corresponding plenum inspection activities did not warrant further review.

During plenum removal preparatory activities, IIF platform shield plates would be moved to allow access to the plenum. None of these plates weighed more than 2400 pounds. The NRC's second safety evaluation for the remaining three activities conservatively assumed that the shield plate was dropped. The drop was assumed to cause the IIF platform to collapse, which caused core disruption and release of gaseous radioactivity. The worst case assumption for this scenario was that all the krypton-85 remaining in the fuel was released. The total activity of the remaining krypton-85 was conservatively estimated at 37,000 curies, a quantity less than the 44,000 curies of krypton-85 that was released to the environment in the controlled purge of June 1980. In the unlikely event that a shield plate was dropped, the containment building could be isolated, and the purge system would be secured as necessary to contain the release of krypton-85. Any gaseous activity that evolved from the reactor coolant could then be purged in a controlled process as was done in 1980.

The NRC believed that the likelihood of a shield plate drop and subsequent core disruption that would result in a significant krypton-85 release during the proposed activities was extremely small. The licensee indicated that the planned movement of shield plates would be restricted to those areas previously approved in NRC SERs [89, 90] for the movement of heavy loads and that the number of shield plate movements would be minimized. The postulated IIF platform collapse due to a shield plate drop was bounded by the previously analyzed reactor vessel head drop. The head drop would impart greater impact energy to the plenum and also result in a more concentrated load. In the previous SER [91] for the removal of the reactor vessel head, the NRC concluded the following: (●) the reactor coolant system integrity would be intact; (●) potential gaseous releases of radioactivity to the environment would be well below allowable limits; (●) subcriticality would be maintained; and (●) decay heat removal capability would be available. These conclusions also applied to the less severe postulated shield plate drop (lower weights and lower heights).

The NRC concluded that there were adequate measures and preparatory activities to mitigate the consequences of postulated accidents during plenum removal.

7.4.7 Plenum Assembly Removal

- ***Purpose***. To remove the plenum assembly from the reactor vessel to gain access to the core region for defueling. The plenum was moved to the deep end of the fuel transfer canal that contained reactor coolant for shielding.

- ***Evaluation: Load Drop (Reactor Vessel with Platform)***. [92] The licensee's safety evaluation addressed heavy load handling over the reactor vessel with the installed internals indexing fixture (IIF) platform. To ensure that the plant remained in a safe condition, all planned activities for this task were evaluated with regard to maintaining the stability of the reactor core. Stability of the core could be compromised by either: (●) a dilution of the boron concentration in the reactor coolant system (RCS) that caused the boron concentration to fall below the concentration limit required to maintain the core in a subcritical condition; or (●) a loss of RCS water that would uncover the core. The safety evaluation provided the following guidelines for handling heavy loads over the IIF platform:

    - *General*. All load handling activities to be performed over the IIF platform (before its removal from the IIF) were evaluated against the criteria in the safety evaluation report (SER) and described below. An analysis was performed of postulated load drops onto the IIF platform with the plenum assembly supported on jacks. This analysis examined several categories with resulting load weight versus maximum lift height limitations. (Note: Lift heights were measured from the top of the IIF platform to the bottom-most portion of the suspended load.) Loads that would be handled above the IIF platform were divided into four categories. The guidelines for the four categories were used during removal and replacement of the platform cover plates for access to the plenum area.

- *Load Categories*. The first two categories bounded the handling of objects that could fit through a 24-inch by 48-inch opening and weighed up to 5500 pounds. Category 3 included loads up to 5500 pounds that could not fit through a 24-inch by 48-inch opening. Category 4 included the handling of objects less than or equal to 1800 pounds that could be handled over an open slot and would fit through the opening.

  – *Plate Movement*. Any number of plates could be removed at one time as long as they were handled in accordance with Category 4 loads when over open slots in the IIF platform. Plates that weighted in excess of 1800 pounds, such as the triangular plates, should be handled the same as those in Category 3. Plates being relocated on the IIF platform could be placed only on plates that were installed in the platform. Plates could not be stacked more than two high, including the installed plate. The applicable procedure or unit work instruction for the handling of the load would then define the maximum allowable load lift heights.

  – *Category 1:* This category included objects that when rotated in any orientation could fit through a 24-inch by 48-inch opening. Compliance with the maximum lift heights would ensure that, in the event the load was dropped, neither the IIF platform nor any of the individual platform plates would collapse. The 24-inch by 48-inch opening size represented the largest single opening in the IIF platform within the perimeter of the IIF cylinder. The maximum lift heights provided in the SER ranged from 17 feet for a 500-pound object to 1 foot for a 3000-pound object.

  – *Category 2*. This category included objects greater than or equal to 10 feet in length and able to pass through a 24-inch by 48-inch slot if turned on end. Loads less than 3000 pounds could be treated under Category 1. Loads between 3000 and 5500 pounds could not be lifted more than 1 foot, while loads greater than 5500 pounds were excluded and needed to be rigged horizontally. Compliance with these guidelines would ensure that in the event the load was dropped, neither the IIF platform nor any of the individual platform plates would collapse.

  – *Category 3*. This category included objects that, when rotated in any orientation, would not fit through a 24-inch by 48-inch opening. Under this category, the SER provided maximum lift heights ranging from 59 feet for a 500-pound load, to 1 foot for a 5500-pound load. Compliance with these guidelines would ensure that the IIF platform did not collapse. However, it was possible that, due to the configuration of objects in this category, an IIF platform plate could be deformed sufficiently to allow it to be driven through its own slot onto the plenum.

    If an IIF platform plate (maximum weight 1800 pounds for the largest plate, which could be driven through its own slot) was pushed through the IIF platform, the plate would fall through the water in the IIF and onto the plenum assembly. Because of the shape of the plate, even after the plate was deformed, the water would reduce the velocity where the plate impacted the plenum assembly. In addition, the plate was likely to strike the control rod guide tubes that were above the top of the plenum assembly. Therefore, the impact load transmitted to the plenum assembly would be considerably reduced. The 55-ton plenum assembly was currently supported about 7.5 inches above its normal or rest position on four jacks. These jacks were each rated at 50 tons and transferred all loads to the core support assembly and reactor vessel flange. Since the postulated impact energy would be minimal, the evaluation considered the likelihood of the jacks failing or the transfer of any significant impact energy to the lower grid assembly to be extremely unlikely. On this basis, the SER concluded that an IIF platform plate falling onto the plenum assembly would not have sufficient impact energy to result in the breaking of an in-core instrument guide tube.

An analysis of a postulated drop of the reactor vessel head assembly onto the plenum was presented in the previous SER [93] for the removal of the reactor vessel head. This analysis demonstrated that the total impact energy of about 457,000 foot-pounds onto the plenum assembly would not compromise the structural integrity of the reactor vessel and its support skirt. In addition, resulting reactor vessel displacements did not cause stresses on attached piping to exceed their faulted stress limits given in Section III of the American Society of Mechanical Engineers *Boiler and Pressure Vessel Code*, 1974 Edition, thus precluding failure of attached piping. In comparison, the drop of an 1800-pound plate onto the jacked plenum would impart less than 10,000 foot-pounds or about 2 percent of the energy of the postulated head drop.

– *Category 4*. This category included objects less than or equal to 1800 pounds that could be handled over an open slot and could fit through the open slot. The SER provided two maximum lift heights: 1 foot for a 200 to 1800-pound load and 4 feet for a load less than 200 pounds. Loads to be handled over an open slot were limited to the largest single platform cover plate that could be postulated to fall through the open slot during removal or tools that would be operated from the IIF platform level. The maximum lift height for the platform cover plate during removal was restricted to 1 foot. This height was adequate for removal. The maximum lift height for tools was 4 feet. This height was adequate to allow lifting the tool over any temporary safety rails placed over the open slot.

The evaluation for objects in this group was bounded by the 1800-pound plate that could be driven through its own slot as presented for Category 3. Because of the configuration of the triangular plates, they could not be dropped or driven through their own slots and strike the plenum.

- ***Evaluation: Load Drop (Reactor Vessel without Platform)***. [94] The licensee's safety evaluation stated that the handling of heavy loads over the reactor vessel following the removal of the IIF platform would be governed by the licensee's SER [95] for heavy load handling over the reactor vessel. This SER was the controlling document for all heavy load handling activities occurring above or in the reactor vessel including plenum lift. This SER demonstrated that a drop into the vessel of any heavy load up to the limits described in the SER would not impact the health and safety of the public.

- ***Evaluation: Load Drop (Fuel Transfer Canal)***. [96] A postulated drop of the plenum assembly was considered in the licensee's safety evaluation to determine if such a drop into the shallow end of the fuel transfer canal (FTC) on the 322-foot elevation level could cause a rupture of the in-core instrumentation guide tubes that were routed within the in-core instrumentation cable chase on the 282-foot elevation. The resulting point impact loading on the 322-foot elevation floor afforded by a 12-foot drop of the plenum assembly onto the floor was assessed. This assessment determined that, because of the floor thickness and geometry, the impact load would be transferred to the primary and secondary shield walls mostly in shear. Local concrete spalling at the underside of the floor was judged not to occur, and even if minor spalling were to occur, there would be no damage to the in-core tubes for the following reasons: (•) the location of the floor and walls with respect to the in-core instrument trench would preclude a direct hit by debris generated by local spalling, and (•) heavy bottom reinforcement was provided (Number 11 reinforcing bars at 6-inch centers each way).

Based on the results of this analysis, the SER concluded that a drop of the plenum assembly in the shallow end of the FTC would not cause any damage to the in-core instrument guide tubes. A postulated drop of the plenum assembly in the deep end of the FTC on the 308-foot elevation would not impact plant safety as the consequences of such a drop would not affect the stability of the core, drain, or reduce the water level in the RCS, or decrease containment building accessibility.

- ***Evaluation: Load Drop (Polar Crane Failure)***. [97] The licensee's safety evaluation considered a mechanical failure of the polar crane or its rigging that could result in a plenum assembly drop. A mechanical or electrical failure could result in the plenum assembly being suspended above the reactor vessel or FTC. Any of these failures could create a radiation hazard for operations personnel (refer to Section 3.1 of the SER for the expected dose rates). Shielding could be provided by filling the FTC to normal refueling level with borated water if recovery from these failures required shielding of the plenum assembly.

This SER demonstrated (refer to Sections 4.1.2 and 4.1.3 of the SER) that the postulated worst-case plenum assembly drops would not uncover the fuel in the reactor vessel or cause criticality. None of these postulated polar crane failures would significantly increase airborne activity levels in the containment building above normal recovery levels or impact containment building integrity. Therefore, polar crane failures would not present undue risk to the health and safety of the public.

- ***NRC Review: Load Drop***. [98] The NRC's safety evaluation considered the plenum lift and transfer activities that would necessitate the handling of heavy loads over the reactor vessel. The primary safety issues arising from a load drop accident included the potential for nonisolatable RCS leakage due to in-core instrument tube failure and the potential for disruption of fuel resulting in the release of trapped krypton-85 gas. These issues were addressed by the licensee's SER [99] for heavy load handling over the reactor vessel, which was subsequently approved [100] by the NRC. This approval specifically addressed the proposed plenum lift and transfer activities. The NRC's approval of the proposed plenum assembly removal, with respect to heavy load handling considerations, was based on the low probability of a plenum drop accident and on the licensee's consequence analysis indicating that the maximum RCS leakage resulting from a plenum drop would be well within the capability of available RCS makeup systems. The NRC evaluation considered drop potential, RCS leakage, RCS makeup contingencies, and drop consequences.

- *Minimize Drop Potential*. Several factors minimized the potential for a plenum drop accident. The polar crane and tripod were successfully used to remove the reactor vessel head, a load weighing more than twice the weight of the plenum assembly and lift rigging (170 tons versus 73 tons). The lifting pendant assemblies were designed in accordance with NUREG-0612 criteria and applicable industry standards. Each had a design rating of 25 tons, with factors of safety of 3 for yield stress and 5 for ultimate stress, and was load tested at 150 percent of rated load. The NRC's onsite office staff, along with an expert consultant, observed the successful mockup testing of the lifting assemblies and verified the operation of the locking and unlocking features of the lifting arm assemblies. In addition, annual preventive maintenance of the polar crane had previously been completed to ensure that the crane was in a satisfactory condition for the lift.

- *RCS Leakage*. The licensee analyzed the consequences of the unlikely event of a plenum drop accident. In the licensee's previous SER (101) for the removal of the reactor vessel head, the NRC concurred with the licensee's assumption that the maximum RCS leakage through in-core instrument tubes damaged by a plenum drop would be about 20 gallons per minute. In the event of a plenum drop, redundant RCS level monitors would detect the leakage early. At the maximum postulated leak rate of 20 gallons per minute and the minimum RCS level at the 321.5-foot elevation, the leak would take over 14 hours for the core to become uncovered, which would give operators adequate time to initiate RCS makeup.

- *RCS Makeup Contingencies*. The available makeup methods included gravity flow from the borated water storage tank (BWST) and recirculation of the containment building sump water. Recirculation capabilities included the option of decay heat removal pumps or the 25-gallon-per-minute submersible pump currently installed in the sump. The defueling water cleanup system pumps, rated at 200 gallons per minute, were also available to provide sump recirculation and could be installed within 24 hours. The minimum volume of borated water available for gravity feed from the BWST was about 220,000 gallons, based on a minimum BWST inventory of 310,000 gallons. At a makeup flow rate of 20 gallons per minute, gravity feed from the BWST could maintain the RCS at a safe water level for about 7.6 days, thereby allowing ample time to select and operate one of the available sump recirculation systems. If a recirculation system was required, sampling capability of the containment building sump water would be available to ensure that a minimum RCS boron concentration of 4350 parts per million was maintained.

- *Radiological Release*. In the previous licensee's SER (102) for the removal of the reactor vessel head, the NRC also concurred with the licensee's bounding analysis for the potential release of krypton-85 due to fuel disruption from a load drop accident. The licensee concluded that the maximum resulting offsite doses from such an unlikely event would be several orders of magnitude below the limits specified in 10 CFR Part 100.

- *Conclusion*. Based on the NRC's safety evaluation of the licensee's SER (103) for the removal of the reactor vessel head, the NRC concluded that the potential for a load drop accident during plenum assembly lift and transfer was extremely remote and that sufficient measures were available to mitigate the consequences of such an unlikely event.

7.4.8 Makeup and Purification Demineralizer Resin Sampling (NA)

7.4.9 Makeup and Purification Demineralizer Cesium Elution (NA)

7.5 Defueling Tools and Systems

7.5.1 Internals Indexing Fixture Water Processing System (NA)

7.5.2 Defueling Water Cleanup

7.5.2.1 Defueling Water Cleanup System

- ***Purpose***. To remove organic carbon, radioactive ions, and particulate matter from the reactor vessel and fuel transfer canal/spent fuel pool "A" (FTC/SFP-A). The defueling water cleanup system (DWCS) actually comprised two independent subsystems (sometimes called trains): the reactor vessel cleanup system and the FTC/SFP-A cleanup system. The DWCS was totally contained within areas that had controlled ventilation and could be isolated. This limited the environmental impact of the system during normal system operations, shutdown, or postulated accident conditions.

- ***Evaluation: Load Drop***. (104) The licensee's safety evaluation considered load handling inside the containment building that consisted of the transfer of the DWCS filter canisters from the deep end of the FTC to the fuel handling building via the fuel transfer system. Load handling within the fuel handling building consisted of the movement of submerged demineralizer system (SDS) ion exchange liners, DWCS liners, DWCS filter canisters, and transfer casks. The heavy load drop analysis for the SDS shipping casks was provided in the previous safety evaluation report (SER) (105) for the control of heavy loads. The DWCS liners would be moved using the existing liner transfer casks for the EPICOR II system. A previous SER (106) for the control of heavy loads inside the containment building addressed the handling of heavy loads in the containment building and in the fuel handling building.

The radiological concerns associated with a load drop of the SDS ion exchange liners and the DWCS liners were bounded by the analysis in the technical evaluation report (107) for the SDS, whereas the radiological concerns associated with a load drop of a filter canister were bounded by the accident analysis in the previous SER (108) for the bulk defueling of the reactor vessel. These analyses showed that the health and safety of the public were not endangered by these hypothetical accidents.

- ***NRC Review: Load Drop***. [109] The NRC's safety evaluation considered the potential need for load handling in the containment building and fuel handling building in support of DWCS construction work. The NRC determined that the potential consequences of these activities were bounded by the safety evaluations previously submitted and approved. These evaluations included the SER [110] for SFP-A refurbishment and the SER [111] for the handling of heavy loads in the containment building. Lifting of heavy loads around DWCS equipment during operation and lifting of DWCS components, such as filters and ion exchangers, would be administratively controlled. These controls would ensure that load handling pathways and potential load drop consequences were within the constraints of the previously approved safety evaluations.

The NRC concluded that the heavy load handling necessary to support DWCS installation and operation could be carried out without undue safety consequences when controlled so that loads handled, lift heights, and load travel pathways were within the bounds of the previously approved safety evaluations. Heavy load handling within SFP-A and the FTC was restricted over fuel containing canisters and would require an additional safety evaluation approval before the heavy load lifts.

7.5.2.2 Cross-Connect to Reactor Vessel Cleanup System (NA)

7.5.2.3 Temporary Reactor Vessel Filtration System (NA)

7.5.2.4 Filter-Aid Feed System and Use of Diatomaceous Earth as Feed Material (NA)

7.5.2.5 Use of Coagulants (NA)

7.5.2.6 Filter Canister Media Modification (NA)

7.5.2.7 Addition of a Biocide to the Reactor Coolant System (NA)

7.5.3 Defueling Canisters and Operations

7.5.3.1 Defueling Canisters: Filter, Knockout, and Fuel

- ***Purpose***. To provide loading, handling, storage of the canisters (filter, knockout, and fuel), and long-term storage of the core debris, ranging from very small fines to partial length fuel assemblies.

- ***Evaluation: Load Drop***. [112] The licensee's safety evaluation stated that canisters were capable of withstanding bounding accidents. Vertical drops of about 6 feet in air followed by 19.5 feet in water, or about 11.5 feet in air, were considered along with a combination of vertical and horizontal drops. These drops were analyzed to bound a drop in any orientation. For these cases, the structural integrity of the poison components must be maintained, and the canister must remain subcritical. Deformation of the canister was acceptable. Although not expected based on the drop test results, leakage of core material from the canister, up to its full contents, was allowed, provided that the contents left in the canisters remained subcritical.

- *Method*. An equivalent drop in air was calculated for the worst case, and this equivalent airdrop was used as the basis for the structural analysis. Structural analysis methods were used to determine the extent of the deformation of the shell and canister internals. Impact velocities were calculated for the specified canister drops. Based on these velocities, strain energy methods were used to compute the impact loads associated with the various postulated drops. Vector combinations of the horizontal and vertical components were used to determine the effect of a drop at any orientation.

  In the vertical drop cases, the same deformation would occur regardless of the canister type, since deformation was shell dependent. Test results from the actual canister drops verified that for the bottom impact, all deformation occurred below the lower support plate in the lower head region. An upper bound shell deformation was computed using a computer code and the results were presented in Figure 3.1-1 of the safety evaluation report (SER) along with the actual test results.

- *Testing (Fuel Canister)*. A dynamic load drop test was performed to address a maximum load drop into a fuel canister as a one-time event. Based on the drop test, the maximum weight that could be permitted to be dropped in air from the top of the fuel canisters into a fuel canister was 602 pounds as a one-time event. For loads dropped from the top of the fuel canister through water starting at zero velocity, the allowable one-time value was 850 pounds (measured in air). For multiple occurrences, when the load was dropped in water as measured in air, the allowable drop weight was limited to 550 pounds.

  The analysis in the test study was based on permitting some permanent deformation of the lower support plate and the support plate/shell weld but not enough to cause damage to the recombiner packets in the lower head. Canister shell stresses remained in the elastic range. The ability to dewater the canister could be affected if the drain tube/bulkhead weld was damaged as a result of a load drop. In such a case, the ability to dewater the canister would be evaluated on a case-by-case basis.

- *Evaluation Results (Filter and Knockout Canisters)*. To determine the consequences of a vertical and horizontal drop of the filter and knockout canisters, their internals were analyzed with finite element methods using a computer program. This analysis incorporated the actual nonlinear properties of the material. Geometric constraints imposed by the shell were accounted for by limiting the displacement of the supports. Results of these analyses for each canister type were as follows:

  – *Filter Canister*. In the filter canister, criticality control was provided by the central boron-carbon poison rod, coupled with the mass of steel in the filter element drain tubes and tie rods. Using the end caps of the filter modules as deflection limiters, the entire tube array deflection was limited to 1.6 inch under postulated accidents. This analysis was conservative because it did not consider the five circumferential bands around the array or the viscosity of the filter cake bed where both would maintain the standard spacing. Using the maximum calculated deformed geometry (before the array bounced back closer to its original position), the criticality criterion given in Section 3.2 of the SER was met. The analysis was performed considering the original 0.5-micron filter media design. The results of the testing [113] showed that filter modules equipped with filter media having the larger pore opening had a greater load-carrying capability, which allowed the licensee to conclude that the deflections discussed above were bounding when applied to canisters using filter modules with filter media having a larger pore opening.

– *Knockout Canister*. In the knockout canister, the central B_4C poison rod coupled with four absorber rods provided criticality control. Results from the structural analysis showed that the poison rods remained essentially elastic during all postulated accidents and the maximum instantaneous displacements were less than 0.75 inch. The minor modifications made to some of the knockout canisters to convert them to "deep-bed" filters (refer to Section 2.3 of the SER) were within the bounds of the values used in the analysis and testing of the knockout canisters. Thus, the deep-bed filters were expected to exhibit structural behavior similar to the knockout canisters during a drop accident. As in the case of the filter canister, the resultant deformed geometry successfully met the criticality criterion given in Section 3.2 of the SER.

o *Evaluation Results (Fuel Canister)*. The fuel canisters, with their square-within-a-circle geometry, exhibited different drop behavior than other canisters. For both the vertical and side drops, the fuel canister internals would not experience significant deformations other than the shell deformations. Lightweight concrete filling the void between the square inner shroud and the circular outer shell provided continuous lateral support to both the outer shell and the shroud. This resulted in a distributed loading function for horizontal drops with no calculated deformation to the shroud shape. Testing demonstrated that the lower support plate remained in place for design drops while supporting a mass equal to the shroud, payload, and the concrete. The lack of significant deformation after a drop meant that criticality analysis for the standard design was applicable to the drop cases as well.

• ***NRC Review: Load Drop***. [114] The NRC review of the licensee's structural analysis determined that proper codes and standards were used in the design of the defueling canisters. The structural analysis showed sufficient margins of safety when applying the maximum predicted loads during normal onsite operations and handling along with subsequent transportation. The structural analysis for accident conditions used industry standards, and the NRC provided reasonable assurance that the maximum expected deformation was factored into the criticality analysis and accepted the analytic techniques.

- *Canister Design Requirements*. [i] The defueling canisters were designed to the requirements [115] of the the American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code*. The canister design pressures were 150 pounds per square inch gauge (psig) internal and 30 psig external. Fabrication, inspection, and testing of the canisters were performed to the standards of the ASME Code. The canisters were nuclear safety rated, and the licensee's procurement specifications required that they be manufactured under the controls of a quality assurance program to meet the quality assurance requirements of Appendix B of 10 CFR Part 50 [116] and American National Standards Institute Standard N45.2, "Quality Assurance Program Requirements for Nuclear Facilities."

  Structural analysis by the canister designers included evaluations of the loads imposed on the canisters during normal operations, as well as postulated load drops and shipping accidents. An acceptance criterion for normal operations was based on the ASME *Pressure Vessel Code*. In addition, analysis was performed to show acceptable safety margins when applying the specified stress factors of NUREG-0612 and American National Standards Institute Standard N14.6, [117] "Radioactive Materials—Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 kg) or More," for the normal handling condition. The design criteria for postulated accident conditions were for the predicted deformed geometry following an accident and required that the canisters and their contents remain subcritical, although leakage of material was permissible.

- *Results (Normal Operations)*. Canister structural analysis for the normal operation and handling condition was performed using standard analytic techniques. This analysis demonstrated acceptable design margins that met the requirements of the ASME code and other applicable regulatory requirements, industry codes, and standards.

- *Methods (Accident Conditions*). The approach used to demonstrate that the canister design met the specification for the postulated accident conditions used a combination of analytical methods and component testing. The accident conditions considered included transportation accident and load drop.

  - *Transportation Accident*. The design specifications for the shipping cask intended for transporting the filled defueling canisters were to limit the loads imposed on the canisters to no more than 40 g-force axial and 100 g-force lateral during hypothetical transportation accidents in accordance with 10 CFR Part 71, [118] "Packaging and Transportation of Radioactive Material." A detailed evaluation of the proposed cask conformance to this specification was performed and included for both analysis and impact testing of a scale model. This evaluation was under review by the NRC's Transportation Certification Branch as part of the licensing process for the cask. Analysis and supporting drop tests of the canister were performed to demonstrate that the fixed poisons installed in the canister remained intact and capable of performing their intended

[i] Editor's Note: If codes and standards that are cited in the source documents are complete (e.g. title, issue date), then they will be listed in the endnote section; otherwise, they will be only cited in the text to this NUREG/KM.

criticality control function when subjected to these loads, or that subcriticality could be maintained by other geometrical constraints.

  – *Onsite Accident (Load Drop)*. For onsite handling accidents, canister drops of 6 feet 11 inches in air followed by 19 feet 6 inches in water, or 11 feet 7 inches in air, were considered to be credible. This did not include a potential drop in the truck bay of the fuel handling building during cask loading. This potential canister drop would be evaluated in the fuel shipping SER. Combinations of vertical and horizontal drops were considered. These drops were determined to impart loads on the canisters in excess of those for the transportation accident. Structural analyses were performed to determine the extent of the canister shell and internals deformation resulting from these loads. Vector combinations of the vertical and horizontal load components were used to predict the effects of a drop in any orientation. The conservatively modeled worst case deformed geometry for each type of canister was factored into the criticality analysis.

- o *Results (Load Drops)*. Results of load drop evaluations for vertical and horizontal drop conditions included the following:

  – *Results (Vertical Drop)*. Deformation of the canisters due to a vertical drop was found to be shell dependent according to data analysis from a drop test program. The predicted deformation in this case was a bulging of the canister shell below the lower support plate. No significant deformation of the canister internals, significant to the criticality analysis, was expected to occur from a purely vertical drop. This was demonstrated during actual drop tests for a bottom end impact. This also bounded the top end impact, and for purposes of criticality analysis, the deformed shape was assumed to exist at both ends of the canister.

  – *Results (Horizontal Drop)*. For the horizontal drop case, the filter and knockout canisters' internals were analyzed with finite element methods using a computer code. The analysis incorporated the actual nonlinear properties of the material and accounted for geometric constraints imposed by the canister shells. The criticality calculations used the deformations predicted by these analyses with additional conservatisms on poison structure locations. The deformed geometry for the fuel canister was determined by a 30-foot drop of a simulated partial-length unit. The testing showed insignificant deformation of the neutron absorber material shroud from the lateral loads imposed.

7.5.3.2 Replacement of Loaded Fuel Canister Head Gaskets

- ***Purpose***. To replace head gaskets on the loaded fuel canisters in spent fuel pool "A" (SFP-A) because of excessive leakage. The replacement of gaskets involved: (●) moving a loaded canister from the SFP-A storage rack to the dewatering station rack with the canister handling bridge; (●) removing the canister head; and (●) transferring the head to a worktable. At the worktable: (●) the head was rotated for access to the gaskets; (●) the metallic gaskets were removed; (●) new synthetic gaskets were installed; and (●) the head was inspected before reinstallation.

- ***Evaluation: Load Drop***. [119] To perform the gasket replacement, the canister was first moved to the dewatering station. The licensee's safety evaluation stated that the handling of canisters in the fuel handling building was previously evaluated in its safety evaluation report (SER) [120] for heavy load handling inside containment. At the dewatering station, the canister head would be removed and placed on a worktable that was attached to the dewatering system platform shield wall. The combined weight of the canister head and worktable (about 300 pounds) would be considerably less than the maximum weight of a fully loaded canister (3355 pounds); therefore, the consequences of dropping the head and worktable into SFP-A were bounded by the evaluations in the heavy load handling SER. Additionally, appropriate measures would be taken to ensure that the canister head was completely disengaged before being removed from the pool, thus preventing an inadvertent lifting of an open canister.

- ***NRC Review: Load Drop***. [121] The NRC's safety evaluation concluded that the heavy load handling and criticality control aspects of the proposed activity were bounded by the agency's SER [122] for the defueling canisters and its SERs [123, 124] for fuel canister storage racks.

7.5.3.3 Use of Debris Containers for Removing End Fittings

- ***Purpose***. To use modified fuel canisters as "debris containers" for removing fuel assembly upper end fittings, control component spiders, or other structural material from the reactor vessel. This activity was performed to expedite access to the vacuumable fuel and debris in the core. The modified fuel canister did not have internal neutron-absorbing plates, concrete filler, recombiner catalyst, dewatering capability, or a relief valve. After the debris containers were loaded, they would be closed and stored in the spent fuel pool "A" racks until final dispositioning of the containers and their contents. There were no plans to use these debris containers for shipment. Since these canisters would not have relief valves installed (a prerequisite for shipping), they could be easily identified.

- ***Evaluation: Load Drop***. [125] A loaded debris container filled with water weighed about 2000 pounds, considerably less than the maximum weight of a loaded fuel container (3350 pounds). All lifting and handling of the container would use equipment and procedures for the maximum weighted fuel canister. The licensee's safety evaluation concluded that the proposed activity presented no additional lifting and handling safety concerns.

- ***NRC Review***. [126] Editor's Note: The NRC's safety evaluation report did not specifically address this evaluation topic.

7.5.3.4 Fuel Canister Storage Racks

- ***Purpose***. To provide storage for the three different types of canisters (fuel, filter, and knockout) filled with debris material from the reactor vessel. Storage for 263 canisters was available in the racks located in spent fuel pool “A” and in the deep end of the fuel transfer canal (FTC).

- ***Evaluation: Load Drop***. [127] The licensee’s safety evaluation stated that the fuel canister storage racks (FCSRs) were designed to withstand the impact energy of a postulated defueling canister drop. Drop loads other than from canisters were controlled in accordance with maximum lift heights and handling requirements.

    - *Drop Height*. The maximum design drop height from the canister transfer shield (CTS) was about 6 feet in air and 6.5 feet in water for a drop in the FTC and for a drop in the spent fuel pool. Damage caused by such an accidental drop was local and would not cause a reduction in the spacing between canisters to less than the 17.3 inches used in criticality analyses for the defueling canister. The dropping of the CTS was not considered a credible event as the CTS was an integral part of the canister handling bridge.

    - *Drop Configuration*. The FCSRs and the CTS were configured to prevent a canister falling from the transfer shield to drop horizontally on top of the FCSRs. In addition, the storage racks were designed to preclude a canister from falling between canister positions. The design of the rack included provisions to prevent a canister that was dropped outside of the rack from leaning against the rack or rolling against the rack. This safety feature ensured that the side of the dropped canister was no less than 4 inches from the side of the nearest canister in the rack.

    - *Drop on Empty FCSRs*. The dropping of heavy loads on the FCSRs without canisters present (in either the fuel pool or the FTC) posed no safety concern as there was no opportunity for a criticality event, radiation release, or uncovering of fuel. Furthermore, the FCSRs were constructed in such a way that should damage occur, the damaged FCSR section could be removed and repaired or replaced.

    - *List Loads of FCSR Pieces*. During the installation or replacement of sections of the FCSRs, the potential existed for dropping a section in a load handling accident. To avoid creating any potential safety concerns related to a load handling accident, the sections of the FCSR to be installed in the containment would be handled within the limits set in the safety evaluation report (SER) [128] for heavy load handling inside containment.

    - *Lift Loads Other Than Canisters*. The handling of loads when canisters were in the racks, other than canisters over the FCSRs in the fuel pool or deep end of the FTC would be restricted to loads less than the design load drop for the FCSRs (3355 pounds). In addition, the load lift height would be limited so the potential energy would be less than that of a suspended fuel canister. The following equations would be used to determine the maximum plant elevation (h, maximum plant elevation in feet) where a given weight (W, weight in pounds and less than 3355 pounds) could be raised over the FCSRs in the fuel handling building (FHB) or containment building (CONT). The two equations presented as the canister lift heights for the FHB and CONT, which were slightly different, are [h_{FHB} = ((37,000 x W) + 321)] and [h_{CONT} = ((37,000 x W) + 322)]. [j]

- ***NRC Review: Load Drop***. (129, 130) The NRC's safety evaluation considered load handling for three cases: (●) load drops associated with fuel rack installation activities; (●) load drops over the installed fuel racks before use without loaded fuel canisters; and (●) load handling over the racks with loaded fuel canisters present.

  - *During Installation*. The consequences of load drops during installation activities included the effects of impacts on the floor slabs and other safety-related systems and components not directly associated with the FCSRs. The consequences of such load drops in the fuel handling building were bounded by the analysis performed in the NRC's SER (131) for spent fuel pool "A" refurbishment. The consequences of load handling accidents during rack installation in the containment building were bounded by the analysis in the NRC's SER (132) for heavy load handling in the containment building. Installation activities would be procedurally controlled to ensure that the weight of loads handled, load lift heights, and load travel pathways were constrained to stay within the bounds of these previously approved safety evaluations.

  - *Over Racks (before Use)*. Load drops over the racks after installation but before their use for fuel storage presented no significant hazards as a result of rack assembly failure since there would be no material present that could produce a release of radioactive material. In addition, damage to rack modules in this event could be repaired without undue risk to plant workers. Load handling in these situations would be controlled in such a manner that postulated load drop consequences for other systems or components would be bounded by the currently approved load drop analyses.

  - *Over Racks (with Loaded Canisters)*. The structural design of the racks was sufficient to withstand the postulated drop effects of a filled fuel canister (3355 pounds) without gross failure of the rack. Damage would be limited to the area of impact and would not result in a reduction of the 18-inch center-to-center spacing of adjacent fuel canisters. However, the effects of such a drop on the fuel canisters in storage were not fully evaluated. Those effects would be evaluated in the licensee's technical evaluation report for the fuel canister.

  - *Conclusion*. The NRC concluded that heavy load handling necessary to support FCSR installation and load handling over the racks before their use for storage of fuel could be carried out without undue safety consequences. This conclusion assumed that load handling was controlled in such a manner that the loads handled, lift heights, and load travel pathways were within the bounds of the previously approved safety evaluations. Load handling over the fuel racks with loaded fuel canisters was not fully analyzed and would be addressed in subsequent safety evaluations.

---

[j] Editor's Note: These canister lift height equations were the corrected equations from Revision 1 of the SER.

- *Conclusion (Update*). [133] Revision 1 of the licensee's SER contained corrections in the equations for determining the allowable lift height of loads handled over the racks. The NRC agreed with the licensee's conclusion that the lift height equations in the revised SER were acceptable to ensure proper load lift height control. Further, the NRC determined that revisions in the proposed SER did not alter the conclusions in the NRC's previous SER.

7.5.3.5 Canister Handling and Preparation for Shipment

- ***Purpose***. To transfer defueling canisters from spent fuel pool "A" (SFP-A) to the shipping cask for offsite shipment. Defueling canisters were transferred to locations within the containment building and fuel handling building using a transfer shield. The transfer of canisters to the shipping cask used a different device called a "fuel transfer cask."

- ***Evaluation: Load Drop (NUREG-0612 Compliance)***. [134] The licensee's safety evaluation demonstrated compliance with the general recommended guidelines in Section 5.1.1 of NUREG-0612 for all heavy load handling activities associated with canister handling and preparation for shipping offsite. The heavy load handling cranes used for these activities were the fuel handling building crane (both main and auxiliary hoists), the mini hot cell (MHC) jib crane, and the canister lifting mechanism within the fuel transfer cask. The canister handling bridge was used to transfer the canisters from the storage racks to the dewatering station and from there to either the fuel transfer cask loading station or back to the storage racks. The safety evaluation report (SER) [135] for the bulk defueling of the reactor vessel addressed this transfer.

The following specific evaluations corresponded to each of the items described in Section 5.1.1 of NUREG-0612: (●) safe load paths; (●) procedures; (●) crane operators; (●) special lifting devices; (●) standard lifting devices; (●) crane inspection; (●) testing and maintenance; and (●) crane design. In addition, NUREG Section 5.1.5 required assurance that a load drop impacting the floor would not damage safe-shutdown equipment located under the floor.

- *Safe Load Paths*. A safe load path was defined for the movement of heavy loads associated with canister transfer and preparation for shipment within the fuel handling building. The designated transfer cast load path minimized the potential for damage to defueling canisters and other equipment important to safety.

  – *Avoided Pathways*. The load path avoided the following areas: (●) defueling canisters staged in SFP-A; (●) SFP-A/SFP-B transfer gates; (●) submerged demineralizer system equipment in SFP-B; (●) ion exchangers in the defueling water cleanup system; and (●) the ventilation system for the fuel handling building.

  – *Damage Reduction*. The following measures were taken to minimize potential damage during movement of heavy loads: (●) Loads were lifted high enough to avoid any obstructions along the load path. (●) Heavy loads, including the fuel transfer cask, were lowered into the truck bay area south of the rail bay. In particular, the fuel transfer cask was lowered over its storage stand or the shipping cask, which were more than 20 feet from the exclusion zone in the truck bay. (●) The potential for a load hangup that involved the fuel transfer cask was reduced by minimizing hoist movements by the fuel handling building crane while moving the bridge/trolley. (●) Loads of up to 3000 pounds could be lifted no more than 10 feet above the floor level in the exclusion zone. (●) All heavy load paths were identified in the appropriate procedures or unit work instructions. These procedures or instructions contained a plant layout drawing that was marked to clearly define the safe load path handling area for the designated load. (●) Safe load paths would be clearly marked on the floor, and the lift supervisor would ensure adherence to the safe load paths. (●) The lift supervisor was qualified by procedures and had no responsibilities other than directing the lift operation. (●) Any changes to these safe load paths required a revision to the appropriate procedures or unit work instructions.

- *Procedures*. For heavy load handling operations over or in proximity to irradiated fuel or safe-shutdown equipment, approved procedures or unit work instructions governed the lift operation. These approved procedures and instructions contained the sequence of steps necessary to complete the lift operation and included approved rigging sketches. The rigging sketch identified the load, the maximum weight of the load, and the equipment to be used. The fuel handling building crane was operated in accordance with an approved procedure.

- *Crane Operators*. Overhead crane operators were trained and qualified in accordance with an approved procedure in accordance with American National Standards Institute (ANSI) Standard B30.2-1976, "Overhead and Gantry Cranes." [(136)] The conduct of overhead crane operators was governed by procedures in accordance with the ANSI standard.

- *Special Lifting Devices*. The fuel transfer cask and MHC used special lifting devices to interface with the main hoist of the fuel handling building crane and the jib crane, respectively. The MHC could also be handled by the auxiliary hoist of the fuel handling building crane. These and any other special lifting devices that would be identified at a later date would be designed, constructed, inspected, and tested in accordance with ANSI Standard N14.6, [(137)] "Radioactive Materials—Standard for Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 kg) or More." Safety factors were based on the combined maximum static and dynamic loads that could be imparted to the special lifting devices (excluding seismic loads).

- *Standard Lifting Devices*. Lifting devices that were not specially designed were installed and used in accordance with the guidelines of ANSI Standard B30.9-1971, "Slings," [(138)] and used in accordance with the TMI-2 lifting and handling program. The sling rating was based on the sum of the static and maximum dynamic loads (not including seismic loads).

- *Crane Inspection, Testing, and Maintenance*. Licensee procedures governed the inspection and preventive maintenance of the fuel handling building crane and complied with ANSI Standard B30.2-1976. The fuel handling building crane was load tested using a load that was 125 percent of the 110-ton rated capacity. The preaccident load test was performed before the issuance of ANSI Standard B30.2-1976. The newly assembled jib crane was load tested in accordance with ANSI Standard B30.11-1980, “Monorail Systems and Underslung Cranes.” [139] Maintenance was performed in accordance with ANSI Standard B30.11-1980 for jib cranes and ANSI Standard B30.16-1981, “Overhead Hoists,” [140] for hoists.

- *Crane Design (Fuel Handling Building Main and Auxiliary Hoists)*. The maximum loads listed for cranes in this section applied only to the activities within the scope of this SER. The design of the canister lift mechanism in the fuel transfer cask was described in the licensee’s SER [141] for bulk defueling of the reactor vessel.

  – *Compliance with Newer Specifications*. The fuel handling building crane was built before the issuance of ANSI Standard B30.2-1976 and Crane Manufacturers Association of America (CMAA) 70-1975, “Specification for Top Running Bridge and Gantry Type Multiple Girder Electric Overhead Traveling Cranes.” [142] The licensee’s design evaluations compared the crane design to the more recent specifications to show that the crane design met the intent of the ANSI standard. The NRC had previously accepted these evaluations for Three Mile Island, Unit 1 (TMI-1).

  – *Modifications to Crane Systems*. As a result of these evaluations, modifications were made to the crane restart protection criteria from CMAA 70 (control switches to include a spring return to the “off” position) and to the crane electrical interlock system to inhibit load handling in the east portion of the truck bay. To further improve the reliability of the fuel handling building crane, the following modifications were made: (●) a second upper limit switch was added; (●) an emergency stop button was added; and (●) a remote-control pendant was relocated.

  – *Crane Safety Factors*. The safety evaluation provided the safety factors for structural (load-bearing) components of the fuel handling building crane. Loads to be handled by the auxiliary hoist were limited to less than 6.6 tons to maintain a minimum safety factor of 6 to yield. The safety factors for the main hoist were given for a design load of 110 tons and the fuel transfer cask load including a maximum weight canister and load block (about 22.8 tons). The safety factors for the auxiliary hoist were given for a design load of 15 tons and for the defueling operations limit of 6.6 tons. The safety factors for rated loads and allowable loads of the main and auxiliary hoists were presented in the SER for each of the crane structural components (wire rope, mechanical drive train gearing, hook forging, cast steel drums and sheaves, major structural components, and welds). The safety factors were based on best engineering estimates by the crane manufacturer.

  – *Seismic Qualification*. Section 9.7.1.6 of the TMI-1 final safety analysis report [143] stated that the structural design of the fuel handling crane was also required to ensure no loss of function while lifting a rated load, during and after a seismic Category 1 event.

- *Crane Design (MHC Jib Crane)*. The MHC jib crane handled the MHC between the shipping cask loading collar and the storage location on the adjacent platform in the fuel handling building loading bay. This crane could also be used to handle other heavy loads. The MHC jib crane was designed in accordance with ANSI Standard B30.11 and CMAA 74-1974, “Specification for Top Running and Under Running Single Girder Electric Overhead Cranes Utilizing Under Running Trolley Hoist.” (144) The MHC jib crane was additionally designed so that the safe-shutdown seismic event could not cause the collapse of the crane or otherwise impart unacceptable loads over the exclusion area or other safe-shutdown areas. The structural safety factors imposed by the specifications, afforded at least a 10:1 margin to ultimate material strength based on the approximate design load of 7.5 tons.

- *Load Drop Impacting the Floor*. Where the safe-shutdown equipment had a ceiling separating it from an overhead handling system, NUREG-0612, Section 5.1.5(2), required an evaluation to show that the largest postulated load handled by the handling system would not penetrate the ceiling or cause spalling that could cause failure of the safe-shutdown equipment. The licensee’s analysis demonstrated that should the jack fail while jacking the railcar, the maximum possible dynamic load applied to the rail by the wheels would not exceed the capacity of the floor.

• ***Evaluation: Load Drop (Design Features)***. (145) The description of heavy load handling equipment, along with the evaluation of load handling equipment and activities, that was presented in the licensee’s safety evaluation demonstrated that the potential for a load drop was extremely small. Key design and operational features included: (●) Heavy load handling devices were conservatively designed in accordance with industry standards. (●) Heavy loads would be transported along a designated load path that minimized the potential for damage to equipment important to safety. (●) Cranes were inspected, tested, and maintained in accordance with industry standards and were operated by qualified personnel. (●) The jib crane support structure was designed to withstand the impact of the fuel transfer cask into the shipping cask at 1 foot per minute horizontally or up to 5 feet per minute vertically, without causing the failure of the jib crane support structure or loss of the shipping cask restraint function. (●) This impact was assumed to occur with the shipping cask in the upright position and restrained by the jib crane support structure.

• ***Evaluation: Load Drop (Accidents)***. (146) Even though the potential for a load drop was very small, the licensee’s safety evaluation considered the following consequences of dropping a canister from the fuel transfer cask:

- *Drop into Loading Station*. If the canister and grapple were to drop when lifting a canister from the cask loading station into the cask, the canister would fall back into the loading station canister guides. No unacceptable damage to either the cask loading station or spent fuel pool would result from this drop. The distance the canister would fall was less than what it was designed for. Therefore, dropping a canister while it was being raised into the cask resulted in no unacceptable consequences to equipment.

- *Drop during Movement*. Before moving the fuel transfer cask from the cask loading station, the door on the bottom of the cask was closed and remained closed until the cask was positioned over the shipping cask cell where the canister was lowered. The door was designed to withstand the drop of the canister and grapple. Thus, the consequences of dropping the canister during movement from the spent fuel pool to the shipping cask would not result in unacceptable damage to any equipment.

- *Drop into Shipping Cask*. A drop of a canister and grapple into the shipping cask could occur when the fuel transfer cask was positioned over the shipping cask cell for canister lowering, with the door open on the bottom of the fuel transfer cask. This load drop was within the bounds of the design-basis vertical drop for the defueling canisters, so the canisters and contained poisons would still meet design requirements (final design technical report for the defueling canisters) following this drop. Thus, there was no potential for a criticality event resulting from the drop of a defueling canister into the shipping cask. Because of the presence of the impact limiters, damage to the shipping cask from a canister/grapple drop was not considered credible. If a canister dropped into the shipping cask, both the canister and shipping cask would be evaluated to verify their acceptability for shipment.

- *Leakage Following Drop*. The design features of the canisters and the handling equipment presented the potential for a very small leak. It was expected that no leakage would occur under design drop conditions. If leakage did occur as a result of dropping a canister into the shipping cask, the resulting offsite dose would be bounded by the analysis discussed in the radiological release section of this SER.

- *Damage to Floor Slab*. Further, an analysis was performed to determine the effects on the truck bay floor slab from the drop of a defueling canister/grapple into the shipping cask. This analysis showed that the structural integrity of the slab would not be impaired. Furthermore, little, if any, damage to the floor slab would result from this drop.

- ***NRC Review: Load Drop***. [147] The NRC's safety evaluation stated that most of the significant aspects of physical handling and loading defueling canisters into the shipping cask were related to the structural design and integrity of equipment during heavy load handling. The NRC review concluded that the licensee had invoked appropriate industrial codes, standards, and specifications in the design of the equipment to ensure that canister handling and preparation could be performed safely for shipment.

- *Defueling Canisters*. The defueling canisters were designed and fabricated as coded pressure vessels in accordance with American Society of Mechanical Engineers, *Boiler and Pressure Vessel Code*, Section VIII, "Rules for Construction of Pressure Vessels." The canisters were designed to withstand the effects of unrestrained drops of 6 feet 11 inches in air, followed by 19 feet 6 inches in water, or 11 feet 7 inches in air, and still maintain fuel debris confinement in a critically safe geometry. Such performance bounded all postulated canister drops during handling except for a potential drop from the fuel handling building overhead crane in the truck bay. The detailed structural evaluations of the canisters were provided in the licensee's technical evaluation report [148] for the defueling canisters and the NRC's SER [149] for the defueling canisters.

- *Canister Integrity*. Final canister weights were verified to ensure that the canisters conformed to the design limits factored into the canister structural and criticality analysis and to ensure that cask loading conformed to the requirements of the certificate of compliance for the Model 125-B shipping cask. This verification of weight also provided a check on canister integrity by confirming that there was no water in-leakage during storage in the spent fuel pool. The canisters were weighed by the weighing systems integral to the canister handling bridges. The licensee would implement administrative procedures that provided for adequate determinations and documentation of canister tare (unladen) weight, loaded weight, and dewatered weights to ensure conformance to the applicable loading specifications from the shipping cask certificate of compliance and the licensee's technical evaluation report for the defueling canisters and the associated NRC SER.

- *Shipping Cask*. The shipping cask was designed to the requirements of 10 CFR Part 71 and applicable industrial codes and standards. The safety analysis report on the Model 125-B shipping cask and the NRC-approved cask certificate of compliance [150] with the attached safety evaluation presented detailed evaluations.

- *Cask Unloading Station*. The cask unloading station was designed in accordance with ANSI Standard N14.6, "Standard for Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 kg) or More for Nuclear Materials," and was designed to accommodate the effects of both static and dynamic loads. The system complied with Section 6 of ANSI Standard N14.6 because the lifting jack was designed so that no single failure would cause an uncontrolled towering of the load. In the unlikely event of total catastrophic failure of the unloading station, the result would be a drop of the loaded cask, which weighed about 80 tons without the impact limiters, a distance of about 5 feet to the fuel handling building floor with no compromise of the package integrity. The cask would be positioned so that this load drop event could occur only outside of the load handling exclusion area of the fuel handling building. This load handling exclusion area was imposed because of the presence of redundant electrical circuits beneath the floor that were required to be operable to ensure the safe-shutdown capability of Unit 1. Prohibiting load handling in this area prevented floor impacts that could potentially impair both circuits. Consequently, an event of this kind would not cause failure of safety-related equipment that would result in loss of required safe-shutdown functions in Unit 1.

- *Cask Hydraulic Lift Assembly*. The lift assembly was designed with redundant hydraulic cylinders in which either one was capable of restraining the full weight of the cask. In the event of hydraulic system failure, the cylinders were provided with hose break valves. These valves were essentially excess flow check valves that prevented uncontrolled lowering of the cask following a loss of hydraulic pressure.

- *Mini Hot Cell*. The MHC was a small, shielded transfer cask used to remove and install the shield plugs from the top of the seven canister-holding cavities in the shipping cask. The hot cell jib crane was designed in accordance with ANSI Standard B30.11 and had a design safety factor of 10:1 to ultimate material strength based on a 7.5-ton load rating. The lifting system integral to the hot cell was designed to ANSI Standard B30.16, with a safety factor of

10:1 to ultimate material strength when used for handling a single shield plug. The fuel transfer cask hoisting system was also designed to ANSI Standard B30.16, with appropriate safety factors applied to the load-bearing components when handling the intended loads.

- *Fuel Handling Building Crane*. The NRC evaluated the crane against the requirements of NUREG-0612. The evaluation concluded that the crane was acceptable for heavy load handling. The details of this evaluation were documented in a previous NRC safety evaluation. [k] In addition, the licensee had defined load travel pathways to minimize the potential for dropped loads impacting components important to safety.

- *Conclusion*. Based on the above evaluations, the NRC concluded that the design of the fuel handling equipment associated with the canister handling and preparation for shipment program was adequate to ensure that the probability of a load drop was extremely small. In addition, based on a previous NRC safety evaluation, the potential releases of radioactive material that could result from a related load handling accident would produce offsite doses well within the limits of 10 CFR Part 100.

- ***NRC Review: Load Drop (Update)***. [151] Later NRC safety evaluations considered subsequent revisions [152, 153] to the licensee's SER.

Section 2.3.1 of the licensee's SER was revised by eliminating the requirement to leave canisters in the spent fuel pool for 2 weeks after initial dewatering to verity canister integrity. The NRC's safety evaluation concurred with this change because the process of purging and pressurizing the canisters with inert gas, followed by visual monitoring for gas leakage, should provide adequate assurance of canister integrity. The integrity was confirmed later by weighing the canister before shipment as a final check for water in-leakage during storage.

Section 3.2 was revised to remove the restriction on the total number of canisters that could exceed the 2300-pound canister weight limit. Since this restriction was included originally because of physical constraints at INEL rather than because of onsite or transportation-related safety issues, the NRC's safety evaluation concurred with the licensee's proposal to remove the restriction provided that the licensee comply with any loading restrictions requested by INEL or imposed by the certificate of compliance for the Model 125-B shipping cask.

7.5.3.6 Canister Dewatering System (NA)

7.5.3.7 Use of Nonborated Water in Canister Loading Decontamination System (NA)

7.5.4 Testing of Core Region Defueling Techniques (NA)

7.5.5 Fines/Debris Vacuum System (NA)

[k] Editor's Note: The letter cited in the SER as "NRC letter, Docket No. 50-289, J. Stolz to H. Hukill, dated January 11, 1985," could not be found in public or nonpublic ADAMS.

7.5.6 Hydraulic Shredder

- ***Purpose***. To use a hydraulically powered shredder to reduce the size of fuel pins and other core debris and to facilitate the loading of fuel canisters or debris buckets.

- ***Evaluation: Heavy Loads***. [154] The licensee's safety evaluation considered the installation of the shredder that necessitated the movements of loads over the reactor vessel and internals indexing fixture. These loads included a number of defueling work platform plates (to gain greater access into the reactor vessel), the shredder, its support structure, and its mounting carriage.

    - *Guidelines*. All movements of loads in the containment and over the reactor vessel would be conducted in accordance with the guidelines given in the safety evaluation report (SER) [155] for heavy load handling inside containment and the SER [156] for heavy load handling over the reactor vessel. Adherence to those guidelines would ensure that the potential for a load drop was minimized and that the consequences of any postulated load drop were within the bounds of the aforementioned SERs, which were both approved by the NRC.

    - *Load Weight*. A postulated failure of the shredder support structure during operation would result in an impact loading of about 7300 pounds onto the debris bed. The projected loading was based on the following estimates: (●) shredder and motor (3800 pounds); (●) support structures and auxiliary components (2200 pounds); and (●) core debris in shredder and transfer container (1300 pounds).

    - *Load Height*. The impact energy onto the rubble bed was a function of the distance from the bottom of the shredder to the rubble bed and the weight of the dropped components. During initial operation, the shredder would be close to the surface or the rubble (less than 4 feet), but the distance would increase as core debris material was removed. Additional sections of the support columns were available to lower the shredder in 4-foot increments to minimize the distance from the shredder to the rubble. Thus, the farthest distance from the shredder to the rubble was expected to be about 8 feet. The maximum number of support structures was included in the weight estimates above. Any postulated failure of the support structures was bounded by the SER for heavy load handling over the reactor vessel performed for loading accidents.

    - *Drop Prevention*. The most critical time for a load drop to occur was during installation and removal of the shredder. During lifting operations over the reactor vessel, efforts were made to minimize the potential for drops into the vessel that could affect the integrity of the reactor vessel in-core nozzles. The licensee concluded that load drop during installation was unlikely due to the following drop prevention measures: (●) The lifting rig for the shredder was designed with a capacity of about 7000 pounds, which was greater than the estimated weight of the shredder, motor, inlet, and outlet chutes, and supports. (Note: The shredder would not contain significant quantities of core debris nor would the debris container be attached during installation or removal.) (●) The 7000-pound design weight included a 1.15 dynamic load factor. (●) Using a detailed finite element analysis, results determined that the lifting rig had safety factors of greater than 3 and 5 to yield and ultimate, respectively. (●) The lifting rig was load tested to 200-percent capacity followed by an examination of the structural welds. This satisfied the NUREG-0612 requirements for special lifting and handling devices. (●) The shredder would be installed using either the containment building service crane or the polar crane auxiliary hook. (●) Relevant work instruction would include an approved rigging diagram. (●) Either crane had adequate rated capacity for installation of the shredder.

- *Maximum Lift Height and Conclusion*. Potential loadings directly onto the reactor vessel and supports as a result of a load drop were bounded by the SER for heavy load handling over the reactor vessel provided that the bottom of the shredder remained at or below the 333-foot elevation during transfer. If the impact was transmitted directly onto the debris bed, the limiting consequence would be the failure of in-core nozzle welds. This event was also evaluated in the safety evaluation mentioned above. Thus, potential structural failures or load handling accidents were bounded by previously approved evaluations.

- ***NRC Review***. [157] Editor's Note: The NRC's SER did not specifically address this topic.

7.5.7 Plasma Arc Torch (NA)

7.5.7.1 Use of Plasma Arc Torch to Cut Upper End Fittings (NA)

7.5.7.2 Use of Plasma Arc Torch to Cut the Lower Core Support Assembly (NA)

7.5.7.3 Use of Plasma Arc Torch to Cut Baffle Plates and Core Support Shield (NA)

7.5.7.4 Use of Air as Secondary Gas for Plasma Arc Torch (NA)

7.5.8 Use of Core Bore Machine for Dismantling Lower Core Support Assembly (NA)

7.5.9 Sediment Transfer and Processing Operations (NA)

7.5.10 Pressurizer Spray Line Defueling System (NA)

7.5.11 Decontamination Using Ultrahigh Pressure Water Flush

- ***Purpose***. To use ultrahigh pressure water flush at 20,000 to 55,000 pounds per square inch to remove surface coatings and surface contamination inside the containment building.

- ***Evaluation: Load Drop***. [158] The licensee's safety evaluation considered decontamination of surfaces and components by the use of ultrahigh pressure water flush that could necessitate the movement of heavy loads at various locations within the containment. Any heavy loads would be handled in accordance with the licensee's safety evaluation report [159] for heavy load handling inside containment or be evaluated on a case-by-case basis and be subject to NRC approval.

- ***NRC Review: Load Handling***. [160] The NRC's safety evaluation stated that heavy load handling related to the ultrahigh pressure decontamination program would be performed in accordance with approved procedures to ensure that operations conformed to the bounding conditions of the NRC-approved load handling safety considerations. The NRC concluded that the licensee's proposed program did not present the potential for damage to components from load handling that could result in any undue risk to the health and safety of the public.

7.6 Evaluations for Defueling Operations

7.6.1 Preliminary Defueling

- ***Purpose***. To allow movement of debris within the reactor vessel, to allow installation of defueling equipment, and to identify core debris samples for later removal and analysis by INEL. The movement of debris within the vessel was prepared for early defueling and required some rearrangement of debris in the reactor vessel before the actual removal of fuel. These activities included the loading of small pieces of core debris into debris baskets but did not include actual loading of fuel debris into fuel canisters.

- ***Evaluation: Heavy Loads***. [161] The licensee's safety evaluation stated that handling of loads over the reactor vessel would be in accordance with the licensee's safety evaluation report [162] for heavy load handling over the reactor vessel.

- ***NRC Review: Load Drop***. [163] The NRC's safety evaluation concluded that: (●) the potential for a load drop accident during the proposed limited defueling activities was remote; (●) adequate controls existed to restrict lift heights; and (●) measures were available to mitigate the consequences of a postulated accident.

  - *Worst Case Accident*. The NRC's safety evaluation report [164] for heavy load handling over the reactor vessel was applicable to the proposed preliminary defueling activities restricted to the reactor vessel. In that evaluation, the NRC determined that the worst case accident resulting from a postulated heavy load drop over the reactor vessel would be the simultaneous failure of the 52 in-core instrumentation tubes resulting in a total leakage rate of 20 gallons per minute.

  - *Accident Mitigation*. In a previous NRC safety evaluation [165] for a change in the recovery technical specifications, the agency concluded that reliable sources of borated makeup water would be available to substantially exceed the worst case reactor coolant system leakage rate of 20 gallons per minute. Either the gravity feed from the borated water storage tank or the containment building sump recirculation system would supply the makeup water.

- *Exposure*. In the safety evaluation report for heavy load handling over the reactor vessel, the NRC approved a bounding analysis performed by the licensee that assumed an instantaneous release of all unaccounted for krypton-85 from the core due to a heavy load drop. This conservative analysis resulted in a dose of 9.7 millirem to the whole body for an individual located at the site boundary and a whole-body dose of 1.8 millirem to an individual located at the low-population zone boundary. These doses were several orders of magnitude below the accident limits of 10 CFR Part 100.

- *Lift Controls*. In the heavy load lift evaluation, the NRC also approved the licensee's proposed lift height/weight matrix that would preclude failure of the defueling work platform due to postulated load drops of certain defueling equipment.

7.6.2 Early Defueling

- ***Purpose***. To remove end fittings, structural materials, and related loose debris that would not involve removal of significant amounts of fuel material, to remove intact segments of fuel rods and other pieces of core debris, and to remove loose fuel fines (particles) by vacuuming operations.

- ***Evaluation: Load Drop***. [166] Editor's Note: The licensee's safety evaluation for early defueling was practically identical to its subsequent safety evaluation report (SER) [167, 168] for bulk defueling; therefore, this section does not include the evaluation text. Please refer to the next section for details.

- ***NRC Review: Load Drop***. [169] The licensee's safety evaluation stated that handling of heavy loads during early defueling activities was addressed by the licensee's SER [170] for handling of heavy loads over the reactor vessel and its SER [171] for handling of heavy loads inside containment. In the NRC's SER for heavy load handling over the reactor vessel, the worst case accident, identified for all anticipated loads lifted over the vessel through completion of defueling, was the postulated drop of the plenum assembly. For this bounding case, the licensee postulated that a drop of the plenum assembly would cause the simultaneous failure of all 52 in-core instrumentation tubes, resulting in a total reactor coolant system leakage rate of 20 gallons per minute. As described in the NRC's SER [172] for preliminary defueling activities, the agency concluded that reliable sources of borated makeup water would be available to substantially exceed the worst case reactor coolant system leakage rate and that adequate leak detection capability was provided.

The licensee also performed a bounding analysis to calculate offsite doses due to an instantaneous release of all unaccounted for krypton-85 resulting from a load drop. This conservative analysis yielded offsite doses several orders of magnitude below the accident limits specified in 10 CFR Part 100. In addition, the licensee developed a lift height/weight matrix to control the lifting of defueling equipment so that a load drop would not cause the defueling platform to collapse. As specified in the NRC's SER for heavy load handling over the reactor vessel, the licensee would analyze alternative load paths in determining that the best pathway for movement of a heavy load was over the vessel.

Based on the evaluations, the NRC concluded that the licensee met the requirements of NUREG-0612, implemented adequate measures to prevent a heavy load drop over the reactor vessel, and took adequate measures to mitigate the consequences of a potential load drop accident over the vessel during early defueling activities.

7.6.3 Storage of Upper End Fittings in an Array of 55-Gallon Drums

- ***Purpose***. To use shielded 55-gallon drums to store end fittings and other structural material removed from the reactor vessel.

- ***Evaluation: Load Drop***. (173) The licensee's safety evaluation noted that each storage container weighed about 3000 pounds when loaded and would be positioned on the defueling platform with a lifting rig. The lifting rig would be load tested to 200 percent of the loaded weight. Restrictions on lifting this weight would be observed. When the container was filled, it would be lifted out of the protective lead shield of the platform, moved to the 347-foot elevation of the containment building, and positioned near the reactor head storage area. Once the container was properly located, the crane would be remotely decoupled from the container and moved back to the defueling platform for the next container. Two postulated accident conditions that were evaluated included a loaded storage container dropped onto the containment floor and a heavy load dropped onto the containers in storage.

  - *Loaded Container Dropped*. In the unlikely event that a loaded storage container was dropped onto the containment floor, the evaluation assumed that the container would be damaged, and its contents spilled onto the floor. Since each storage container had a maximum of 30 kilograms of fuel, criticality was not a concern as the fuel available for spillage was less than the 70 kilograms required for a criticality event. Pyrophoricity concerns and releases due to a dropped storage container were bounded by the analysis performed for a dropped fuel canister in the safety evaluation report (174) for bulk defueling.

  - *Load Drop onto Containers*. The potential for a heavy load drop onto the containers existed when the containers were in storage. In the event of a heavy load dropped into the storage area, the evaluation assumed that all the storage containers in the area at the time of the drop were destroyed. As mentioned previously, 70 end fittings, each containing 2 to 3 kilograms of fuel, would be stored in the containers. Thus, on a worst case basis, if all the containers were destroyed, 210 kilograms of fuel could be spilled. These 210 kilograms of fuel would pose a potential for criticality. However, this potential was determined to be small enough to be considered nonexistent.

    This conclusion was based on the following: (●) Fuel on the floor with 4350 parts per million borated water would be subcritical. (●) Fuel on a dry floor (i.e., no moderator) would be subcritical. (●) Fuel mixed with unborated water could become critical; however, unborated water at sufficient depth on the 347-foot elevation of the containment building would be highly unlikely. (●) Fuel in the end fittings was assumed to be within the flow space of the end fittings. Attempts would be made to remove this fuel from the flow space before its transfer to the storage container; therefore, it was unlikely that fuel remaining in the flow space would be dislodged by a heavy load drop. The fuel in the end fittings flow space would be subcritical in any type of water.

- *Conclusion*. The licensee concluded that a criticality due to the release of fuel from all storage containers as a result of a heavy load drop was unlikely. Releases due to a heavy load drop onto the storage containers were bounded by the analysis performed for a dropped fuel canister in the licensee's safety evaluation reports (175, 176) for bulk defueling.

- ***NRC Review: Load Drop***. (177, 178) The NRC's safety evaluation noted that the effects of dropping a 55-gallon drum filled with end fittings or dropping a load onto a stored drum were bounded by previously analyzed fuel canister drop accidents.

7.6.4 Defueling (Also Known as "Bulk" Defueling)

- ***Purpose***. To develop defueling tools and to conduct activities necessary to remove the remaining fuel and structural debris located in the original core volume. An additional activity was to address vacuuming of debris from the lower head of the vessel, a primary activity associated with lower head defueling.

- ***Evaluation: Load Drop***. (179) The licensee's safety evaluation stated that heavy load handling in both the containment building and fuel handling building was addressed in the licensee's safety evaluation report (SER) (180) for load handling in the containment building, with the exception of heavy loads handled over the reactor vessel. Heavy loads handled over the reactor vessel were addressed in the licensee's SER (181) for load handling over the reactor vessel. Handling of the hydraulic shredder was specifically addressed in the licensee's SER (182) for the use of the hydraulic shredder for defueling.

- ***NRC Review: Load Drop***. (183) The NRC's safety evaluation stated that the safety consideration of load drops was adequately addressed in previous NRC safety evaluations, and that the NRC's earlier conclusions were applicable to the proposed activities. Editor's Note: Refer to the NRC's SER (184) for early defueling.

7.6.5 Use of Core Bore Machine for Bulk Defueling (NA)

7.6.6 Lower Core Support Assembly Defueling

- ***Purpose***. To dismantle and defuel the lower core support assembly (LCSA) and to partially defuel the lower reactor vessel head. Structural material removed from the LCSA included the: (●) lower grid rib assembly; (●) lower grid forging; (●) distribution plate; and (●) in-core guide support plate. The lowest elliptical flow distributor and the gusseted in-core guide tubes would not be removed under this proposed activity. The use of the core bore machine, plasma arc cutting equipment, cavitating water jet, robot manipulators, automatic cutting equipment system, and other previously approved tools and equipment was included in this activity and associated safety evaluations.

- ***Evaluation: Load Drop***. [185] The licensee's safety evaluation assessed two events: (●) dropping a large piece of the lower grid forging and (●) dropping a small LCSA piece though a 6-inch elliptical flow distributor hole. The safety evaluation also reviewed load handling controls.

  - *Lower Grid Forging Pieces*. Sections of the LCSA were cut into pieces using the plasma arc torch. These pieces were then rigged and lifted out of the reactor vessel. The heaviest pieces that could be removed were two halves of the lower grid forging. These pieces weighed about 9000 pounds. Should the forging drop from an elevation just above the guard rails of the defueling work platform, this piece would fall slightly less than 50 feet to impact the in-core guide support plate. The shear stress developed by impact within the smallest ligament of the in-core guide support plate would exceed the ultimate stress of the plate material. However, the energy of the fall would likely be absorbed in the support plate, leaving little excess energy to cause a similar failure of the elliptical flow distributor. Therefore, the integrity of the reactor vessel was expected to be maintained.

    As an alternate to this removal procedure, the grid forging could be cut into smaller pieces for easier removal from the vessel and to facilitate storage. Should this option be used, the weight of the assumed falling plate would be less than 9000 pounds. Further, the evaluation assumed that since other pieces of the LCSA would weigh less than the lower grid forging, the dropping of such pieces onto the remaining LCSA structures, which were identical or stronger than the in-core guide support plate, would not result in structural failure of the LCSA.

    As the pieces of the grid forging were removed, they would have to be lifted over the upper tubes of the remaining in-core guide tubes. The remaining in-core guide tubes, however, would be bolted to the in-core guide support plate and welded to the elliptical flow distributor. Since the shear strength of the gusset welds, the elliptical flow distributor welds, and the bolting at the support plate exceeded the compressive strength of the 1-15/16-inch-diameter in-core guide tube above the support plate, shearing the in-core guide tube from the elliptical flow distributor was determined not to be likely. Further, based on the small target provided by an in-core guide tube [l] stub above the support plate, an impact from above the in-core guide tube stub would likely cause the stub to bend at or near the nut holding the guide tube into the in-core guide support plate. Therefore, the evaluation concluded that the in-core guide tube could not impact the reactor vessel lower head or a degraded in-core nozzle, as it would remain welded to the elliptical flow distributor.

[l] The in-core guide tube is separated from the in-core reactor vessel nozzle by a few inches.

- *Elliptical Flow Distributor Holes*. Removing all the plates above the elliptical flow distributor would open many 6-inch-diameter holes that would allow direct access to the reactor vessel lower head. The licensee's and the NRC's safety evaluation reports [186, 187] for use of the core bore machine for dismantling the LCSA showed that potential load drops of objects less than 6 inches in diameter, including dropped in-core guide tubes, would not impart excessive stresses to intact in-core nozzles. Further, objects less than 6 inches in diameter that fell through a hole in the elliptical flow distributor would not impact a potentially degraded in-core nozzle since these nozzles were not lined up with the open holes. The potentially degraded in-core nozzles were also shielded from dropped objects by the gusseted in-core guide tubes that still remained in the elliptical flow distributor. Consequently, the evaluation concluded that dropped objects during the LCSA disassembly and defueling process could not strike potentially degraded in-core nozzle welds; therefore, the potential for reactor vessel leakage due to dropped loads was remote.

- *Load Handling Controls*. The potential for load drop accidents into the reactor vessel was also minimized by careful control of load handling activities and the use of conservatively designed and tested load handling equipment. Load handling activities were performed in accordance with approved procedures for such activities. Each specific load handling activity was controlled by a work instruction or procedure. Load handling activities would be performed by personnel who had been trained and qualified for these activities. To preclude the potential for a load drop on a potentially damaged in-core nozzle, the elliptical flow distributor plate and the gusseted in-core guide tubes would be left in place for this phase of LCSA disassembly and defueling.

- ***NRC Review: Load Drop***. [188] The NRC's safety evaluation stated that the removal of gusseted in-core guide tubes and the elliptical flow distributor was not included in the scope of this safety evaluation. These two structures formed part of the protection for heavy load drops inside and over the reactor vessel during LCSA defueling.

7.6.7 Completion of Lower Core Support Assembly and Lower Head Defueling

- ***Purpose***. To remove the elliptical flow distributor and gusseted in-core guide tubes of the lower core support assembly (LCSA) and to defuel the reactor vessel lower head (RVLH).

- ***Evaluation: Load Drop (Overview)***. [189] The licensee's safety evaluation considered the potential for direct load drops on the RVLH and the in-core nozzles during LCSA and RVLH defueling. Initially, the RVLH and in-core nozzles would be subject to a potential load drop during severing of the upper in-core guide tubes. Should the guide tubes be severed, the tube would be free to fall and impact the RVLH debris bed, an intact in-core nozzle, or the lower head proper. In addition, portions of the elliptical flow distributor could fall on the lower head.

- *Drop Targets*. After completing core bore machining operations, the in-core guide tubes could be removed from the reactor vessel. This would create 52 approximate

6.5-inch-diameter holes in the LCSA, each centered over an in-core nozzle. These holes would provide a direct path for relatively small objects (i.e., long-handled tools) to impact an in-core nozzle. The remaining sections of the LCSA would be cut into pieces using the plasma arc torch. After the LCSA pieces had been cut from the main LCSA structure, they would be rigged and lifted out of the reactor vessel. After cutting the elliptical flow distributor, a hole would be formed in the LCSA, exposing a large area of the RVLH to a variety of different load drops.

- *Evaluation*. Calculations demonstrated that a load drop on an undamaged in-core nozzle weld would not result in a nozzle weld failure (refer to Appendix A to the safety evaluation report (SER) for details]. The SER analysis of reactor vessel integrity (Section 4.11) demonstrated a high probability that the in-core nozzle welds had maintained their original integrity during the 1979 accident. Additionally, there was a low probability that the configuration of a heavy load drop would directly impact an in-core nozzle weld. Therefore, the potential for reactor vessel leakage due to dropped loads was remote.

- *Operating Experience*. There were two instances of dropped loads involving the canister sleeve during defueling within the reactor vessel. Investigations by the licensee identified the cause of these drops as a failure of the sleeve-locking drive retention spring that maintained the locking bar in position. To eliminate this potential failure mechanism, a new positive acting locking bar retention mechanism would be designed and installed before the elliptical flow distributor was cut.

- *Load Handling Control*. The potential for other load drop accidents into the reactor vessel would be minimized by careful control of load handling activities and the use of conservatively designed and tested load handling equipment. Load handling activities were performed in accordance with approved procedures for such activities. Each specific load handling activity was controlled by a work instruction or procedure. Trained and qualified personnel would perform these load handling activities.

- ***Evaluation: Load Drop over the Reactor Vessel (Appendix A)***. [190] During core support assembly and lower head defueling, pieces of the LCSA would be cut from the assembly and removed from the reactor vessel to allow access to core debris. Eventually, a hole would be created through the LCSA that would expose a large area of the reactor lower head to direct impact from heavy loads. The licensee performed an evaluation to better determine the potential damage that could be incurred by the in-core nozzles because of dropped loads. Simple calculations were used to ascertain if more complex analyses were warranted.

- *Potential Load Drops*. The following objects were considered for potential accident loads: (●) light-duty fuel handling pole; (●) end effector handling tool; (●) loaded defueling canister; and (●) loaded defueling canister in the sleeve. (Refer to Table A in Appendix A to the SER for the maximum achievable drop heights in air above the reactor vessel and the water inside the vessel.)

- *Analysis Assumptions*. To maintain a simple approach, the analyses made the following major assumptions:

  – *Kinetic Energy*. Upon impact, all kinetic energy of the falling object would be transmitted to the instrumentation nozzle and result in strain. This assumption was conservative since some of the energy would also be converted to strain in the dropped object and the RVLH.

  – *Compressive Stress-Strain*. The compressive stress-strain curve for a short column of Inconel was identical to the tensile stress-strain curve. This assumption was conservative since ductile metals would fail in tension before failing in compression without buckling.

  – *Static Stress-Strain*. The static stress-strain curve for Inconel was appropriate for dynamic loadings. Since some metals exhibit higher strength but lower ductility with increasing load application speeds, this assumption could be slightly nonconservative.

  – *Strain*. The strain was uniform over the entire nozzle. This assumption did not account for the possibility of the nozzle bending. The use of this assumption gave an upper bound on the permissible drop heights.

  – *Material Properties*. As-constructed material properties were used for the nozzle and weld materials. However, nozzle material properties could have been degraded by elevated temperatures during the accident.

  – *Drag*. When dropped through water, the objects would be subject to drag, which could vary significantly, depending on the orientation of the falling object relative to the direction of movement. An examination of various sharp-edged bodies indicated potential drag coefficients varying from 0.5 to 1.5, which indicated a significant effect on the calculated impact velocity for a waterdrop height of 30 feet or more. In lieu of actual calculation of drag coefficients for all dropped objects, the drag coefficients range from 0.5 to 1.5 was used.

- *Permissible Drop Heights*. Assuming that the impact load was entirely in the axial direction and along the centerline of the nozzle, an upper bound on the permissible drop heights could be established. The analysis conservatively assumed that all the kinetic energy of the impacting object would be absorbed in the nozzle. Since the nozzle's stress-strain curve was known, the limiting impact velocity could be determined. Knowing the impact velocity allowed the determination of the drop heights by iteration. The allowable drop heights were calculated and Table B in Appendix A to the SER presented the following results (●) cross-sectional drop area; (●) maximum strike velocity; (●) airdrop height for both drag coefficients; and (●) waterdrop height for both drag coefficients.

  A comparison of the calculated allowable drop heights in Table B versus the maximum potential heights previously given in Table A showed that even for the very low drag coefficient (0.5) for objects A and B (i.e., the light-duty pole and the end effector handling tool, respectively), the potential drop heights did not exceed the allowable drop heights.

The loaded defueling canister, with the minimum drag coefficient exceeded the allowable waterdrop height by about 2 feet (34.1 feet verses 36.6 feet), and with the canister positioning system sleeve, the minimum drag coefficient exceeded the allowable height by about 4 feet (19.5 feet versus 24 feet). Based on the maximum drag coefficient of 1.5, both objects had potential drop heights less than allowable drop heights.

- *Dropped Canister*. A more realistic evaluation of the criteria for the dropped fuel canister indicated that when the loaded canister was in a "droppable" position, it was within the canister positioning system sleeve, within the port of the defueling work platform, or over the port in the defueling work platform. For each of the positions where the load could drop, the canister would strike the canister positioning system, thereby decreasing its velocity. Further, the assumption that all of the impact energy would be transmitted to the in-core nozzle was highly conservative given that the fuel canister would absorb most of the shock due to its design (a vessel with a thin shell of 0.25 inches thick). In all likelihood, dropping the fuel canister on one end, onto the in-core nozzle, would result in significant bending and possibly puncture of the bottom head of the defueling canister with little or no deflection of the in-core nozzle. Consequently, only the loaded canister in the sleeve did not satisfy the drop criteria.

- *Operating Experience*. The canister sleeve handling tool and the canister positioning system both had locking devices to prevent the drop of a loaded canister and sleeve. The locking device on the canister sleeve handling tool would be verified to be engaged before lifting the canister and sleeve. The locking device on the canister positioning system would be verified to be engaged after the canister sleeve was positioned on the canister positioning system. Unfortunately, in spite of this verification, canisters and sleeves were dropped twice. Investigations as to the cause of these drops indicated that the leaf spring, which held the locking device in place, failed. The licensee noted that it was difficult to detect this hardware failure from the defueling work platform by sight alone.

  To preclude future drops of the fuel canister and sleeve combination, a new, positive-acting locking bar retention mechanism would be designed, tested, and installed before cutting the elliptical flow distributor. In the future, the position of this retention mechanism would be observed to ensure that the locking device remained in the locking position. Consequently, the drop of a loaded canister and sleeve should have a low probability of occurrence.

- *Load Strike (Off-Center of Nozzle)*. All of the analyses of potential load drops considered that the dropped tool struck the exposed in-core nozzle on centerline. Realistically, the impacting object could strike the nozzle off-center, creating both an axial load and a bending moment. An impact load on the nozzle taper would produce a lateral load, and an additional moment would be created. The magnitudes of the lateral load and bending moment were difficult to establish. However, by using the energy approach and simple inelastic equations for the deflection of an end-loaded cantilever beam, the maximum energy absorbed could be comparable to the “axial load only” condition.

Analysis determined that the nozzle was capable of absorbing a side load of about 6 percent of an axial load. If a substantial part of the postulated impact energy was applied horizontally, the nozzle was likely to fall. However, such a failure would be expected to be above and parallel to the inside surface of the RVLH. Therefore, nozzle failure due to off-center loading could fail the nozzle but not cause significant leakage since the segment of the 0.75-inch Schedule 160 Inconel pipe in the reactor vessel and its weld would likely remain intact.

- *Reactor Vessel Penetration Seal Integrity*. The greatest load transmitted to the vessel would be from an axial impact load on the in-core instrument nozzle. Since the nozzle outer diameter above the vessel wall (i.e., 2 inches) was greater than the reactor vessel penetration diameter (about 1 inch), the nozzle would have to shear through the vessel wall to punch a hole through the lower head. The ultimate axial stress capability of the nozzle was well below the ultimate strength of the vessel wall, so the nozzle would fail before the lower head was penetrated. An undamaged nozzle, therefore, could not be pushed through the vessel wall.

  Of the potential failure mechanisms, the evaluation concluded that the worst case anticipated in-core nozzle failure mechanism was shearing at the inside surface of the RVLH. As previously noted, the 0.75-inch Schedule 160 portion of the instrument tube that penetrated the vessel wall was welded directly to the vessel wall. The 2-inch outside diameter in-core instrument nozzle was welded separately to the vessel wall and the 0.75-inch pipe. Failure of the nozzle was unlikely to cause failure of the 0.75-inch pipe to vessel weld that provided the penetration seal. For conservatism, however, the evaluation assumed that this weld failed as a result of the postulated load drop accident.

  Failure of the tube-to-vessel-wall weld would not result in the tubes being forced out of the lower head by the head of water in the vessel. The tubes consisted of Schedule 80 stainless-steel pipe and were supported at the floor below the vessel. The maximum clearance, considering manufacturing tolerance, between the outside diameter of the tube and the inside diameter of the bore in the vessel wall was 0.005 inch. There was insufficient flexibility in the tubes to allow them to drop the 5.5 inches required to fall free to the bottom of the vessel head.

- *Ex-Vessel Failure*. In-core tube failure outside of the vessel was not considered credible. The only credible leakage path from the vessel following a heavy load drop was through the annulus around the tube penetrations through the vessel wall. This leakage was previously calculated to be about 0.40 gallon per minute per nozzle penetration. A previous safety evaluation for a recovery technical specification change request (No. 46) [191] demonstrated the capability to provide borated water makeup to the reactor coolant system in excess of 17 gallons per minute, even in the event of a loss of offsite power.

- ***NRC Review: Load Drop***. [192] The NRC's safety evaluation noted that the agency had previously reviewed a progressive series of licensee submittals on defueling. Most of the equipment, techniques, and safety issues in the subject licensee's SER were previously reviewed. The principal consideration involved in the SER for the completion of defueling was the removal of a portion of the elliptical flow distributor. This presented the potential for the defueling equipment and dropped loads to interact with the in-core instrument penetrations and RVLH.

  - *Impact Area*. Observations made during previous defueling efforts showed little damage to the in-core instrument penetrations and none to the lower head. Since many of the penetrations and much of the lower head were hidden under core debris, the potential for damage could not be precluded. Thus, the potential area of interaction could be intact or partially degraded. In addition, adequate forces could be generated from defueling equipment or a dropped load to shear an intact penetration if applied horizontally or obliquely. The potential for damage to the lower head due to jet impingement and ablation by molten material during the accident was limited to the area beneath fuel assemblies R6 and R7 and the area outside the core baffle plates.

  - *Mitigation*. In the unlikely event of a complete shear of a penetration, an annular gap would exist between the in-core instrument string and the lower head. The maximum leakage through this annular gap would be 0.4 gallon per minute for each sheared penetration. This leakage rate would be well within the licensee's capability to supply makeup water to the reactor coolant system using gravity feed or pumping. If an unspecified mechanism provided adequate force to push the instrument string through the lower head, a 1-inch-diameter hole and a 120-gallon-per-minute leak could result. Active pumping of borated water would be required to maintain the reactor vessel level. Keeping the reactor vessel level would not be required to maintain subcriticality or to protect the health and safety of the public. However, radiation and airborne activity could limit access to the containment building, and fuel debris could be flushed to the reactor vessel cavity.

  - *Conclusion*. The NRC concluded that the proposed activities could be accomplished without significant risk to the health and safety of the public provided that they were in accordance with the limitations stated in the licensee's and the NRC's SERs. The NRC restricted activities near the area of potential ablation of the lower head to preclude the creation of a leakage path larger than 1 inch and to keep fuel particle size and total mass within the bounds of the licensee's criticality analysis. Further, the NRC required that the elliptical flow distributor be left intact outward of the midline of the "P" row of fuel assemblies to protect the portion of the lower head immediately below from potential load impacts. If later visual inspection verified that the lower head in this area was undamaged or that erosion was less than 0.5 inch in depth, this restriction would be removed.

7.6.8 Upper Core Support Assembly Defueling

- ***Purpose***. To cut and move the baffle plates and to defuel the upper core support assembly (UCSA). This evaluation addressed the following activities: (●) cutting the baffle plates for later removal; (●) removing bolts from the baffle plates; (●) removing the baffle plates, and (●) removing core debris from the baffle plates and core former plates.

- ***Evaluation: Load Drop***. [193] The licensee's safety evaluation stated that during UCSA defueling, the reactor vessel lower head and in-core nozzles would be subject to potential direct load drops not evaluated in previous safety evaluation reports (SERs). Five scenarios were evaluated, and these evaluations demonstrated that postulated load drops would not result in a nozzle weld failure. The potential for other load drop accidents was minimized by careful control of load handling activities.

  - *Drop During Assembly or Disassembly*. The first and second potential load drops (i.e., manual tool positioner and manual tool positioner with assembled mast and torch) were introduced during the assembly and disassembly of the plasma arc cutting equipment. The installation of the manual tool positioner through the T-slot of the defueling work platform with only the torch saddle plate assembly attached represented a load drop of 3200 pounds from a distance of 47.6 feet. The manual tool positioner with the mast and torch attached represented a potential load drop of about 4100 pounds from a distance of 23 feet. Calculations demonstrated that these load drops would not result in a nozzle weld failure (refer to Appendix A to the SER for Cases C and D).

  - *Baffle Plate Dropped inside Vessel*. The third potential load drop could occur once the baffle plates were cut, bolts were removed, and the baffle plate section was ready to be moved. If the baffle plates were to remain in the vessel, the largest baffle plate section was about 2500 pounds (including support clamps) and the drop distance was about 9.5 feet using a transfer lift of less than 2 feet. Calculations demonstrated that this load drop would not result in a nozzle weld failure (refer to Appendix A to the SER for Case B).

  - *Baffle Plate Dropped outside Vessel*. The fourth and fifth potential load drops (i.e., baffle plate section (1 of 16 sections) and transfer shield) could occur during the evolution option that removed the baffle plates from the reactor vessel for temporary disposition in one of the two modified core flood tanks. [m] The most severe load drop potential would exist with the drop of a baffle plate section from its loaded position inside the transfer shield and the transfer shield drop onto the defueling work platform. The baffle plate section (including support clamps) represented a load of about 1500 pounds dropped at a distance of 44 feet to the bottom head/in-core nozzles. Calculations demonstrated that this load drop would not result in nozzle weld failure (refer to Appendix A to the SER for Case A).

    The transfer shield would be designed so that the shield would not pass through the T-slot. The weight of the transfer shield and baffle plate section that could potentially be dropped

[m] The core flood tanks were modified to store parts from the lower core support assembly. The top of the tank was cut off and the piping valve closed and flanged.

on the platform would be about 24,000 pounds. The shield would be limited to a maximum lift height of 332.1-foot elevation, based on the previous SER [(194)] for heavy load handling over the reactor vessel, which provided load lift limitations to prevent defueling work platform collapse if a load was dropped anywhere on the platform.

The SER that was cited in a previous report, "Safety Analysis for Lower Core Support Assembly Forging Removal from the Reactor Vessel and Lifting and Handling in the Reactor Building," dated July 1988, allowed increased lift heights as long as the load path was limited to within 3.5 feet of either side of the north-south or east-west centerlines of the platform, according to the formula 34,120/(W+331.5). With a load of 24,000 pounds, the allowable lift height was at the 332.9-foot elevation. For the remainder of the load path from the vessel to the core flood tank, the restrictions committed to in the licensee's SER [(195)] for heavy load lifting inside containment building would apply.

- *Load Handling Controls*. The potential for other load drop accidents into the reactor vessel was also minimized by careful control of load handling activities and the use of load handling equipment, which had been conservatively designed and tested. Load handling activities were performed in accordance with approved procedures for such activities. A work instruction or procedure would control each specific load handling activity. Trained and qualified personnel would perform these load handling activities.

- ***Evaluation: Load Drop over the Reactor Vessel (Appendix A)***. [(196)] During UCSA defueling, a large area of the reactor vessel lower head could be exposed to direct impact from heavy loads. The licensee performed analyses to better determine the potential damage that could be incurred by the in-core nozzles because of dropped loads. Simple calculations were used to ascertain if more complex analyses were warranted.

- *Potential Drop Loads*. The following objects were considered as potential accident loads: (●) (Case A) baffle plate section (1 of 16 cut sections); (●) (Case B) baffle plate section (1 of 8 cut sections); (●) (Case C) manual tool positioner; and (●) (Case D) manual tool positioner with mast.

- *Assumptions*. To maintain a simplistic approach, the following major assumptions were applied:

  - *Kinetic Energy*. Upon impact, all kinetic energy of the falling object would be transmitted to the instrumentation nozzle and would result in strain. This assumption was conservative since some of the energy would also be converted to strain in the dropped object and the reactor vessel lower head.

  - *Compressive Stress-Strain*. The compressive stress-strain curve for a short column of Inconel was identical to the tensile stress-strain curve. This assumption was conservative since ductile metals would fail in tension before failing in compression without buckling.

- *Static Stress-Strain*. The static stress-strain curve for Inconel was appropriate for dynamic loadings. This assumption could be slightly nonconservative as some metals exhibited higher strength but lower ductility with increasing load application speeds.

- *Strain*. The strain was assumed to be uniform over the entire nozzle. This assumption did not account for the possibility of the nozzle bending and gave an upper bound on the permissible drop heights.

- Material Properties. As-constructed material properties were used for the nozzle and weld materials. However, nozzle material properties could have been degraded by elevated temperatures during the accident.

- *Drag*. When dropped through water, the objects under consideration would be subject to drag, which could vary significantly, depending on the orientation of the falling object relative to the direction of movement. An examination of the potential drag coefficients for various sharp-edged bodies indicated the coefficients could vary from 0.5 to 1.5. This indicated that the drag coefficient would have a significant effect on the calculated impact velocity for a waterdrop height of 30 feet or more. Drag coefficients did not need to be considered for the drop heights of the baffle plate sections (1/8th and 1/16th) while drag coefficients equal to 1.16 (for the rectangular saddle plate assembly) were used for objects C and D.

o *Permissible Drop Heights*. Assuming that the impact load was entirely in the axial direction and along the centerline of the nozzle, an upper bound on the permissible drop heights could be established. The analysis conservatively assumed that all the kinetic energy of the impacting object would be absorbed in the nozzle. Since the nozzle's stress-strain curve was known, the limiting impact velocity could be determined. Knowing the impact velocity allowed the determination of the drop heights by iteration.

  The following drop heights were used based on the planned evolution involving the particular object and were found to be acceptable (i.e., they would not result in in-core nozzle weld failure): (●) Case A—baffle plate section (1 of 16 cut sections) weighed 1500 pounds and had airdrop and waterdrop heights of 7.5 feet and 36.6 feet, respectively; (●) Case B—baffle plate section (1 of 8 cut sections) weighed 2500 pounds and had a waterdrop height of 9.5 feet (larger sections would not be lifted from the reactor vessel); (●) Case C—manual tool positioner weighed 3200 pounds and had airdrop and waterdrop heights of 11.0 feet and 36.6 feet, respectively; and (●) Case D—manual tool positioner with mast weighed 4100 pounds and had a waterdrop height of 23 feet (the larger tool would not be lifted from the reactor vessel).

o *Load Strike (Off-Center of Nozzle)*. All of the analyses of permissible drop heights considered that the dropped tool struck the exposed in-core nozzle on centerline. Realistically, the impacting object could strike the nozzle off-center, creating both an axial load and a bending moment. An impact load on the nozzle taper would produce a lateral load, and an additional moment would be created. The magnitudes of the lateral load and bending moment were difficult to establish. However, by using the energy approach and simple inelastic equations for the deflection of an end-loaded cantilever beam, the maximum energy absorbed could be comparable to the “axial load only” condition.

Analysis determined that the nozzle could absorb a side load of about 6 percent of an axial load. If a substantial part of the postulated impact energy was applied horizontally, the nozzle was likely to fail; however, it would be expected to fail above and parallel to the inside surface of the reactor vessel lower head. Therefore, nozzle failure due to off-center loading could fail the nozzle but not cause significant leakage since the segment of the 0.75-inch Schedule 160 Inconel pipe in the reactor vessel and its weld would likely remain intact.

- *Reactor Vessel Penetration Seal Integrity*. The greatest load transmitted to the vessel would be for an axial impact load on the in-core instrument nozzle. Since the nozzle outer diameter above the vessel wall (i.e., 2 inches) was greater than the reactor vessel penetration diameter (about 1 inch), the nozzle would have to shear through the vessel wall to punch a hole through the lower head. The ultimate axial stress capability of the nozzle was well below the ultimate strength of the vessel wall so that the nozzle would fail before the lower head was penetrated. An undamaged nozzle, therefore, could not be pushed through the vessel wall.

  Of the potential failure mechanisms, the evaluation concluded that the worst case anticipated in-core nozzle failure mechanism was shearing at the inside surface of the reactor vessel lower head. As previously noted, the 0.75-inch Schedule 160 portion of the instrument tube that penetrated the vessel wall was welded directly to the vessel wall. The 2-inch outside diameter in-core instrument nozzle was welded separately to the vessel wall and the 0.75-inch pipe. Failure of the nozzle was unlikely to fail the 0.75-inch pipe to vessel weld that provided the penetration seal. For conservatism, however, it was assumed that this weld failed as a result of the postulated load drop accident.

  Failure of the tube-to-vessel-wall weld would not result in the tubes being forced out of the lower head by the head of water in the vessel. The tubes consisted of Schedule 80 stainless-steel pipe and were supported at the floor below the vessel. The maximum clearance, considering manufacturing tolerance, between the outside diameter of the tube and the inside diameter of the bore in the vessel wall was 0.005 inch. There was insufficient flexibility in the tubes to allow them to drop the 5.5 inches required to fall free to the bottom of the vessel head.

- *Ex-Vessel Failure*. In-core tube failure outside of the vessel was not considered credible. Consequently, the only credible leakage path from the vessel following a heavy load drop was through the annulus around the tube penetrations through the vessel wall. This leakage was previously calculated to be about 0.40 gallon per minute per nozzle penetration. A previous safety evaluation for a recovery technical specification change request (No. 46) [197] demonstrated that there was a capability to provide borated water makeup to the reactor coolant system in excess of 17 gallons per minute even if offsite power were lost.

- ***NRC Review: Load Drop***. [198] Before UCSA defueling, a central portion of the elliptical flow distributor would be cut and removed from the reactor vessel. This exposed the lower head and in-core instrument penetrations to a dropped load. During UCSA defueling the potential existed for a load drop involving sections of the core baffle plates or defueling tools. The NRC had previously reviewed [199] the licensee's heavy load program and found the program acceptable. This program included: (●) definition of safe load paths; (●) development of load handling procedures; (●) periodic inspection and testing of cranes; (●) qualifications, training, and specific conduct of crane operators; (●) compliance of special lifting devices with the guidelines of American National Standards Institute (ANSI) Standard N14.6-1978; [200] (●) lifting devices that were not designed for specific applications were installed and used in accordance with the guidelines of ANSI Standard B30.9 [201]; and (●) design of cranes to ANSI Standard B30.2 [202] or Crane Manufacturers Association of America Specification 70. [203]

The licensee and the NRC also evaluated the consequences of potential load drop accidents during UCSA defueling. The NRC concluded that, in general, the consequences fell within the bounds analyzed in its SER [204] for the lower core support and lower head defueling and the licensee's SER [205] for completion of lower core support and lower head defueling. The forces generated were not large enough to force an in-core instrument penetration through the lower head or to directly breach the lower head. The potentially ablated area outward of the midline of the "P" row of fuel assemblies would remain protected by the elliptical flow distributor unless prior examination showed this area to be undamaged.

7.7 Evaluations for Waste Management (NA)

7.7.1 EPICOR II (NA)

7.7.2 Submerged Demineralizer System (NA)

7.7.2.1 Submerged Demineralizer System Operations (NA)

7.7.2.2 Submerged Demineralizer System Operation Liner Recombiner and Vacuum Outgassing System (NA)

7.8 Endnotes

Reference citations are exact filenames of documents on the Digital Versatile Discs (DVDs) to NUREG/KM-0001, Supplement 1, [206] "Three Mile Island Accident of 1979 Knowledge Management Digest: Recovery and Cleanup," issued June 2016, DVD document filenames start with a full date (YYY-MM-DD) or end with a partial date (YYYY-DD). Citations for references that are not on a DVD begin with the author organization or name(s), with the document title in quotation marks. Contents of the DVDs are available at the Idaho National Laboratory Web site (https://tmi2kml.inl.gov).

[1] USNRC, “Control of Heavy Loads,” Generic Letter 80-113, December 22, 1980 (Amended in Generic Letter 81-07, February 3, 1981) [Available at nrc.gov]
[2] USNRC, “Control of Heavy Loads at Nuclear Power Plants,” NUREG-0612, July 1980 [Available at nrc.gov]
[3] USNRC, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition,” NUREG-0800 (Formerly NUREG-75/087) [Available at nrc.gov]
[4] (1986-06-02) GPU Safety Evaluation, Heavy Load Handling Inside Containment, Rev. 3
[5] (1985-04-19) GPU Safety Evaluation, Heavy Load Handling Over Reactor Vessel SER
[6] (1983-07-20) GPU Safety Evaluation, Underhead Characterization, Core Sampling Addendum
[7] USNRC, “Three Mile Island Accident of 1979 Knowledge Management Digest: Recovery and Cleanup,” NUREG/KM-0001, Supplement 1, June 2016 [Available at nrc.gov]
[8] (1983-07-20) GPU Safety Evaluation, Underhead Characterization, Core Sampling Addendum
[9] (1983-08-19) NRC Safety Evaluation, Addendum to the Underhead Characterization Study (re 07-30-1983)
[10] (1986-06-11) GPU Safety Evaluation, Core Stratification Sample Acquisition, Rev. 4
[11] (1985-09-11) GPU Safety Evaluation, Heavy Load Handling Inside Containment, Rev. 2
[12] (1985-04-19) GPU Safety Evaluation, Heavy Load Handling Over Reactor Vessel SER
[13] (1986-05-05) NRC Safety Evaluation, Core Stratification Sample Acquisition SER, Rev. 1 (re 08-30-1985, 12-31-1985)
[14] (1989-08-18) GPU Safety Evaluation, Remove Metallurgical Samples from Reactor Vessel SER
[15] (1989-10-20) GPU Safety Evaluation, Remove Metallurgical Samples from Vessel, Rev. 1 (re 08-18-1989) (effective pages)
[16] (1988-06-27) GPU Safety Evaluation, Completion of Lower Core Support and Lower Head Defueling, Rev. 1
[17] (1986-06-02) GPU Safety Evaluation, Heavy Load Handling Inside Containment, Rev. 3
[18] (1989-11-28) NRC Safety Evaluation, Reactor Vessel Lower Head Metallurgical Sampling (re 08-18-1989)
[19] (1988-06-27) GPU Safety Evaluation, Completion of Lower Core Support and Lower Head Defueling, Rev. 1
[20] (1988-12-01) NRC Safety Evaluation, Lower Core Support and Lower Head Defueling (re 06-06-1988)
[21] (1986-03-28) GPU Safety Evaluation, Reactor Building Decontamination and Dose Reduction Activities for 1986, Rev. 0
[22] (1984-02-03) NRC Safety Evaluation, Containment Decontamination and Dose Reduction (re 09-29, 01-23-1983)
[23] (1983-02-18) GPU Safety Evaluation, Polar Crane Load Test
[24] USNRC, "Single-Failure-Proof Cranes for Nuclear Power Plants," NUREG-0554, May 1979 [ADAMS Accession No. ML110450636]
[25] (1983-02-18) GPU Safety Evaluation, Polar Crane Load Test
[26] (1983-02-18) GPU Safety Evaluation, Polar Crane Load Test
[27] (1982-06-01) BAW, Criticality Calculations to Support Recovery Activities Through Head Removal
[28] NUREG-0557, Evaluation of Long-Term Post-Accident Core Cooling of TMI-2 (1979-05)
[29] (1983-02-18) GPU Safety Evaluation, Polar Crane Load Test
[30] (1983-11-18) NRC Safety Evaluation, Reactor Building Polar Crane Load Test
[31] (1982-05-17) Order Amendment
[32] (1984-03-09) GPU Safety Evaluation, Head Removal, Rev. 5
[33] (1984-03-09) GPU Safety Evaluation, Head Removal, Rev. 5

[34] (1984-03-09) GPU Safety Evaluation, Head Removal, Rev. 5
[35] (1983-02-18) GPU Safety Evaluation, Polar Crane Load Test
[36] (1984-07-17) NRC Safety Evaluation, Reactor Pressure Vessel Head Lift (re -3-09-1984)
[37] USNRC, "Control of Heavy Loads," Generic Letter 80-113, December 22, 1980 (Amended in Generic Letter 81-07, February 3, 1981) [Available at nrc.gov]
[38] (1984-03-09) GPU Safety Evaluation, Head Removal, Rev. 5
[39] (1984-03-09) GPU Safety Evaluation, Head Removal, Rev. 5
[40] USNRC, "Control of Heavy Loads at Nuclear Power Plants," NUREG-0612, July 1980 [Available at nrc.gov]
[41] (1986-06-02) GPU Safety Evaluation, Heavy Load Handling Inside Containment, Rev. 3
[42] (1985-04-18) GPU, Polar Crane Aux Hoist Load Test Results
[43] (1985-04-18) GPU, Polar Crane Aux Hoist Load Test Results
[44] (1986-05-29) GPU Technical Evaluation, Fuel Canister Storage Racks, Rev. 1
[45] (1984-03-09) GPU Safety Evaluation, Head Removal, Rev. 5
[46] (1983-02-18) GPU Safety Evaluation, Polar Crane Load Test
[47] (1984-03-09) GPU Safety Evaluation, Head Removal, Rev. 5
[48] (1983-02-18) GPU Safety Evaluation, Polar Crane Load Test
[49] (1986-05-29) GPU Technical Evaluation, Fuel Canister Storage Racks, Rev. 1
[50] (1986-06-02) GPU Safety Evaluation, Heavy Load Handling Inside Containment, Rev. 3
[51] (1986-05-15) GPU Safety Evaluation, Defueling SER, Rev. 10
[52] (1985-04-18) GPU, Polar Crane Aux Hoist Load Test Results
[53] (1984-07-17) NRC Safety Evaluation, Reactor Pressure Vessel Head Lift (re 3-09-1984)
[54] (1983-02-18) GPU Safety Evaluation, Polar Crane Load Test
[55] (1983-03-15) GPU Response to NRC, Polar Crane SER
[56] (1985-11-15) NRC Safety Evaluation, Heavy Load Handling Inside Containment (re 09-11-1985)
[57] (1987-07-23) NRC Review, Heavy Load Handling Inside Containment (re various letters)
[58] (1985-10-30) NRC (Internal), Criticality Evaluation of Loaded Canister Dropping its Load in Another Canister.pdf
[59] (1985-05-02) NRC Safety Evaluation, Heavy Load Handling Over the TMI-2 Reactor Vessel (re 04-19-1985)
[60] (1985-11-05) NRC Safety Evaluation, Defueling Canisters (re various letters)
[61] (1985-05-02) NRC Safety Evaluation, Heavy Load Handling Over the TMI-2 Reactor Vessel (re 04-19-1985)
[62] (1986-06-02) GPU Safety Evaluation, Heavy Load Handling Inside Containment, Rev. 3
[63] (1987-07-23) NRC Review, Heavy Load Handling Inside Containment (re various letters)
[64] (1985-05-02) NRC Safety Evaluation, Heavy Load Handling Over the TMI-2 Reactor Vessel (re 04-19-1985)
[65] (1986-06-02) GPU Safety Evaluation, Heavy Load Handling Inside Containment, Rev. 3
[66] (1987-07-23) NRC Review, Heavy Load Handling Inside Containment (re various letters)
[67] (1985-05-02) NRC Safety Evaluation, Heavy Load Handling Over the TMI-2 Reactor Vessel (re 04-19-1985)
[68] (1985-04-19) GPU Safety Evaluation, Heavy Load Handling Over Reactor Vessel SER
[69] (1985-04-19) GPU Safety Evaluation, Heavy Load Handling Over Reactor Vessel SER
[70] (1985-04-19) GPU Safety Evaluation, Heavy Load Handling Over Reactor Vessel SER
[71] (1984-03-09) GPU Safety Evaluation, Head Removal, Rev. 5
[72] (1985-01-03) GPU Safety Analysis, Measuring Uranium Fuel Material in Lower Reactor Vessel Region, Rev. 0
[73] (1985-04-19) GPU Safety Evaluation, Heavy Load Handling Over Reactor Vessel SER

[74] (1985-08-08) Order Amendment (also exemption from 10CFR50 AppA)
[75] (1985-04-19) GPU Safety Evaluation, Heavy Load Handling Over Reactor Vessel SER
[76] (1984-11-08) GPU, Criticality Report, Reactor Coolant System
[77] (1985-05-02) NRC Safety Evaluation, Heavy Load Handling Over the TMI-2 Reactor Vessel (re 04-19-1985)
[78] (1985-04-19) GPU Safety Evaluation, Heavy Load Handling Over Reactor Vessel SER
[79] (1984-03-09) GPU Safety Evaluation, Head Removal, Rev. 5
[80] (1984-07-17) NRC Safety Evaluation, Reactor Pressure Vessel Head Lift (re -3-09-1984)
[81] (1985-04-23) Order Amendment (see ROP-27)
[82] (1984-06-18) GPU Safety Evaluation, Preparatory Activities for Plenum Assembly Removal
[83] (1984-03-09) GPU Safety Evaluation, Head Removal, Rev. 5
[84] (1984-03-09) GPU Safety Evaluation, Head Removal, Rev. 5
[85] (1984-07-31) NRC Review, Preparatory Activities for Plenum Assembly Removal SER (re 06-18-1984)
[86] (1984-09-14) NRC Safety Evaluation, Plenum Assembly Removal (re various letters)
[87] (1982-07-13) NRC Review, Control Rod Drive Mechanism Quick Look Camera Inspection (re 07-06-1982)
[88] (1983-07-13) NRC Review, Reactor Vessel Underhead Characterization (re Various Letters)
[89] (1984-07-17) NRC Safety Evaluation, Reactor Pressure Vessel Head Lift (re -3-09-1984)
[90] (1984-01-05) NRC Review, Control of Heavy Loads
[91] (1984-07-17) NRC Safety Evaluation, Reactor Pressure Vessel Head Lift (re -3-09-1984)
[92] (1985-05-01) GPU Safety Evaluation, Plenum Removal and Transfer, Rev. 3
[93] (1984-03-09) GPU Safety Evaluation, Head Removal, Rev. 5
[94] (1985-05-01) GPU Safety Evaluation, Plenum Removal and Transfer, Rev. 3
[95] (1985-04-19) GPU Safety Evaluation, Heavy Load Handling Over Reactor Vessel SER
[96] (1985-05-01) GPU Safety Evaluation, Plenum Removal and Transfer, Rev. 3
[97] (1985-05-01) GPU Safety Evaluation, Plenum Removal and Transfer, Rev. 3
[98] (1985-05-07) NRC Safety Evaluation, Lifting, Transfer, and Storage of Plenum Assembly (re various letters)
[99] (1985-04-19) GPU Safety Evaluation, Heavy Load Handling Over Reactor Vessel SER
[100] (1985-05-02) NRC Safety Evaluation, Heavy Load Handling Over the TMI-2 Reactor Vessel (re 04-19-1985)
[101] (1984-03-09) GPU Safety Evaluation, Head Removal, Rev. 5
[102] (1984-03-09) GPU Safety Evaluation, Head Removal, Rev. 5
[103] (1984-03-09) GPU Safety Evaluation, Head Removal, Rev. 5
[104] (1989-05-12) GPU Technical Evaluation, Defueling Water Cleanup System, Rev. 12
[105] GPU Nuclear Corporation, “Control of Heavy Loads,” GPU Letter LL2-81-0277 to NRC, September 30, 1981 [ADAMS Accession No. 8110060335]
[106] (1986-06-02) GPU Safety Evaluation, Heavy Load Handling Inside Containment, Rev. 3
[107] (1985-08-16) GPU Technical Evaluation, SDS, Rev. 3
[108] (1986-05-15) GPU Safety Evaluation, Defueling SER, Rev. 10
[109] (1985-08-06) NRC Safety Evaluation, Defueling Water Cleanup System (re various letters)
[110] (1984-10-17) NRC Review, Fuel Pool-A Refurbishment SER (re various letters)
[111] (1984-11-01) GPU Safety Evaluation, Reactor Building Heavy Loads, Rev. 0
[112] (1987-09-22) GPU Technical Evaluation, Defueling Canister TER, Rev. 4
[113] (1987-06-10) GPU, Filter Canister Media Modification
[114] (1985-11-05) NRC Safety Evaluation, Defueling Canisters (re various letters)

[115] American Society of Mechanical Engineers, *Boiler and Pressure Vessel Code*, 1983 Edition, Section VIII, "Rules for Construction of Pressure Vessels," Division 1, Part UW, "Requirements for Pressure Vessels Fabricated by Welding," New York, NY.

[116] *U.S. Code of Federal Regulations*, "Domestic Licensing of Production and Utilization Facilities," Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," Chapter I, Title 10, "Energy"

[117] American National Standards Institute, N14.6, "Radioactive Materials—Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 kg) or More," New York, NY [Revision not provided]

[118] *U.S. Code of Federal Regulations*, "Packaging and Transportation of Radioactive Material," Part 71, Chapter I, Title 10, "Energy"

[119] (1986-04-11) GPU Safety Evaluation, Replacement of Loaded Fuel Canister Head Gaskets

[120] (1985-09-11) GPU Safety Evaluation, Heavy Load Handling Inside Containment, Rev. 2

[121] (1986-04-29) NRC Review, Replacement of Loaded Fuel Canister Head Gaskets (re 04-11-1986)

[122] (1985-11-05) NRC Safety Evaluation, Defueling Canisters (re various letters)

[123] (1985-05-03) NRC Review, Fuel Canister Storage Racks TER (re 02-27-1985)

[124] (1986-09-16) NRC Review, Fuel Canister Storage Racks TER (re various letters)

[125] (1986-01-17) GPU, Plan to Use Modified Debris Canisters for removing End Fittings, etc

[126] (1986-01-21) NRC Review, Plan to Use Modified Fuel Canisters to Remove End Fittings, etc (re 01-17-1986)

[127] (1986-05-29) GPU Technical Evaluation, Fuel Canister Storage Racks, Rev. 1

[128] (1985-09-11) GPU Safety Evaluation, Heavy Load Handling Inside Containment, Rev. 2

[129] (1985-05-03) NRC Review, Fuel Canister Storage Racks TER (re 02-27-1985)

[130] (1986-09-16) NRC Review, Fuel Canister Storage Racks TER (re various letters)

[131] (1983-09-09) NRC Review, Plan to Restore SFP-A for Fuel Removal Activities (re 07 29 1983, 08-30-1983)

[132] (1984-12-18) NRC Review, Plans to Expand Reactor Building Heavy Load Travel Boundaries (re 11-01-1984)

[133] (1986-09-16) NRC Review, Fuel Canister Storage Racks TER (re various letters)

[134] (1987-04-02) GPU, Safety Evaluation, Canister Handling and Preparation for Shipment SER, Rev. 4 (Effective Pages Only)

[135] (1986-05-15) GPU Safety Evaluation, Defueling SER, Rev. 10

[136] American National Standards Institute, B30.2-1976, "Overhead and Gantry Cranes," New York, NY

[137] American National Standards Institute, N14.6, "Radioactive Materials—Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 kg) or More," New York, NY [Revision not provided]

[138] American National Standards Institute, B30.9-1971, "Slings," New York, NY

[139] American National Standards Institute, B30.11-1980, "Monorail Systems and Underslung Cranes," New York, NY

[140] American National Standards Institute, B30.16-1981, "Overhead Hoists," New York, NY

[141] (1986-05-15) GPU Safety Evaluation, Defueling SER, Rev. 10

[142] Crane Manufacturers Association of America, Specification 70-1975, "Specification for Top Running Bridge and Gantry Type Multiple Girder Electric Overhead Traveling Cranes," Charlotte, NC

[143] Metropolitan Edison Company, et al., "Final Safety Analysis Report, Three Mile Island Nuclear Station, Unit 1," 1978 [Revision not provided]

[144] Crane Manufacturers Association of America, Specification 74-1974, "Specification for Top Running and Under Running Single Girder Electric Overhead Cranes Utilizing Under Running Trolley Hoist," Charlotte, NC
[145] (1987-04-02) GPU, Safety Evaluation, Canister Handling and Preparation for Shipment SER, Rev. 4 (Effective Pages Only)
[146] (1987-04-02) GPU, Safety Evaluation, Canister Handling and Preparation for Shipment SER, Rev. 4 (Effective Pages Only)
[147] (1986-06-20) NRC Safety Evaluation, Canister Handling and Preparation for Shipment Program (re 02-17, 06-11-1986)
[148] (1985-09-10) GPU Technical Evaluation, Defueling Canister TER (re 08-15-1985)
[149] (1985-11-05) NRC Safety Evaluation, Defueling Canisters (re various letters)
[150] (1986-04-11) NRC, Certificate of Compliance 9200, Rev. 0, for Model 125-B Shipping Cask
[151] (1988-03-03) NRC Review, Canister Handling and Preparation for Shipment Program TER (re various letters)
[152] (1987-04-02) GPU, Safety Evaluation, Canister Handling and Preparation for Shipment SER, Rev. 4 (Effective Pages Only)
[153] (1987-05-26) GPU Safety Evaluation, Canister Handling and Preparation for Shipment, Rev. 5 (Effective Pages Only)
[154] (1986-03-26) GPU, Plan to Use Hydraulic Shredder to Reduce Core Debris
[155] (1985-09-11) GPU Safety Evaluation, Heavy Load Handling Inside Containment, Rev. 2
[156] (1985-04-19) GPU Safety Evaluation, Heavy Load Handling Over Reactor Vessel SER
[157] (1986-04-16) NRC Review, Use of Hydraulic Shredder
[158] (1986-03-14) GPU Safety Evaluation, Decontamination Using Ultrahigh Pressure Water Flush, Rev. 0
[159] (1985-09-11) GPU Safety Evaluation, Heavy Load Handling Inside Containment, Rev. 2
[160] (1986-07-16) NRC Safety Evaluation, Decontamination using Ultrahigh Pressure Water Flush (re 03-14-1986)
[161] (1985-10-24) GPU, Commencement of Preliminary Defueling Operations
[162] (1985-09-11) GPU Safety Evaluation, Heavy Load Handling Inside Containment, Rev. 2
[163] (1985-10-29) NRC Safety Evaluation, Commencement of TMI-2 Preliminary Defueling Operations (re 10-24-1985)
[164] (1985-05-02) NRC Safety Evaluation, Heavy Load Handling Over the TMI-2 Reactor Vessel (re 04-19-1985)
[165] (1985-08-08) Order Amendment (also exemption from 10CFR50 AppA)
[166] (1985-10-10) GPU Safety Evaluation, Early Defueling of the TMI-2 Reactor Vessel, Rev. 4
[167] (1986-05-15) GPU Safety Evaluation, Defueling SER, Rev. 10
[168] (1986-05-20) GPU Safety Evaluation, Defueling SER, Rev. 10 (Correction)
[169] (1985-11-12) NRC Safety Evaluation, Early Defueling of the TMI-2 Reactor Vessel (10-10-1985)
[170] (1985-04-19) GPU Safety Evaluation, Heavy Load Handling Over Reactor Vessel SER
[171] (1985-09-11) GPU Safety Evaluation, Heavy Load Handling Inside Containment, Rev. 2
[172] (1985-10-29) NRC Safety Evaluation, Commencement of TMI-2 Preliminary Defueling Operations (re 10-24-1985)
[173] (1986-08-16) GPU, Storage of Upper End Fittings in Shielded 55 Gal Drums
[174] (1986-05-15) GPU Safety Evaluation, Defueling SER, Rev. 10
[175] (1986-05-15) GPU Safety Evaluation, Defueling SER, Rev. 10
[176] (1986-05-20) GPU Safety Evaluation, Defueling SER, Rev. 10 (Correction)
[177] (1986-09-10) NRC Review, Plans to Store Upper End Fittings (re various letters)

[178] (1986-09-25) NRC Safety Evaluation, Storage of Upper End Fittings in an Array of 55 Gallon Drums (re various letters)
[179] (1985-10-10) GPU Safety Evaluation, Early Defueling of the TMI-2 Reactor Vessel, Rev. 4
[180] (1985-09-11) GPU Safety Evaluation, Heavy Load Handling Inside Containment, Rev. 2
[181] (1985-04-19) GPU Safety Evaluation, Heavy Load Handling Over Reactor Vessel SER
[182] (1986-03-26) GPU, Plan to Use Hydraulic Shredder to Reduce Core Debris
[183] (1986-07-24) NRC Review, Core Region Bulk Defueling Operations Limited to the Core Region (re 05-15, 07-23-1986)
[184] (1985-11-12) NRC Safety Evaluation, Early Defueling of the TMI-2 Reactor Vessel (10-10-1985)
[185] (1988-01-18) GPU Safety Evaluation, Lower Core Support Assembly Defueling, Rev. 2
[186] (1987-12-28) GPU, Use of Core Bore Machine for Dismantling the Lower Core Support Assembly
[187] (1988-01-08) NRC Review, Use of Core Bore Machine for Dismantling Lower Core Support Assembly (re 12-03, 12-28-1987)
[188] (1988-04-01) NRC Safety Evaluation, Lower Core Support Assembly Defueling (re 01-18-1988)
[189] (1988-06-06) GPU Safety Evaluation, Completion of Lower Core Support and Lower Head Defueling, Rev. 0
[190] (1988-06-06) GPU Safety Evaluation, Completion of Lower Core Support and Lower Head Defueling, Rev. 0
[191] (1985-08-08) Order Amendment (also exemption from 10CFR50 AppA)
[192] (1988-12-01) NRC Safety Evaluation, Lower Core Support and Lower Head Defueling (re 06 06-1988)
[193] (1988-09-15) GPU Safety Evaluation, Completion of Upper Support Core Assembly Defueling, Rev. 0
[194] (1985-04-26) GPU Response to NRC, Plenum Removal Safety Evaluation Report (attached) (re 04-05-1985)
[195] (1986-06-02) GPU Safety Evaluation, Heavy Load Handling Inside Containment, Rev. 3
[196] (1988-09-15) GPU Safety Evaluation, Completion of Upper Support Core Assembly Defueling, Rev. 0
[197] (1985-08-08) Order Amendment (also exemption from 10CFR50 AppA)
[198] (1989-04-04) NRC Safety Evaluation, Completion of Upper Core Support Assembly Defueling (re 9-15-1988)
[199] (1988-12-01) NRC Safety Evaluation, Lower Core Support and Lower Head Defueling (re 06-06-1988)
[200] American National Standards Institute, B30.1–1971, “Radioactivity Materials—Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4,500 Kilograms) or More,” ANSI Standard N14.6, 1978, New York, NY
[201] American National Standards Institute, B30.9–1971, “Slings,” New York, NY
[202] American National Standards Institute, B30.2–1976, “Overhead and Gantry Cranes,” New York, NY
[203] Crane Manufacturers Association of America, Specification 70, “Specifications for Top Running Bridge and Gantry Type Multiple Girder Electric Overhead Traveling Cranes,” Charlotte, NC [Year of issue not available]
[204] (1988-12-01) NRC Safety Evaluation, Lower Core Support and Lower Head Defueling (re 06-06-1988)
[205] (1988-01-18) GPU Safety Evaluation, Lower Core Support Assembly Defueling, Rev. 2
[206] USNRC, “Three Mile Island Accident of 1979 Knowledge Management Digest: Recovery and Cleanup,” NUREG/KM-0001, Supplement 1, June 2016 [Available at nrc.gov]

8 REACTOR VESSEL INTEGRITY SAFETY EVALUATIONS

TABLE OF CONTENTS

Note: “NA” (not applicable) means the licensee and the NRC determined that the safety topic was not important for the activity.

8.1 Introduction

8.1.1 Background

The safety topic of reactor vessel and reactor coolant system integrity covers cleanup activities associated with the preparations for defueling operations, defueling, and post-defueling examination of the reactor vessel lower head. This chapter focuses on these activities, as well as cleanup activities that could impact reactor coolant system pressure boundary integrity (an early concern) and decontamination using abrasive equipment.

The purpose of the evaluations was to ascertain that: (●) existing structures could withstand the defueling work platform loads and operating forces; (●) the reactor vessel could withstand the operating forces; and (●) accidents in operation would not damage the structural integrity of the system. To ensure the structural integrity of the system, all applied loads, static and dynamic, would be accounted for; load paths and load distributions would be well understood; critical load combinations would be studied; and structural assemblies would be adequately designed.

The safety evaluations considered possible reactor vessel failure mechanisms that included: (●) load drops over and inside the reactor vessel; (●) core boring/bulk defueling operations affecting the integrity of in-core nozzle welds and instrument tubes (inside and outside the vessel); (●) cutting/burning operations causing damage to the vessel wall; (●) incorrect installation of the in-core instrumentation nozzle seal plugs before cutting; (●) cleaning operations puncturing the in-core tube; (●) corrosion; and (●) defueling operations with a lower reactor vessel head potentially damaged in an accident. The safety evaluations also included activities that could potentially impact the integrity of the reactor coolant system pressure boundary, an early concern of the cleanup campaign, and the integrity of structures, systems, and components from decontamination equipment. Evaluations of mitigation actions included leak detection (level and radiation instrumentations) and capability to make up water to the reactor coolant system using gravity feed or pumping systems.

For the activities with the potential for load drop accidents over and inside the reactor vessel, the safety evaluations provided a detailed analysis that examined the potential consequences of load drops in accordance with guidance in NUREG-0612, [(1)] "Control of Heavy Loads at Nuclear Power Plants." Based on calculations of damage following an accidental drop of a postulated heavy load, Criterion III of the evaluation criteria in NUREG-0612 stated that damage to the reactor vessel or the spent fuel pool was limited so as not to result in water leakage that could uncover the fuel (if the water being lost was borated, makeup water provided to overcome leakage would be from a borated source of adequate concentration). The safety evaluation of a heavy load drop in the vicinity of the reactor core included the identification of loads, targets, and the load/target interactions, as well as the evaluation of the worst credible consequence.

This chapter summarizes the licensee's and the NRC's safety evaluations associated with the integrity of the reactor vessel, reactor coolant system, and other structures.

8.1.2 Chapter Contents

This chapter presents reactor vessel integrity safety evaluations of the postaccident TMI-2 and defueling operations. It describes the many studies performed to give the reader an understanding of the thinking of the analysts at the time, the expectations and the reality, the uncertainties in the data, and the measurement and mitigation methods. It also presents a high-level timeline of cleanup activities.

The evaluations presented in this chapter ensured that all activities that could result in the loss of reactor coolant by impacts caused by load drops onto or near the reactor vessel or by damage to one or more reactor vessel lower head instrumentation nozzles were addressed and consequences evaluated; controls were maintained in accordance with the requirements of the plant's license, technical specifications, procedures, and applicable regulatory requirements; and adequate contingencies were developed for normal and accident conditions.

Key activities of concern for the reactor vessel integrity included: (●) core (bore) stratification sample acquisition; (●) use of the metal disintegration machining system to cut reactor vessel lower head wall samples; (●) the polar crane load test; (●) reactor vessel stud detensioning (pressure boundary); (●) reactor vessel head removal; (●) plenum assembly removal; (●) movement of other heavy loads in the containment building and over the reactor vessel (load drop); (●) use of coagulants to improve the performance of the defueling water cleanup system (corrosion); (●) use of defueling tools such as heavy-duty defueling tools, debris vacuum system, core bore machine for defueling, and ultrahigh pressure water flush for decontamination; (●) defueling activities for the core region, lower core support assembly, lower reactor vessel head, and upper core support assembly; and (●) others.

Additional evaluations of potential impacts on the reactor vessel integrity can be found in NUREG/KM Chapter 7 on load drop evaluations.

The following sections present the safety evaluation for each applicable cleanup activity or system. Section 8 lists endnotes showing references cited throughout this chapter.

8.2 Key Studies

8.2.1 Control of Heavy Loads at Nuclear Power Plants
(USNRC, NUREG-0612, July 1980)

This report [a, 2] summarized work performed by the NRC in the resolution of Generic Technical Activity A-36, "Control of Heavy Loads Near Spent Fuel." This technical activity was one of the generic technical subjects designated as an "unresolved safety issue" pursuant to Section 210

[a] Editor's Note: In October 2005, the NRC issued Regulatory Issue Summary (RIS) 2005-25, "Clarification of NRC Guidelines for Control of Heavy Loads," as a result of recommendations developed through Generic Issue 186, "Potential Risk and Consequence of Heavy Load Drops in Nuclear Power Plants," and findings from the NRC inspection program. This RIS reemphasized the need to follow NUREG-0612 guidance, which addressed good practices for crane operations and load movements. Attachment 1 to this RIS described the application of insights gained from operating experience and inspection to the guidelines of NUREG-0612. The attachment also clarified the guidelines where operating experience or inspection results indicated further explanation was necessary.

of the Energy Reorganization Act of 1974. [b] The report described the technical studies and evaluations performed by the NRC, the agency's guidelines based on these studies, and the agency's plans for implementing its technical guidelines. Section 2 of the report discussed the potential for an accidental load drop that could impact nuclear fuel or safety-related equipment with the possibility for excessive offsite releases, inadvertent criticality, loss of water inventory in the reactor (or spent fuel pool), or loss of safe-shutdown equipment.

Section 5 provided guidance for the control of heavy loads and this section provides various alternative approaches that offer acceptable measures for the control of heavy loads. As stated in NUREG-0612, the objectives of these guidelines were to ensure that either (1) the potential of a load drop was extremely small, or (2) for each area addressed, all of the following evaluation criteria were satisfied:

Criterion I: Based on calculations involving an accidental drop of a postulated heavy load, releases of radioactive material that could result from damage to spent fuel produced doses that were well within 10 CFR Part 100, "Reactor Site Criteria," [3] limits of 300 rem thyroid and 25 rem whole body (analyses should show that doses were equal to or less than a quarter of 10 CFR Part 100 limits).

Criterion II: Based on calculations involving an accidental drop of a postulated heavy load, damage to fuel and fuel storage racks did not result in a configuration of the fuel such that effective neutron multiplication (k_{eff}) was larger than 0.95.

Criterion III: Damage to the reactor vessel or the spent fuel pool based on calculations of damage following an accidental drop of a postulated heavy load was limited so as not to result in water leakage that could uncover the fuel (makeup water provided to overcome leakage should be from a borated source of adequate concentration if the water being lost was borated).

Criterion IV: Damage to equipment in redundant or dual safe-shutdown paths, based on calculations assuming the accidental drop of a postulated heavy load, would be limited so as not to result in the loss of required safe-shutdown functions.

The licensee's safety evaluations typically provided a detailed analysis that examined the potential consequences of load drops in accordance with guidance in NUREG-0612, as applicable. The safety evaluation of a heavy load drop in the vicinity of the reactor core included the identification of loads, targets, and load/target interactions, as well as the evaluation of the worst credible consequence. The evaluations showed that Criterion III was met.

Editor's Note: Refer to Section 2 of NUREG/KM Chapter 7 on load drop evaluations for a detailed summary of NUREG-0612. The complete evaluation for each activity, as applicable, is

[b] Editor's Note: Generic Technical Activity A-36 is described in NUREG-0933, "Resolution of Generic Safety Issues" (available at nrc.gov).

provided in Chapter 7. Only the portion that applied to reactor vessel and reactor coolant system integrity is provided below in this chapter.

8.3 Data Collection Activities

8.3.1 Axial Power Shaping Rod Insertion Test (NA)

8.3.2 Camera Insertion through Reactor Vessel Leadscrew Opening (NA)

8.3.3 Reactor Vessel Underhead Characterization (Radiation Levels) (NA)

8.3.4 Reactor Vessel Underhead Characterization (Core Sampling) (NA)

8.3.5 Core Stratification Sample Acquisition

- ***Purpose***. To perform the activities associated with: (●) installation, operation, and removal of the core boring machine; (●) acquisition of the core samples; (●) transfer of the samples from the machine to the defueling canisters; and (●) viewing of the lower vessel region through the bored holes.

- ***Evaluation: Reactor Vessel Integrity***. [(4)] The licensee's safety evaluation considered downward force limits imposed on the core bore operations and possible damage to the in-core instrumentation tubes.

    - *Downward Force Limit*. The core bore operation exerted a downward force on the core region debris bed and on the core support assembly. This downward force was automatically controlled by the drill unit instrumentation and control system [(c)] at a predetermined maximum setpoint. This downward force would not be imparted to the lower reactor vessel head in-core instrument nozzles unless there was a direct, solid link between the drill bit and a nozzle. Since none of the drill locations would be directly over an in-core nozzle, this link could be created only by the drill bit contacting debris below the flow distributor. To ensure that in-core nozzle loading was precluded, the depth of core bore would be limited to the top of the flow distributor unless a video camera view of the lower head region immediately below the drill showed a clear path. Consequently, downward force of the drill bit could be exerted only in the rubble above the flow distribution plate where the force would be distributed to the core support assembly and was unlikely to impart a load on the nozzles.

    - *Damage to In-core Instrument Tubes*. The potential for core boring activities to damage in-core instrument tubes outside of the reactor vessel was also evaluated. If the drill string/bit were capable of "catching" an in-core instrument string (cabling) and wrapping the string around the drill bit as it rotated, a stress could be imparted to an instrument tube below the

---

[c] The drill unit is instrumented with a control system, which was capable of monitoring and controlling the drilling process. Instrumentation provided visual indication of rotational speed and torque on the drill string and the weight (force) applied to the bit. The drill unit was equipped with a data acquisition system to record information on the material being drilled, such as rubble, solidified mass, standing fuel arrays, and voids.

vessel lower head. This type of event was not considered credible because no drill bit/string configuration could grab and hold an instrument string and also because each core bore would be centered over a fuel assembly with no instrument string. If an adjacent instrumented fuel assembly collapsed into the path of a core bore, the bit would drill through the assembly and sever the string. The only other drill bit contact with an instrument string would involve a loose string from an adjacent fuel assembly location. The instrument strings in an intact core were contained within an instrument tube in the center of a fuel assembly. The licensee did not consider it feasible for the surrounding fuel assembly and instrument tube to have disintegrated or melted during the accident, exposing the instrument string without the instrument string also being destroyed.

- ***NRC Review: Reactor Vessel Integrity***. [5] The NRC's safety evaluation ascertained that: (●) the existing structure could take the platform loads and operating forces; (●) the reactor vessel could take the operating forces, and (●) accidents in operation would not damage the structural integrity of the system. To maintain the structural integrity of the system: (●) all applied loads, static and dynamic, would be accounted for; (●) load paths and load distributions were well understood; (●) critical load combinations would be studied; and (●) structural assemblies would be adequately designed.

  - *Platform Loads*. The structural assembly of the drill indexing platform comprised three major subassemblies identified as the wing assembly, upper level assembly, and lower level assembly. The total load was 22,400 pounds, and 12 load combinations were considered. The structure was designed to meet American Institute of Steel Construction specifications. The defueling work platform consisted of circular and cross beams made of 304 stainless steel. They were built to within the design limits specified in the American Society of Mechanical Engineers [6] *Boiler and Pressure Vessel Code*, Section III, Division 1, Appendix XVII, "Design of Linear Type Supports by Linear Elastic and Plastic Analysis."

    The licensee performed structural analyses to evaluate the structural integrity of the defueling work platform and the shielded support structure. Five loading cases, including two cases for core drilling, were considered. The total weight, which included 100 pounds per square foot of dynamic load and the static platform shielding load, was 81,101 pounds. Maximum stresses of all components of the work platform were well within allowable limits.

  - *Reactor Vessel Loads*. The core drilling equipment was supported by the defueling work platform, which was in turn supported by the floor of the fuel transfer canal. The equipment loads associated with the drilling operation were therefore not imparted to the reactor vessel. The only significant operating force that would be imparted to the vessel was the downward force exerted by the drill bit face, and the drilling equipment was designed such that the maximum force would not exceed 10,000 pounds. The force on the drill bit acting at an inclined angle to the vessel was anticipated to be insignificant at 1118 pounds. These forces transmitted to the vessel would not damage the vessel.

- *Accident Analysis*. The licensee's safety evaluation report addressed damage that could be caused by accidents resulting from the core drilling operation. No damage to the structural integrity of the system was expected.

- *Conclusion*. Based on the above-described analyses, discussions, and findings, the NRC concluded that there was reasonable assurance that the TMI-2 core stratification sampling operation would not impair the structural integrity of the defueling work platform, support structure, and reactor vessel. The NRC found that the proposed activities were acceptable with respect to mechanical forces and reactor coolant system integrity.

8.3.6 Use of Metal Disintegration Machining System to Cut Reactor Vessel Lower Head Wall Samples

- ***Purpose***. To remove metallurgical samples from the inner surface of the lower reactor vessel head using the metal disintegration machining (MDM) system. An international research program used these samples to study core melt and reactor vessel interactions. This program was conducted after the reactor vessel was defueled.

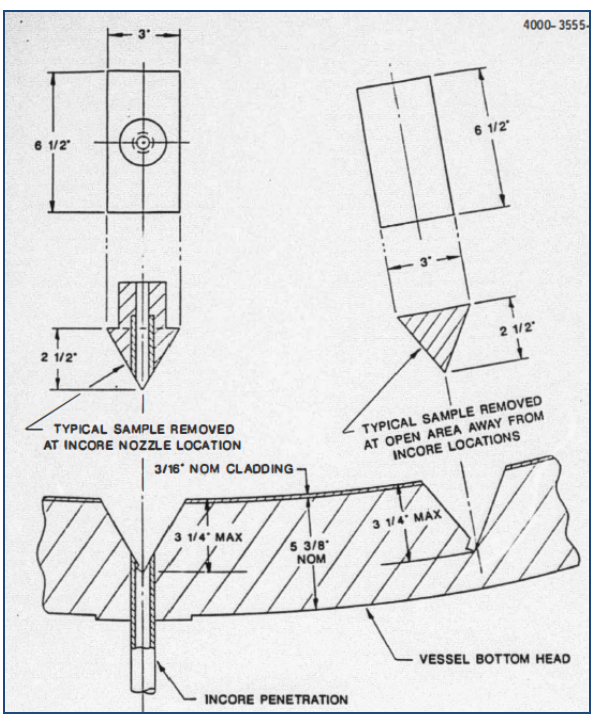
- ***Evaluation***: ***Reactor Vessel Integrity***. [7, 8] After removal of the samples, a local minimum wall thickness of at least 2 inches would remain in the lower head. As stated in the licensee's safety evaluation, this thickness was sufficient to withstand an internal pressure that was significantly higher than the water head (about 20 pounds per square inch gauge pressure imposed on the vessel in its present configuration; the new pressure rating would be very nearly the design pressure rating (2500 pounds per square inch gauge) because a relatively small percentage of the total pressure boundary area (less than 1 percent) would be sampled).

Five scenarios were hypothesized during wall sample removal activities that would affect the reactor vessel integrity. These scenarios were considered to have an extremely small possibility of occurrence since procedural restrictions and hardware designs were intended to prevent them. These scenarios are discussed below:

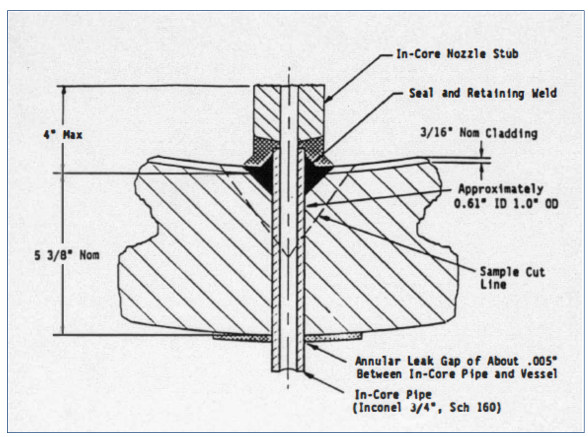
- *Abrasive Wheel Operation*. The abrasive wheel equipment was used to cut off the in-core nozzles. The nozzles would be about 2 to 4 inches above the surface (or possibly closer for special cases). During this cutting operation, the evaluation postulated that the abrasive wheel might inadvertently be in contact with the vessel wall because of a temporary loss of control, resulting in damage to the vessel. Administrative controls were used to ensure the abrasive wheel was not operating unless in position at an in-core nozzle. In addition, the abrasive cutting was a relatively slow process (taking about 1 hour to cut a 2-inch diameter nozzle) and was continuously monitored with video cameras. Any problems with the abrasive cutting operation would be detected before vessel damage could occur.

- *In-core Seal Plug Operations*. An expansion seal plug was used to seal a 0.005 to 0.010-inch annular leak path around the outside diameter of the in-core pipe. This expansion plug was also used to retain the pipe after the retaining weld was removed. The evaluation postulated that this seal plug could be installed incorrectly or in such a manner that the seal was inadequate. The leak path that developed as a result of this condition was evaluated in the licensee's safety evaluation report [9] for the extended core stratification sample acquisition activity (core bore) and was calculated to be no more than about 0.4 gallon per minute for each nozzle. This was well within the water level monitoring and coolant makeup capabilities of the plant. As a worst possible case, the safety analyses showed that makeup capability would exceed the flow of 125 gallons per minute postulated for an opening caused by complete ejection of an in-core tube, even though no clear mechanism existed to cause such a catastrophic failure. A backup plug would be installed to limit leakage in this event. The plug was a simple wedge that would be inserted into the bore hole in the vessel. However, the safety evaluation did not take credit for the plug to limit leakage. Existing leakage monitoring and makeup operations continued in accordance with existing procedures during sampling activities.

- *Cleaning Operations*. In the local areas where samples of the reactor were removed, cleaning operations, such as grinding or wire brushing, were performed to remove debris from the vessel surface. Cleaning was needed to ensure an electrically conductive surface for the MDM process; however, penetration of the vessel wall by these cleaning steps was not credible. In addition, when an in-core instrument penetration was cleaned to allow the installation of the expanding seal plug, the evaluation postulated that the cleaning operation could affect vessel integrity by puncturing the in-core tube. If the in-core tube were punctured, the resulting leakage would be less than, and bounded by, ejection of an in-core tube.



SER Figure 3. Samples being removed from bottom of reactor vessel using the MDM cutting device.



SER Figure 4. Typical in-core nozzle with seal and retaining weld (sample cut line shown).

- *Penetration of the Vessel Wall during MDM Operations*. An electrical discharge type of machining operation called “metal disintegration machining” (MDM) was used to cut samples from the reactor vessel. The evaluation postulated that this cutting technique could accidentally penetrate through the vessel wall; however, this event was not considered credible. The MDM tools were designed such that the cutters were incapable of reaching through the full vessel wall thickness. Even if adjacent cuts were made, the area covered by the foot of the MDM cutting head would preclude cutting deeper than a single sample depth (refer to Figures 3 and 4 of the safety evaluation report). In addition, the cutting process was very slow. The cutting process to remove a single sample was expected to take between 4 and 10 hours. During this time, the cutting operation would be monitored continuously. Any problems with the MDM operations would be spotted and corrected before through vessel damage could occur.

- *Corrosion*. Postsampling corrosion was not considered to be a concern as corrosion was a slow, self-limiting process. In addition, the licensee planned to drain the reactor vessel following refueling operations.

- ***NRC Review: Reactor Vessel Integrity***. [(10)] The licensee and the NRC evaluated a wide range of activities that included load drops that could cause leakage to in-core instrument penetrations. These evaluations [(11, 12)] were associated with reactor vessel lower head defueling and previous defueling activities. Leakage, through the annular gap between an instrument tube and the reactor vessel wall, could produce a leak of 0.4 gallon per minute for each penetration. This leakage would result from the case in which an in-core instrument penetration and its weld were sheared off, but the instrument tube remained in the hole in the reactor vessel wall.

Another set of analyses evaluated the case in which an additional unspecified mechanism would force the instrument tube out of the vessel wall. This would result in a 1-inch-diameter hole and 120-gallon-per-minute leak. The licensee had safety systems to make up this potential leakage. These systems included gravity feed from the borated water storage tank and forced circulation via the containment building recirculation pumps. The cavity under the reactor vessel contained borated water to preclude criticality in the event that any fuel was flushed down with the leaking water.

8.4 Pre-Defueling Preparations

8.4.1 Containment Building Decontamination and Dose Reduction Activities

- ***Purpose***. To conduct decontamination and dose reduction activities in the containment building at elevation levels 305-feet (entry level) and above. Planned activities included: (●) flushing containment building surfaces with deborated water; (●) scrubbing selected components of the polar crane; (●) conducting hands-on decontamination of vertical surfaces; (●) decontaminating the air coolers; (●) flushing the elevator pit; (●) cleaning floor drains; (●) shielding the seismic gap and penetration; (●) decontaminating and shielding hot spots; (●) decontaminating missile shielding; (●) shielding reactor vessel head surface structure;

(●) removing concrete and paint; (●) decontaminating cable trays; (●) decontaminating equipment; (●) remote flushing the containment building basement; and (●) testing remote decontamination technology.

- ***Evaluation: Structural Integrity (Scrabbling)***. [13] Scrabbling concrete walls and floors was used as a decontamination technique to remove contaminated coatings and the surface layer of concrete. The licensee's safety evaluation stated that the thickness of the layer removed would be limited in the case of some structures to ensure their structural integrity. Scrabbling thickness in the containment building was limited by engineering change authorization. Scrabbling thickness greater than these limits would be evaluated on a case-by-case basis to ensure that proper concrete configurations were maintained.

- ***NRC Review***. [14] Editor's Note: The NRC's safety evaluation report did not specifically address this evaluation topic.

8.4.2 Reactor Coolant System Refill

- ***Purpose***. To refill the reactor coolant system (RCS) to the top of the hot legs in order to purge oxygen and to provide an RCS water level that would permit operation of the once-through steam generator (OTSG) recirculation/cleanup system. To operate the OTSG recirculation/cleanup system, the secondary-side water level in the OTSG must be raised to the vicinity of the upper tubesheet to minimize the chance of unborated water leakage from the OTSGs to the RCS.

As an added measure of protection against system overpressurization, the pressurizer would not be vented. This protective measure provided a surge volume for increases to the RCS or for inadvertent introduction of pressurization to the RCS, such as by activating pumps or changing valve lineups.

- ***Evaluation: RCS Integrity***. [15, 16] The licensee's safety evaluation noted that its initial submittal included plans to vent the pressurizer. However, the licensee subsequently decided not to vent the pressurizer in order to provide added protection against system overpressurization by supplying a surge volume for increases to the RCS inventory or for inadvertent introduction of pressurization to the RCS, by activating pumps and changing valve lineups.

- ***NRC Review***. Editor's Note: The NRC's safety evaluation was not located.

8.4.3 Reactor Vessel Head Removal Operations

8.4.3.1 Polar Crane Load Test

- ***Purpose***. To conduct load testing of the polar crane in preparation for reactor vessel head lift and removal. The polar crane load test required four major sequential activities: (1) relocation of the internals indexing fixture; (2) assembly of test load; (3) load test; and (4) disassembly of test load.

- ***Evaluation: Reactor Coolant System Integrity***. [17] The licensee's safety evaluation provided a detailed analysis that examined the potential consequences of load drops in accordance with guidance in NUREG-0612. Based on calculations of damage following an accidental drop of a postulated heavy load, Criterion III of the evaluation criteria in NUREG-0612 stated that damage to the reactor vessel or the spent fuel pool was limited so as not to result in water leakage that could uncover the fuel (makeup water provided to overcome leakage should be from a borated source of adequate concentration if the water being lost was borated). The safety evaluation of a heavy load drop in the vicinity of the reactor core included the identification of loads, targets, and load/target interactions, as well as the evaluation of the worst credible consequence.

    - *Identification of Loads*. For the performance of the polar crane load test, the loads to be moved in the vicinity of the reactor core were the reactor vessel missile shields. These missile shields were constructed in the shape of oblong blocks of concrete and rebar, weighing about 40 tons each. The safety evaluation report provided figures that showed the lift paths.

    - *Load/Target Interaction*. If a shield block were to fall onto the reactor head and service structure, damage to the control rod drive mechanism motor tubes would cause leakage of reactor coolant into the containment building. The maximum leakage would be the draining of the reactor coolant system to the top of the level of the reactor vessel closure head since penetration of the 8-inch-thick steel closure head was not credible. A shield block striking the reactor vessel closure head could also bring about some physical redistribution of loose core debris within the reactor coolant system. The evaluation noted that much of the kinetic energy of the falling shield block would be absorbed in physical deflection of the tall service structure and control rod drive mechanism apparatus above the head and that an instantaneous impact directly on the head would not occur.

    - *Results: Fuel Uncovery (Criterion III)*. The maximum leakage resulting from a drop of a missile shield block onto the reactor head and service structure would be drainage of the reactor coolant system to the level of the top of the reactor vessel closure head. Drainage to this level would not uncover the fuel; therefore, Criterion III would be met. In addition, at least one makeup train capable of delivering water with a boron concentration of 3500 parts per million to the reactor vessel would be available.

- ***NRC Review: Reactor Vessel Integrity***. [18] The NRC's safety evaluation concluded that the severing of the in-core instrument tubes that penetrated the lower reactor vessel head was not a credible event for a number of reasons. First, a significant amount of the kinetic energy in the missile shield would be dissipated in the deformation of the service structure. Second, the missile shield would have to fragment into pieces so that it would fit within the physical constraints of the reactor vessel and surrounding concrete structure, which was an annulus of slightly less than 2 feet of maximum clearance. The fragmented pieces would have to clear the four 28-inch-diameter inlet pipes, two 36-inch-diameter outlet pipes, and two 14-inch-diameter core flood tank pipes that penetrated the reactor vessel. Having cleared the vessel piping, a piece of missile shield no larger than about 9 inches in maximum dimension would have to strike the concrete pad supporting the reactor vessel, bounce at a 90-degree angle and pass through one of the 9.25-inch diameter holes in the reactor vessel support skirt, and strike an in-core instrument tube with enough energy to fail 0.22-inch-thick tubing. The NRC considered such an event to be incredible.

8.4.3.2 First-Pass Stud Detensioning for Head Removal

- ***Purpose***. To perform the first-pass detensioning of the 60 reactor vessel studs and the removal of up to 5 reactor vessel studs to check for stuck nuts and to examine the condition of the removed studs.

- ***Evaluation: Reactor Coolant System Integrity***. [19, 20] The licensee's safety evaluation considered the potential degradation in the pressure-retaining capability of the closure head seals.

  - *Method*. The methodology used to determine the permissible internal pressure after the first detensioning pass and after removal of up to five studs was given in the following steps.
    (1) Execute the computer programs for calculating permissible internal pressure.
    (2) Determine the total stud load (P) based on the stud elongations calculated by the computer program for the conditions following the detensioning of the third pair of studs (the third sequence) in the second detensioning pass: [$P = (N \times A \times E \times \Delta)/(L)$], where P = total stud load (pounds); N = number of studs; A = effective stud area (square inch); E = modulus of elasticity (pounds per square inch); Δ = stud elongation (inch); and L = effective stud length (inch). (At this stage of the detensioning, six studs would be fully unloaded and could be removed.) (3) Determine the springback load(s) for the gaskets using deflection versus gasket-loading data from the manufacturer. The gaskets were initially positioned in grooves in the flanges. The groove depth provided a set gasket deformation recommended by the manufacturer to provide for sealing when the flanges were brought in contact during the initial seating of the gaskets. Since the flanges remained in contact after the first-pass detensioning, removal of five studs, and internal pressure loading up to some pressure (p), the springback load providing the sealing force remained essentially constant; therefore, the initial gasket sealing capability was maintained. (4) Calculate the allowable internal pressure (p): [$p = (P-S)/\pi R^2$], where S = springback load, P = total stud load (pounds) as calculated in Step 2, and R = radius of outermost gasket.

- *Assumptions*. The assumptions used in the analysis included the following: (●) Stud tensioner characteristic curves supplied by the manufacturer were still valid. (●) Studs and flanges were not yielded or deformed (i.e., the existing stud elongations were the same or less than those taken during the last tensioning operation). (●) Variations of stud elongations within the bolt pattern were minor. (●) Gaskets maintained their structural and metallurgical properties. (●) The weight of the closure head was conservatively neglected.

- *Results*. Computer analysis showed that the reactor vessel head seal would be capable of maintaining up to 1600 pounds per square inch gauge (psig) pressure after first-pass detensioning of all 60 studs and the removal of up to 5 studs. To account for uncertainties of key assumptions and to provide for a safety margin, 1000 psig was selected as the maximum internal pressure. Therefore, if the reactor coolant system (RCS) required repressurization during any phase of this stud detensioning or stud removal activity, even to the pressure limit of 600 psig established in the recovery technical specification, the reactor vessel head seal pressure boundary would be capable of maintaining the pressure.

  Further, the evaluation concluded that no damage to the silver plating was expected during the accident and that the surface of the "O" rings was unperturbed and intact. Two "O" ring gaskets of austenitic stainless steel with silver plating were compressed between the reactor vessel head flange and the reactor vessel flange to act as pressure seals. The Electric Power Research Institute [21] performed a temperature analysis for the 60-minute boildown transient that occurred 113 minutes after the turbine trip at the start of the accident. At the flange region, the maximum inside surface temperature of about 900 degrees Fahrenheit (degrees F) was reached at the end of the 60-minute time interval (or 173 minutes following the turbine trip), which was well below the melting point of silver (1615 degrees F).

- ***NRC Review: Reactor Coolant System Integrity***. [22] The NRC's safety evaluation considered the potential degradation in the pressure-retaining capability of the closure head seals. The evaluation considered the remaining bolt force, overpressurization events, and deformation of the two metallic "O" ring gaskets.

- *Bolt Force*. The NRC review indicated that the partial unloading (i.e., first-pass detensioning) of all studs and subsequent full detensioning and removal of up to five studs would still leave the reactor vessel head with about 39 million pounds of bolt force on the vessel flange. This value conservatively neglected the weight of the reactor vessel head. This force corresponded to a vessel pressure-retaining capability of about 1600 psig, which was well in excess of the licensee's objective of a pressure-retaining capability of 1000 psig. The capability for pressurizing the RCS (limited to 600 psig by the recovery technical specifications) before future reactor vessel head removal was desirable in that it would facilitate, for example, the processing of reactor coolant.

- *Overpressurization*. The RCS was protected from any credible pressurization event (i.e., operation of the standby pressure control system for processing reactor coolant) by relief valves that could automatically lift system pressure below the technical specification lift setpoint. This protection from overpressurization was provided both by the postaccident standby pressure control system, which maintained system pressure when in use, and by the RCS piping when the system was isolated from the standby pressure control system. Therefore, the reduction in pressure-retaining capability resulting from first-pass stud detensioning would not affect system ductility concerns.

Moreover, there was little potential for any credible event that could lead to pressurization of the RCS in excess of the technical specification limit. The pressure-retaining capability of the reactor vessel following first-pass stud detensioning was more than double the pressure limited by the technical specification (600 pounds per square inch absolute pressure).

- "*O" Ring Deformation (Detensioning)*. As part of the NRC evaluation of the pressure-retaining capability of the reactor vessel, the NRC examined the impact that first-pass stud detensioning could have on the two metal "O" ring gaskets in the reactor vessel flange. Because of the geometrical configuration of the vessel and the head mating surfaces, along with the location of the studs in the perimeter of the flange, the fully tensioned condition produced a sight deflection or gap between the rating surfaces at the reactor vessel radial location inside the "O" ring gaskets. When the studs were detensioned, this gap was reduced. Thus, first-pass stud detensioning actually increased the compression of the mating surfaces on the flange "O" ring seals, thereby maintaining the full integrity of the flange seals. The NRC also noted that the "O" ring was designed as hollow tubes with slotted holes that permitted the "O" rings to expand against the flange mating surfaces when subjected to rising vessel pressure and ensured a good seal.

- *"O" Ring Deformation (Accident Temperature*). The NRC also examined the licensee's estimates for the maximum temperatures that the "O" ring gaskets experienced during the accident sequence to determine if the silver cladding (0.004 to 0.006 inch thick) on the stainless-steel gaskets degraded and affected sealing capability. The NRC concurred with the licensee's analysis and concluded that temperatures in the vicinity of the "O" rings would be bounded by the estimated maximum temperature of about 405 degrees Celsius, inside the surface of the vessel in the flange region during the accident sequence. The "O" rings were protected to a degree by virtue of their location in the reactor vessel flange, which consisted of a massive amount of metal and was capable of acting as a heat sink during a transient. The NRC concluded that temperatures in the flange during the accident sequence would not have affected the integrity of the silver cladding (the melting point of silver was about 960 degrees Celsius) on the "O" ring seal. Accordingly, if the vessel were refilled and pressurized at some time in the future, the "O" ring should be capable of performing its intended function with little risk of leakage through the flange seal.

8.4.3.3 Reactor Vessel Head Removal Operations

- ***Purpose***. To prepare for reactor vessel head removal to perform the lift and transfer of the head and conduct the activities necessary to place and maintain the reactor coolant system (RCS) in a stable configuration for the next phase of the recovery effort.

- ***Evaluation: RCS Integrity (Reactor Coolant)***. [23] The licensee's safety evaluation provided a detailed analysis of the potential consequences of load drops in accordance with guidance in NUREG-0612. Based on calculations of damage following an accidental drop of a postulated heavy load, Criterion III of the evaluation criteria in NUREG-0612 stated that damage to the reactor vessel or the spent fuel pool was limited so as not to result in water leakage that could uncover the fuel (makeup water provided to overcome leakage should be from a borated source of adequate concentration if the water being lost was borated). The safety evaluation of a heavy load drop in the vicinity of the reactor core included the identification of loads, targets, and load/target interactions, as well as the evaluation of the worst credible consequence.

  - *Identification of Loads*. The reactor vessel closure head with the attached control rod drive mechanisms and the service structure have a combined weight of less than 170 tons. After the head lift prerequisites were completed, the polar crane was used to first lift the head several inches above the reactor vessel flange and then to a height that cleared the guide studs and the control rod guide tubes. The head was then to be moved to the south end of the refueling canal such that it cleared the reactor vessel before any further vertical lift. The detailed instructions for accomplishing this task would specify the exact lifting heights and path coordinates.

  - *Load/Target Interactions*. Attachment 3 of this safety evaluation report analyzed the effects of the reactor vessel head drop onto the vessel itself. These analyses showed that the reactor vessel and appurtenances could withstand the impact of the vessel head and support structure if they were dropped on the vessel flange from a height of 56.1 inches or less. The head lift and removal procedure and the detailed instructions for accomplishing this task were to specify that, until the head cleared the vessel, the head would be lifted no higher than this maximum height.

  - *Results: Fuel Uncovery (Criterion III)*. The analysis of a postulated head drop on the vessel showed that the reactor vessel could withstand the impact of the head and the support structure. Therefore, there would be no water leakage from the RCS and Criterion III would be met.

- ***Evaluation: RCS Integrity (Pressure Boundary)***. [24] The licensee's safety evaluation considered the impact on RCS integrity during this activity. The RCS pressure boundary could not be restored to any assured pressure rating given that all the reactor vessel head closure studs were fully detensioned. The design purpose of the RCS pressure boundary was to protect the health and safety of the public from design-basis accidents with the plant at power. During head removal activities, the reactor would be in shutdown operation. In this condition, the functions of the RCS were to remove decay heat and to maintain boron concentration for reactivity control. Maintenance of the RCS pressure boundary was not necessary for the RCS to perform its shutdown mode functions. During and after head removal activities, the safety function of the RCS was to maintain a sufficient volume of adequately borated water for decay heat removal and for maintenance of subcriticality of the core. In addition, the RCS served as a water shield for the radiation sources inside the vessel. For the head removal, no events were postulated to occur that would cause the RCS to become pressurized; therefore, the RCS pressure boundary was not required during head removal activities. The function of the vessel to contain the reactor coolant would be maintained.

- ***NRC Review: RCS Integrity***. [25] Section 5.1.3 of NUREG-0612 provided guidance concerning the design and operation of load handling systems in the vicinity of the reactor core. The licensee was required to demonstrate that adequate measures were taken to ensure that in the vicinity of the core, either the likelihood of a drop that might damage spent fuel was extremely small or that the estimated consequence of such a drop would not exceed the limits set by the evaluation criteria of NUREG-0612, Section 5.1, Criteria I though III. Based on calculations of damage following an accidental drop of a postulated heavy load, Criterion III required that damage to the reactor vessel or the spent fuel pool was limited so as not to result in water leakage that could uncover the fuel (makeup water provided to overcome leakage should be from a borated source of adequate concentration if the water being lost was borated).

All heavy load movements in the head removal program were made in the containment building, and there were no movements that could impact the spent fuel pool. Heavy loads that would be handled during the heat removal evolution included the reactor vessel head assembly, which comprised the (●) lift ring, (●) vessel closure head, (●) control rod drive mechanism motor tube assemblies, (●) service structure, and (●) attached shielding with its support frames. Other heavy loads included the internals indexing fixture, the fixture cover, and the cover shielding plates. The lifting of any combination of loads was limited to a maximum weight equal to the 170-ton rating of the containment building polar crane.

While in the vicinity of the reactor vessel, the head lift height would be monitored and controlled. Given that all the reactor vessel studs were successfully removed, the head lift height for required clearance would be about 33 inches. Cover placement on the underside of the head following the initial lift could require an additional 12 inches of clearance over the reactor vessel. Thus, the head lift height in the vicinity of the reactor vessel would be no more than 45 inches.

The licensee provided an evaluation of the effects of a load drop on the reactor vessel in Attachment 3 of its head lift safety evaluation report. In this analysis, the licensee stated that, depending on the components that were attached to the head during the lift, the assembly could weigh from 158 to 170 tons. Each weight had a corresponding maximum lift height, equal to the maximum vertical distance that the load could be dropped without a breach of RCS integrity (i.e., failure of the reactor vessel or the attached in-core instrument tubes). For reference, the licensee had analyzed the case in which the service structure shielding blanket was in place and all studs were removed. However, all studs had been removed in preparation for head lift. The weight under these conditions was conservatively assumed to be 174 tons. This assumed weight was about 10 tons more than the actual weight of the head and attached shielding (about 163 tons). The maximum lift height, assuming a worst case point load drop (structure tilts when dropped and hits the vessel or plenum at an angle), was calculated to be 56.1 inches. The NRC structural engineering branch reviewed the licensee’s load drop calculations and confirmed the results.

The NRC concluded that measures to limit head lift height in the vicinity of the reactor vessel were adequate to mitigate the consequences of an accident. Therefore, there was adequate protection against uncovering the fuel, and Criterion III was satisfied.

8.4.4 Heavy Load Handling inside Containment

- ***Purpose***. To ensure that handling of heavy loads in the containment building and spent fuel pool "A" (SFP-A) within the fuel handling building was in accordance with the safety requirements of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," (26) issued July 1980.

- ***Evaluation: Reactor Vessel/SFP-A Integrity***. (27) The licensee evaluated the results of load drops postulated in this safety evaluation report (SER) against Criterion III on reactor vessel integrity of NUREG-0612. As loads would not be handled over the in-core instrument tubes, the load drops postulated in this SER could not drain the reactor vessel below the bottom of the reactor vessel hot leg at the 314-foot elevation. Drainage to this level would not uncover the fuel. The makeup system could provide makeup water via redundant pathways to the reactor vessel.

The drop of a heavy load, handled in accordance with the guidelines in this SER, in the deep end of the fuel transfer canal (FTC) or in SFP-A could result in local damage to the stainless-steel liner plate. The extent of this damage would be determined by the shape and weight of the dropped load and could range from denting to perforation of the liner plate. The perforation of the liner plate could result in water being lost from SFP-A/FTC; this water would be collected by the liner leakage collection system and directed to the auxiliary building sump for SFP-A leakage or containment building sump for FTC leakage. The borated water storage tank would provide necessary makeup. The catastrophic failure of the slab in the deep end of the FTC was not considered credible because of the existence of a concrete support wall located at the center of the slab.

The technical evaluation report for defueling canisters described an analysis to determine the potential for criticality to occur in SFP-A/FTC because of a catastrophic failure of the liner causing SFP-A/FTC to be drained of water. This analysis determined that a criticality event would not occur.

- ***NRC Review: Reactor Vessel/SFP-A Integrity***. (28) The NRC reviewed the results of load drops postulated in this SER against Criterion III of NUREG-0612. The NRC's review of the licensee's analysis for load drop and leakage in the reactor vessel was provided in a previous NRC SER (29) for heavy load handling over the reactor vessel. If SFP-A or the FTC was to drain, there would be no immediate effects. The entire core had a decay heat of less than 12 kilowatts; heat generation in individual stored canisters would be less than 100 watts and posed no problem. Potential for gas generation in the canisters was evaluated in a previous NRC SER (30) for the defueling canisters and found acceptable. Thus, the canisters would remain stable for long periods of time in a drained pool.

8.4.5 Heavy Load Handling over the Reactor Vessel

- ***Purpose***. To permit the handling of heavy loads (greater than 2400 pounds including rigging weight) in the vicinity of the reactor vessel. This included any load handling activity that could result in a heavy load drop onto or into the vessel either directly or indirectly (as the result of the collapse of structures or equipment installed over the vessel).

- ***Evaluation: Reactor Vessel Integrity***. [31] The licensee's safety evaluation stated that the only postulated failure mechanism that could potentially lower the water level in the reactor vessel below the bottom of the coolant pipe nozzles was damage to the in-core instrument tubes at the point where they penetrated the lower vessel head. A total of 52 such penetrations were distributed in the lower head. The analyses of load drop impacts on reactor vessel integrity included: (●) review of leakage experience; (●) identification of loads; (●) identification of targets; (●) evaluation of load/target interactions; and (●) evaluation of postulated failure mechanisms.

Editor's Note: Chapter 7 of this NUREG/KM on load drop evaluations gives details of the licensee's safety evaluation.

- ***NRC Review: Reactor Vessel Integrity***. [32] Chapter 7 of this NUREG/KM on load drop evaluations contains details of the NRC evaluation.

8.4.6 Plenum Assembly Removal Preparatory Activities

- ***Purpose***. To confirm the adequacy of plenum removal equipment and techniques, the preparatory activities included: (●) video inspection of potential interference areas (●) video inspection of the core void space; (●) video inspection of the axial power shaping rod (APSR) assemblies; (●) measurement of the loss-of-coolant accident restraint boss gaps; (●) measurement of the elevations of the APSR assemblies; (●) cleaning of the plenum and potential interference areas; (●) separation of unsupported fuel assembly end fittings; and (●) movement of the APSR assemblies.

- ***Evaluation: Reactor Vessel Integrity***. [33] As part of the safety evaluation of a load drop, the licensee concluded that the structural integrity of the reactor vessel and its support skirt would not be compromised. In addition, the resulting reactor vessel displacements would not cause stresses on attached piping to exceed their faulted condition stress limits given in the American Society of Mechanical Engineers, *Boiler and Pressure Vessel Code*, Section III, "Rules for Construction of Nuclear Facility Components," 1974 Edition. [34] This would preclude failure of the attached piping.

- ***NRC Review: Reactor Vessel Integrity***. (35) The NRC's safety evaluation in the agency's previous safety evaluation report (36) for the removal of the reactor vessel head concluded that reactor coolant system integrity would be intact. The NRC concluded that there were adequate measures to mitigate the consequences of postulated accidents during plenum removal preparatory activities.

8.4.7 Plenum Assembly Removal

- ***Purpose***. To remove the plenum assembly from the reactor vessel to gain access to the core region for defueling. The plenum was moved to the deep end of the fuel transfer canal, which contained reactor coolant for shielding.

- ***Evaluation***. (37) Editor's Note: The licensee's safety evaluation report did not specifically address the reactor coolant system or reactor vessel integrity; however, its evaluation did consider the effects of a load drop over the reactor vessel. Chapter 7 of this NUREG/KM on load drop evaluations contains details of the licensee's safety evaluation.

- ***NRC Review***. (38) Editor's Note: The NRC's safety evaluation report did not specifically address reactor coolant system or reactor vessel integrity; however, the agency's evaluation did consider the effects of a load drop over the reactor vessel. Chapter 7 of this NUREG/KM on load drop evaluations contains details of the NRC's evaluation.

8.4.8 Makeup and Purification Demineralizer Resin Sampling (NA)

8.4.9 Makeup and Purification Demineralizer Cesium Elution (NA)

8.5 Defueling Tools and Systems

8.5.1 Internals Indexing Fixture Water Processing System

- ***Purpose***. To provide reactor coolant water processing capability during head removal until plenum removal. The internals indexing fixture (IIF) processing system was selected based on the limited reactor coolant processing capability in the drained-down condition and the desire to provide adequate water cleanup capacity to minimize radiation dose rates around the IIF.

- ***Evaluation***. (39) The licensee's safety evaluation did not specifically address this topic.

- ***NRC Review: Reactor Vessel Integrity (Leak)***. (40) The licensee's safety evaluation addressed reactor vessel leak detection and specified that IIF processing would be stopped every 72 hours to measure leakage from the reactor vessel, over a 2-hour or 4-hour period. Because of the sensitivity of the existing reactor vessel level detectors, a leak check over a 2-hour period could fail to detect leaks as large as 1.3 gallons per minute. Since station procedures would initiate emergency actions when leak rates reached 1 gallon per minute, the NRC felt that the 4-hour leak detection period should be used to lower the leak detection threshold to below 1 gallon per minute. The NRC site staff, by means of the normal procedure approval process, would ensure that leak checks were performed over a 4-hour period.

8.5.2 Defueling Water Cleanup

8.5.2.1 Defueling Water Cleanup System (NA)

8.5.2.2 Cross-Connect to Reactor Vessel Cleanup System

- ***Purpose***. To modify the fuel transfer canal/spent fuel pool "A" (FTC/SFP-A) cleanup system portion of the defueling water cleanup system (DWCS) to allow processing of the FTC/SFP-A water through the "B" train of the DWCS reactor vessel cleanup system. The purpose of this modification was to provide the capability to effectively process the FTC/SFP-A water in a similar manner to the reactor vessel cleanup process without the installation of additional body-feed and coagulant equipment in the fuel handling building. In addition, the proposed modification would authorize the use of FTC/SFP-A filtered effluent as a water source for the body-feed tank and as dilution water for the coagulant addition unit.

- ***Evaluation: Reactor Coolant System Integrity (Leak)***. [41] The licensee's safety evaluation stated that during processing of FTC/SFP-A water in the proposed modified system configuration, no net change in FTC/SFP-A water levels would be observed since the water in the FTC and SFP-A communicated through the fuel transfer tubes. If a rupture occurred in the FTC/SFP-A cleanup system, the available FTC pump could deliver FTC/SFP-A water to the fuel handling building or containment building. This event would lower the water level in the canal and the reactor vessel pool. However, a level loss would be detected by redundant level indicating systems, one each for the FTC and SFP-A, which were provided with low-level alarms in the control room. Upon receipt of either low-level alarm, the system would be shut down manually, and the loss of water inventory would be investigated.

- ***NRC Review: Reactor Coolant System Integrity (Leak)***. [42] The NRC's safety evaluation concluded that the likelihood of accidents and the consequences of those accidents involving an impact of system leakage, a boron dilution potential, and a potential for inadvertent criticality were within the bounds of the analysis in the original approval of the licensee's safety evaluation report [43] for the DWCS.

8.5.2.3 Temporary Reactor Vessel Filtration System (NA)

8.5.2.4 Filter-Aid Feed System and Use of Diatomaceous Earth as Feed Material (NA)

8.5.2.5 Use of Coagulants

- ***Purpose***. To demonstrate the use of coagulants and body-feed material to improve the performance of the defueling water cleanup system (DWCS) filter canisters in maintaining water clarity. Operating experience with the DWCS had not achieved the desired clarity in the reactor coolant system (RCS) water to support defueling operations within the reactor vessel. The DWCS filters required changeout because of high differential pressure without the expected high filter throughput. The root cause of shortened filter canister life was expected to be the presence of hydrated metallic oxides in colloidal suspension within the RCS that were plugging the filter media. The addition of the coagulant with body-feed was expected to agglomerate the colloids to filterable sizes, thus forming a filter cake on the filter media.

- ***Evaluation***: ***RCS Integrity (Chemistry)***. [44,45] The licensee's safety evaluation stated that the RCS and makeup water to the RCS must meet recovery technical specification requirements for boron (4950 to 6000 parts per million (ppm)), pH (7.5 to 8.4), and chloride (less than or equal to 5 ppm). The treated and filtered water returning to the reactor vessel must maintain the RCS within the above requirements. Even though the addition of diatomaceous earth would not affect these RCS requirements, the quantity of diatomaceous earth that could be introduced was minimized since the water was filtered before its return to the reactor vessel. The addition of the coagulant, however, could impact these RCS requirements. The extent of impact depended on the quantity of coagulant added to the RCS and subsequent removal by filtration.

  - *Boron Concentration*. Section 3.1.1 in the criticality section of the safety evaluation report addressed the impact on boron concentration in the RCS from the addition of coagulant. Existing boron meters would monitor the RCS boron concentration. Weekly chemical analysis of the RCS samples would also monitor the boron concentration. Thus, adequate means were implemented to ensure that water returning to the reactor vessel would not reduce the boron concentration in the RCS below the allowable limit. Testing showed that the boron concentration in water treated with coagulant and body-feed was not impacted by filtration, ion exchange, or processing through activated charcoal for removal.

  - *pH Level*. Laboratory testing showed that the addition of coagulant could reduce the pH of the treated water. This reduction was shown to be less than 2 percent at coagulant concentrations of 50 ppm. The effect on pH was even less at lower coagulant concentrations. Testing also showed that the pH of the water treated with coagulant and body-feed was unaffected by filtration.

  - *Chloride Concentration*. Testing showed that the addition of the coagulant increased the chloride concentration in the treated water. This increase was dependent on the concentration of the coagulant in the treated water. Laboratory analyses showed that the chloride concentration increased by about 0.05 ppm for a 1-ppm concentration of coagulant. Repeated RCS treatments would build up the chloride in the RCS. The coagulant addition to the RCS would be terminated before the RCS chloride concentration exceeded the 5-ppm concentration limit.

- ***NRC Review***. [46] Editor's Note: The NRC's safety evaluation report did not specifically address this evaluation topic.

8.5.2.6 Filter Canister Media Modification (NA)

8.5.2.7 Addition of a Biocide to the Reactor Coolant System

- ***Purpose***. To evaluate the effects of adding hydrogen peroxide as a biocide to the reactor coolant system (RCS) to aid the removal of biological contamination. This biological growth reduced water clarity and fouled the water processing system filters.

- ***Evaluation: RCS Integrity (Chemistry)***. [47] The licensee's safety evaluation considered the effect of the low concentrations (i.e., 200 parts per million) of hydrogen peroxide (H_2O_2) on RCS resin/zeolite processing performance. The general conclusion was that low levels of H_2O_2 would not seriously affect the performance or stability of the resin/zeolites. Further, H_2O_2 would have no effect on the diatomaceous earth filter operation. This conclusion was based on previous experience in the fuel transfer canal. Laboratory testing checked the effect of the low concentration of H_2O_2 on the effectiveness of the defueling canister catalyst. Catalyst performance was tested in a DOE laboratory with simulated TMI-2 RCS fluid at 500 parts per million of H_2O_2 without lasting detrimental effects.

- ***NRC Review: RCS Integrity (Chemistry)***. [48] The NRC's safety evaluation concluded that no significant accelerated corrosion of reactor internal components was expected at the proposed concentrations of H_2O_2. Laboratory testing showed that H_2O_2 did not cause a permanent degradation of the catalyst material used in the defueling canisters. No significant reduction in the performance of the zeolite and organic resins used in the water processing systems was expected. Any reduction in the resin's ion exchange capacity would require an increase in the frequency of resin replacement but would have no impact on the safety of cleanup systems' operation.

8.5.3 Defueling Canisters and Operations (NA)

8.5.3.1 Defueling Canisters: Filter, Knockout, and Fuel (NA)

8.5.3.2 Replacement of Loaded Fuel Canister Head Gaskets (NA)

8.5.3.3 Use of Debris Containers for Removing End Fittings (NA)

8.5.3.4 Fuel Canister Storage Racks (NA)

8.5.3.5 Canister Handling and Preparation for Shipment (NA)

8.5.3.6 Canister Dewatering System (NA)

8.5.3.7 Use of Nonborated Water in Canister Loading Decontamination System (NA)

8.5.4 Testing of Core Region Defueling Techniques

- ***Purpose***. To use hydraulic heavy-duty defueling tools for limited bulk defueling operations on the hard crust layer of the damaged core.

- ***Evaluation: Reactor Vessel Integrity***. [49] The licensee's safety evaluation concluded that the most significant safety impact that could result from this activity would be the potential effect on in-core instrument strings and associated in-core instrument nozzle and welds. The safety evaluation report [50] for bulk defueling determined that the minimum strength of an in-core instrument nozzle weld was about 5400 pounds in tension or compression. Additionally, tests determined that the breaking strength of an in-core instrument string was about 4000 pounds force. Therefore, the licensee expected that an instrument string would fail before its associated nozzle weld. However, because the analysis of the in-core instrument nozzle weld was not reviewed and approved by the NRC, the licensee proposed to impose a net uplift force limit of about 2000 pounds, in excess of the specific tool weight, on the tools to be tested. This limit would provide a significant margin of safety for the in-core instrument nozzle weld. The proposed uplift force limit equated to about 50 percent of the force required to cause an in-core instrument string failure and about 37 percent of the minimum calculated in-core weld strength. Therefore, the maintenance of integrity of the in-core instrument nozzle welds would be ensured.

- ***NRC Review: Reactor Vessel Integrity***. [51] The NRC's safety evaluation stated that the proposed activities presented a somewhat greater likelihood for damage to the in-core instrumentation guide tubes than those activities previously approved; however, the probability of a nonisolatable reactor coolant system (RCS) leak was low. The most likely failure mode during the proposed activity would be the failure of an in-core instrument guide tube nozzle or weld due to inadvertent grappling and lifting of an in-core instrument string using the heavy-duty spade bucket tool. Data indicated that as a result of the 1979 accident, most of the in-core instrument guide tubes would probably not extend very far into the original core volume. Therefore, it was unlikely that operations conducted on the hard crust layer would inadvertently grapple an in-core instrument string. Additionally, the licensee would limit the lifting force applied to the tested tools to 2000 pounds in excess of tool weight, thereby limiting the potential loads placed on the in-core guide tubes, nozzles, and welds. The NRC found that this limitation was likely to preclude damage to the in-core guide tubes that would cause a leak that could not be isolated. If an instrument string was advertently grappled during the proposed activities and RCS leakage results, as discussed in the NRC's safety evaluation report [52] for early defueling, adequate leak detection capability and RCS makeup capacity were available to identify the leak and maintain the required volume of borated water in the reactor vessel.

The NRC concluded that the potential for failure of the in-core instrumentation guide tube nozzles and welds as a result of the proposed activities was acceptably low. In addition, adequate methods of detection and corrective actions were available in the unlikely event of RCS leakage that could not be isolated. The long-term use of these tools and methods would be addressed in the NRC review of reactor vessel defueling. The NRC concluded that the proposed limited testing of the heavy-duty tong tool, the heavy-duty spade bucket, and the hydraulic impact chisel for bulk defueling operations could be performed without significant risk to the health and safety of the public.

8.5.5 Fines/Debris Vacuum System

- ***Purpose***. To modify the fines/debris vacuum system using a knockout canister and a filter canister in series. Modifications included: (●) use of a vacuum nozzle to allow larger debris particles to be vacuumed into the knockout canisters; (●) use of mechanical probes and water jets on the end of the vacuum nozzle to loosen the packed rubble; (●) use of a larger vacuum tool to allow debris removal from the lower head; and (●) temporary use of the vacuum system without a filter canister. The initial use of the fines/debris vacuum system was previously approved in the safety evaluation report [53, 54] for early defueling.

- ***Evaluation: Reactor Vessel Integrity***. [55] The licensee's safety evaluation stated that vacuuming debris in the lower head region did not present any additional safety concerns not previously evaluated in the SER [56] for early defueling, except that the vacuuming could be in the proximity of the in-core instrument nozzles, which protruded from the lower head vessel wall. However, the manipulator arm could not deliver an impact force able to compromise the integrity of the in-core instrument nozzles or of the instrument tube-to-vessel-wall welds, which provided the instrument tube penetration seals. Even though lower head vacuuming was not in the scope of the SER for early defueling, the licensee considered this operation to be bounded by evaluations provided in that report since this operation could not endanger reactor vessel integrity.

- ***NRC Review: Reactor Vessel Integrity***. [57] The use of a larger vacuum tool permitted vacuuming debris from the lower head. The only potential safety concern was impacting the in-core instrument tubes with the vacuum nozzle. The NRC's safety evaluation stated that the low-impact load imparted by the vacuum nozzle on the in-core instrument tubes was unlikely to cause damage. The potential consequences of in-core instrument tube failure were analyzed previously in the NRC's SER [58] for the support of a technical specification modification for changes in borating requirements.

The NRC's safety evaluation of the modification noted that the only credible leakage path from the reactor vessel below the nozzles was through the postulated failure of the in-core instrument tubes, which penetrated the bottom of the vessel. The evaluation considered the credible causes of failure of these in-core instrument tubes and the resulting reactor coolant leakage rates. The worst case potential leak rate resulting from a load drop onto the reactor vessel that breached the in-core instrument tubes was previously reported in the NRC's SER [59] for heavy load handling over the reactor vessel. The bounding leakage rate postulating the breaking of all 52 in-core instrument tubes was about 20 gallons per minute. Other potential causes of in-core instrument tube failures were also considered (e.g., corrosion failures). However, such other causes of failure would not likely result in a leak rate higher than the 20 gallons per minute that was assumed for the simultaneous break of all 52 tubes. Heavy load handling with potential consequences more severe than those analyzed in the safety evaluation for heavy load handling was not expected during the remaining defueling operation. Should such a requirement arise, the licensee was required to submit a safety analysis for NRC review and approval before the operation. Other conditions, such as pathway restrictions, could be imposed so that the potential reactor coolant system leak rate due to a load drop accident would be kept below the estimate of 20 gallons per minute.

8.5.6 Hydraulic Shredder

- ***Purpose***. To utilize a hydraulically powered shredder in order to reduce the size of fuel pins and other core debris and to facilitate the loading of fuel canisters or debris buckets.

- ***Evaluation: Reactor Vessel Integrity***. [60] The licensee's safety evaluation concluded that potential loadings directly onto the reactor vessel and support structures as a result of a load drop were bounded by its safety evaluation report [61] for heavy load handling over the reactor vessel. This conclusion assumed that the bottom of the shredder remained at or below the 333-foot elevation during transfer. If the impact was transmitted directly onto the debris bed, the limiting consequence would be the failure of in-core nozzle welds. This event was also previously assessed in the licensee's safety evaluation. Therefore, potential structural failures or load handling accidents were bounded by previously approved evaluations.

- ***NRC Review***. [62] Editor's Note: The NRC's safety evaluation report did not specifically address this evaluation topic.

8.5.7 Plasma Arc Torch

8.5.7.1 Use of Plasma Arc Torch to Cut Upper End Fittings

- ***Purpose***. Use of a plasma arc torch to cut the upper end fittings inside the reactor vessel to allow the fittings to be placed directly in fuel canisters for transfer from the reactor vessel to the fuel canal. This was the first use of the plasma arc torch inside the reactor vessel.

- ***Evaluation: Reactor Vessel Integrity***. [63] The licensee's safety evaluation stated that cutting of end fittings with the plasma arc torch would be performed below the water level of the reactor vessel. In addition, the torch would not be operated near the internals indexing fixture cylinder or the core support assembly upper cylinder. The process of cutting with the remote, underwater plasma torch required a specific and deliberate operator action. The operator-controlled settings for location, thickness of metal, water depth, speed of cut, and cut location precluded the torch from accidentally cutting material significantly different from that initially programmed. In addition, it was not possible to pass an arc current through more than 0.5 to 0.75 inch of water; therefore, damage to the reactor pressure vessel or the internals indexing fixture was precluded.

- ***NRC Review***. [64] Editor's Note: The NRC's safety evaluation report did not specifically address this evaluation topic.

8.5.7.2 Use of Plasma Arc Torch to Cut the Lower Core Support Assembly

- ***Purpose***. To use the plasma arc torch to cut the lower core support assembly, including the flow distributor head.

- ***Evaluation***. [65] Editor's Note: The licensee's safety evaluation report did not specifically address this topic.

- ***NRC Review***. [66] Editor's Note: NRC reviews were combined under one safety evaluation for the defueling of the lower core support assembly.

8.5.7.3 Use of Plasma Arc Torch to Cut Baffle Plates and Core Support Shield (NA)

8.5.7.4 Use of Air as Secondary Gas for Plasma Arc Torch (NA)

8.5.8 Use of Core Bore Machine for Dismantling Lower Core Support Assembly

- ***Purpose***. To use the core bore machine, in conjunction with the automatic cutting equipment system, to dismantle the lower core support assembly (LCSA) and facilitate defueling by providing access to the reactor vessel lower head.

- ***Evaluation: Reactor Vessel Integrity***. [67] The licensee's safety evaluation focused on the reactor vessel integrity, specifically the integrity of the in-core nozzle welds to the reactor vessel lower head. In-core guide tubes were mounted to the rib section, in-core support plate, and elliptical flow distributor. The guide tube was positioned over the in-core nozzle but not affixed; both were separated by several inches. The guide tube fitted inside the core bore drilling bit and casing as it cut through the layers of the LCSA. Upon the last cut through the flow distributor, the guide tube would drop out of the core bore casing and onto the lower reactor vessel region. Several scenarios were evaluated. (Refer to the figure in Attachment 1 of the safety evaluation report showing the arrangement of the lower core support structure.)

  - *In-core Nozzle Weld Failure*. Previous licensee [68] and NRC [69] evaluations for the core bore operations established two possible in-core nozzle configurations as a result of the

1979 accident. In the worst case, the damage to the reactor vessel lower head consisted of an in-core nozzle melted to the inside diameter of the reactor vessel lower head with a nozzle-to-vessel weld thickness of only 0.030 inch. The significance of this configuration was that if the weld experienced significant damage, the in-core nozzle above the weld would have melted. The other possible configuration was that the in-core nozzle was undamaged. The static load-bearing capabilities (based on 70 percent of the ultimate strength) of the two nozzle configurations were established for undamaged and damaged in-core nozzles. Capabilities for an undamaged in-core nozzle were axial (tension and compression) at 158,000 pounds, bending (moment) at 42,000 inch-pounds, and twisting (torque) at 87,000 inch-pounds. Capabilities for a damaged nozzle weld (0.030-inch in thickness) were axial at 5400 pounds, bending at 1400 inch-pounds, and torque at 5800 inch-pounds.

- *In-core Guide Tube Drop*. Severing of the 15 in-core guide tubes from the LCSA would cause them to drop when the flow distributor head was cut. The potential for hang up within the tool assembly casing was extremely remote because up to the time the in-core guide tube was free to fall, the tool assembly had been moving and the tube had been stationary. The guide tube would drop essentially straight down as a result of being confined within the tool assembly and cutter head, where the tube would impact the reactor vessel lower head debris bed, an intact in-core nozzle, or the lower head proper.

  The maximum weight of a severed guide tube was about 250 pounds. The tube could drop a maximum of 20 inches assuming the tapered nozzle section had been melted off. If the in-core guide tube was undamaged, the tube would drop only 8.5 inches. A 250-pound in-core guide tube that was dropped a maximum of 20 inches could impact the hypothetically melted nozzle and weld that imparted a dynamic compressive stress of about 18,700 pounds per square inch on the nozzle and weld structure. This stress was about one-half of the yield strength and one-quarter of the ultimate strength of the weld material.

  If the reactor vessel lower head debris bed was present at this time, the above calculated loads would be further reduced. In fact, if the tapered end of the in-core guide tube was able to penetrate the debris bed only 1 foot, the impact would be totally absorbed by the debris bed. Further, if an in-core guide tube was already buried in the debris bed, it would most likely not drop at all.

- *Guide Tube Adhered to the Nozzle (Weld Failure)*. If an in-core nozzle was undamaged, a configuration could be postulated in which the in-core nozzle and in-core guide tube were bridged together by solidified debris. In this configuration, torque from the core bore machine could be transmitted directly to the in-core nozzle once the guide tube was severed from the in-core guide support plate. However, as mentioned previously, guide tube hangup within the tool holder would be extremely unlikely. Therefore, the transmission or torque from the core bore machine to the in-core nozzle was not likely. In addition, the core bore machine has a torque limit of 6000 to 36,000 inch-pounds. This was less than the torque required to damage an intact nozzle weld. Thus, even if torque was transmitted from the core bore machine to an intact in-core nozzle, failure would not occur. If the in-core nozzle weld experienced significant damage, the in-core nozzle above the weld would have melted to the reactor vessel lower head inside diameter and such a bridge was not possible.

- *Objects Dropped Through Bore Hole*. After completion of the guide tube boring, the severed guide tubes would be removed from the reactor vessel. The removal of guide tubes would provide a 6.5-inch-diameter pathway to the lower head for load drops. This potential load drop pathway was qualitatively addressed. Video inspections of the lower head during the past years allowed visual observation of eight in-core nozzles. Five of these eight nozzles were below one of the 15 in-core guide tubes that would be bored. The video inspections revealed that seven in-core nozzles were undamaged, with the other nozzle slightly melted at the upper end. Therefore, the evaluation concluded that the in-core nozzle-to-reactor vessel welds of these eight nozzles were undamaged.

  Given that these eight nozzles with intact vessel welds were near the other in-core nozzles that were under the 15 in-core guide tubes to be severed, the evaluation concluded that the other in-core nozzles were unlikely to be significantly damaged. Consequently, should a dropped object pass through one of the 6.5-inch-diameter holes, the object would most likely impact a strong, relatively undamaged nozzle and would not result in any in-core nozzle-to-reactor vessel weld failure.

- *Support Post Drop*. Posts support the rib section to the grid forging. Most of the support posts, when boring was completed, would have a cruciform piece of the lower grid rib section attached to their upper end. When the boring was completed, these support posts would not drop any distance since the lower grid forging would not be bored. The 16 outer periphery support posts would have a cruciform piece of lower grid rib section attached to its upper end and a piece of the forging attached to its lower end. The outside diameter of this piece would be 5.25 inches and would weigh about 120 pounds. When the boring was completed, the piece would drop out of the drill bit and fall onto the in-core guide support plate, a distance of 0.5 inch. The total energy imparted to the in-core guide support plate would be 126 pounds. If the support post was filled with core rubble, the maximum impact to the in-core guide support plate would be about 175 pounds. These forces would not damage the in-core guide support plate. Consequently, the boring of the support posts would be completely isolated from the reactor vessel lower head and the in-core nozzles.

- *Conclusion*. The above discussion demonstrated that loads imparted to an in-core nozzle weld would remain below the minimum loads necessary to cause a failure of the weld. Further, operating procedures would limit weight on the drill bit to 9000 pounds.

- ***NRC Review: Reactor Vessel Integrity***. [70] The NRC's safety evaluation stated that safety issues associated with mechanical forces affecting the reactor vessel and internals did not significantly differ from those previously reviewed and approved. In addition, the drilling operation would not cause significant risk of a failure of the in-core instrument penetrations, and the licensee had the capability to detect and mitigate a failure.

- *Evaluation*. The NRC previously evaluated [71] the potential for transmission of forces from the drilling rig to both intact and damaged in-core instrument penetrations through the instrument string. The NRC found that neither additional damage to the in-core instrument penetrations nor unacceptable leakage would occur. In the revised scope of activities, the forces from the drilling rig could be transmitted to the in-core instrument penetrations through the cut sections of in-core guide tubes. Available evidence from video inspection demonstrated that the in-core instrument penetrations were intact; however, the possibility existed that penetrations outside of camera range sustained damage during the 1979 accident. Although very unlikely, it was possible that drill forces transmitted through a guide tube could bend or break an in-core instrument penetration. However, this mechanism would result in only a small annular gap leakage pathway between the instrument string and the reactor vessel lower head. The NRC had previously analyzed [72] the results of the potential annular gap leakage and found it to be 0.4 gallons per minute. This was well within the capability to make up leakage that exceeded 100 gallons per minute. In addition, the licensee installed an alarming water level instrument and would take hourly water level readings while drilling to detect any potential leakage.

- *Operating Limitations*. The NRC also evaluated the potential for the drill bit to reach the reactor vessel lower head and cause a leak. The NRC approval letter required that the operating procedures incorporate the following limitations: (•) the elevation of the drill unit chuck would be determined as independently verified within a tolerance of 1 inch; (•) the length of each section of drill string would be measured, independently verified to be within 0.125 inch, and marked on the drill string; (•) full thread engagement at assembly joints would be observed and independently verified within 0.125 inch; and (•) the top casing section (stop casing) would be independently verified to be selected for the appropriate drilling location before drilling in each radius. These limitations, when coupled with the welded collar on the top casing section, precluded the drill bit from reaching the in-core instrument penetrations and provided a margin in excess of 6 inches to the reactor vessel lower head.

- *Conclusion*. The NRC concluded that the drilling operation would not cause significant risk of a failure of the in-core instrument penetrations and that the licensee had the capability to detect and mitigate a failure if such a leak occurred. The NRC approval letter incorporated provisions to prevent the drill bit from contacting the reactor vessel lower head. The NRC, therefore, concluded that the core bore machine could be used in dismantling the LCSA without significant risk to the health and safety of the public.

8.5.9 Sediment Transfer and Processing Operations (NA)

8.5.10 Pressurizer Spray Line Defueling System

- ***Purpose***. To flush fuel fines and core debris from the pressurizer spray line to the pressurizer vessel and the reactor coolant system cold-leg loop 2A. The source of flush water for the pressurizer spray line defueling system (PSLDS) was the defueling water cleanup system. Defueling consisted of flushing the pressurizer spray line in a series of steps to adequately remove fuel fines and debris in each different flowpath from the spray line tie-in.

- ***Evaluation: Reactor Coolant System Integrity***. [73] The licensee's safety evaluation stated that the principal consequence of any hose or line break was loss of reactor vessel inventory. The PSLDS was designed to mitigate the consequences of such an accident to the extent possible. In case of a hose or line rupture on the PSLDS, the defueling water cleanup system would trip the reactor vessel pumps if there was a low level in the internals indexing fixture. In addition, the low level would trigger an alarm at control panels located in the main control room and the fuel handling building. This type of accident could deliver about 500 to 1000 gallons of reactor vessel water to the area of the rupture. The affected area would be within the containment, which had a sump to contain the spill. The recovery from this event would be accomplished by isolating the ruptured section and replacing the ruptured hose or pipe.

- ***NRC Review***. [74] The NRC's safety evaluation did not specifically address this topic.

8.5.11 Decontamination Using Ultrahigh Pressure Water Flush

- ***Purpose***. To use ultrahigh pressure (UHP) water flush at 20,000 to 55,000 pounds per square inch (psi) to remove surface coatings and surface contamination inside the containment building.

- ***Evaluation: Reactor Coolant System Integrity***. [75] The licensee's safety evaluation noted that the effects of a UHP water jet, which ranged from 0.005 to 0.025 inch in diameter at pressures of 20,000 to 55,000 psi at the nozzle, included the following:

  - *Metals*. UHP jets that were emitted from a rotating or traversing nozzle would not damage metals, including steels, alloys, and cast iron. No deterioration of surface finish or removal of metal would occur, even with repeated passes of the jets. Stationary (fixed) jets aimed at one spot could cause minor surface deterioration or dimpling of surfaces if allowed to dwell at close standoff distances (under 1.0 inch) for extended periods of time (greater than 5 minutes).

  - *Anodized Surfaces*. UHP jets could damage or completely remove anodized surfaces from aluminum and galvanized surfaces from steel.

  - *Soft Metals and Coatings*. UHP jets could damage surface finishes or remove material from soft metals including lead, pure copper, and soft aluminum if care was not taken. Paint and similar coatings could be removed without damaging these surfaces provided that a rapidly traversing and rotating nozzle was used with nozzle/material relative velocities exceeding 75–100 inches per second.

  - *Conclusion*. Evaluations performed in support of the safety evaluation report [76] for heavy load handling inside containment identified no pipe or tubing breaks that could result in the draining of the reactor vessel below the bottom of the hot leg, with the exception of an in-core instrument nozzle or guide pipe break outside the vessel. The in-core guide pipe was 0.5-inch-thick stainless-steel pipe. It was not considered credible that the UHP water jet would induce a failure of the guide pipe. However, administrative controls, physical barriers, or both would ensure that this area was avoided and protected.

- ***NRC Review: Reactor Coolant System Integrity***. (77) The NRC's safety evaluation considered the potential effects on important safety components for impingement by the UHP water jet. The water jet diameter ranged in size from 0.005 to 0.025 inch at pressures of 20,000 to 55,000 psi. Harder metals, such as steels, alloys, and cast iron, would not be damaged by rotating or traversing jets. Prolonged impingement of the jet on a hard surface could cause some surface deterioration. Although there was only a small potential for damage from the jet during decontamination operations, the licensee would evaluate each area to be documented and ensure that vital components were either avoided or protected.

The operations would be administratively controlled to prevent the jet from dwelling for an extended time at any point. In addition, other evaluations showed that there were no reactor coolant system piping systems in which a failure induced by the UHP jet could cause draining of the reactor vessel below the level of the hot-leg nozzles, except for the in-core instrument piping. Procedural controls or physical barriers would ensure that the areas around the in-core instrument piping would be avoided or protected. In addition, a safety evaluation performed in support of an unrelated technical specification change (78) demonstrated that the reactor water level could be safely maintained in the event of an in-core instrument pipe break.

The NRC concluded that the licensee's proposed program did not present the potential for damage to components from water impingement that could result in any undue risk to the health and safety of the public.

8.6 Evaluations for Defueling Operations

8.6.1 Preliminary Defueling

- ***Purpose***. To allow movement of debris within the reactor vessel, to allow installation of defueling equipment, and to identify core debris samples for later removal and analysis by INEL. The movement of debris within the vessel was prepared for early defueling and required some rearrangement of debris in the reactor vessel before the actual removal of fuel. These activities included the loading of small pieces of core debris into debris baskets but did not include actual loading of fuel debris into fuel canisters.

- ***Evaluation: Reactor Vessel Integrity***. (79) Editor's Note: The licensee's safety evaluation report did not specifically address the reactor coolant system and reactor vessel integrity; however, its evaluation did consider the effects of a load drop over the reactor vessel.

- ***NRC Review: Reactor Vessel Integrity***. [80] The NRC's safety evaluation stated that its previous safety evaluation report [81] for heavy load handling over the reactor vessel was applicable to the proposed preliminary defueling activities that were restricted to the reactor vessel. This previous evaluation determined that the worst case accident resulting from a postulated heavy load drop over the reactor vessel would be the simultaneous failure of the 52 in-core instrumentation tubes resulting in a total leakage rate of 20 gallons per minute. In an unrelated safety evaluation [82] for a technical specification change, the NRC concluded that reliable sources of borated makeup water would be available via gravity feed from the borated water storage tank. This water source and the operation of the containment building sump recirculation system would substantially exceed the worst case reactor coolant system leakage rate of 20 gallons per minute.

8.6.2 Early Defueling

- ***Purpose***. To remove end fittings, structural materials, and related loose debris that would not involve removal of significant amounts of fuel material, to remove intact segments of fuel rods and other pieces of core debris, and to remove loose fuel fines (particles) by vacuuming operations.

- ***Evaluation: Load Drop/Reactor Vessel Integrity***. [83] Editor's Note: The licensee's safety evaluation for early defueling was practically identical to its subsequent safety evaluation report (SER) [84, 85] for bulk defueling; therefore, this section does not include the evaluation text. Please refer to the next section for details.

- ***NRC Review: Load Drop/Reactor Vessel Integrity***. [86] The NRC's safety evaluation stated that the handling of heavy loads during early defueling activities was addressed by the licensee's SER [87] for handling of heavy loads over the reactor vessel and the SER [88] for handling of heavy loads inside containment. In the NRC's SER for heavy load handling over the reactor vessel, the worst case accident identified for all loads anticipated to be lifted over the vessel through completion of defueling was the postulated drop of the plenum assembly. For this bounding case, the licensee postulated that a drop of the plenum assembly would cause the simultaneous failure of all 52 in-core instrumentation tubes, resulting in a total reactor coolant system leakage rate of 20 gallons per minute. As described in the NRC's SER [89] for preliminary defueling activities, the agency concluded that reliable sources of borated makeup water would be available to substantially exceed the worst case reactor coolant system leakage rate and that adequate leak detection capability was provided.

8.6.3 Storage of Upper End Fittings in an Array of 55-Gallon Drums (NA)

8.6.4 Defueling (Also Known as "Bulk" Defueling)

- ***Purpose***. To develop defueling tools and to conduct activities necessary to remove the remaining fuel and structural debris located in the original core volume. An additional activity was to address vacuuming of debris from the lower head of the vessel, a primary activity associated with lower head defueling.

- ***Evaluation: Reactor Vessel Integrity***. [90, 91] The licensee's safety evaluation stated that the only postulated failure mechanism that could result in the draining of the reactor vessel water was damage to the in-core instrument tubes or nozzle welds. Damage to an in-core instrument tube outside the vessel could occur by pulling on an in-core instrument string if the instrument string could impart excessive loading to the instrument tube. An in-core instrument tube nozzle could be damaged by directly or indirectly imparting a load to the nozzle. Failure of either a nozzle weld or instrument tube could result in nonisolatable reactor coolant system leakage.

During defueling activities, structurally intact in-core instrument strings were not expected to be found in the loose rubble; however, strings could be found in other regions of the vessel. The effects of applying limited loads to the in-core instrument strings within the reactor vessel were evaluated.

- *Lower Head Integrity (Melt)*. The safety evaluated cited the Babcock & Wilcox report, "Evaluation of the Structural Integrity of the TMI-2 Reactor Vessel Lower Head, Final Report," dated June 1985, that reported the results of a thermal-hydraulic and structural analysis of the reactor vessel lower head during the accident. The results of the thermal analysis were not factored into the structural analysis, and consequently, they were relatively independent of each other. However, the thermal analysis showed that the upper part of the in-core instrument nozzle did reach temperatures much greater than its melting point because the nozzle was hot material and had no cooling except for conduction down the nozzle and into the lower head. The approximately 0.5-inch length of the nozzle above the surface of the lower head provided sufficient cooling to keep this welded portion of the nozzle from melting. The 1985 evaluation further showed that, although a significant portion of the lower head could have been 1600 degrees Fahrenheit (degrees F) or hotter, temperatures significantly higher were unlikely because the ultimate strength of the base material at these elevated temperatures was insufficient to prevent head failure. Gross head failure did not occur since the vessel had demonstrated significant pressure-retaining capability. Therefore, the evaluation concluded that melting and distortion of any significance in the lower head were highly unlikely.

- *In-core Instrument Strings*. If the material in the lower head were sufficient to have melted the upper portion of the in-core nozzle, logic would indicate that the in-core instrument string had also melted. Electrical resistance data confirmed that many of the in-core thermocouples in the center of the core terminated at or near the reactor vessel wall. It would, therefore, appear highly unlikely that in-core instrument strings remained above the core support structure lower grid in the center 50 percent of the core cross section.

- *In-core Nozzle Strength*. The 1985 evaluation also developed a structural loading criterion based on a calculation of the minimum wall thickness in various segments of the lower head needed to support the pressures experienced in the reactor coolant system during the TMI-2 accident. At a metal temperature of 1600 degrees F, a 0.030-inch-thick in-core nozzle-to-vessel weld would support the pressure of 2000 pounds per square inch experienced during the accident. Note that this weld, to support pressure, was required only to bridge an axial gap between the nozzle outside diameter and the vessel wall bole inside diameter of 0.005 inch. Little metal was required in such a small annulus to support the pressure.

A significant conservatism in the analysis was the choice of 1600 degrees F as the maximum weld temperature considered in the structural portion of the report. This temperature was chosen because it represented the highest temperature for which approved material ultimate strength data existed. This temperature was about 1000 degrees F below that required to melt Inconel 600 (i.e., 2540 degrees F). As an example of this conservatism, Inconel at 1900 degrees F would be expected to exhibit an ultimate strength of about 7000 pounds per square inch. Therefore, if the weld reached 1900 degrees F, the thickness would have to be about 0.058 inch.

Assuming a conservative temperature of 1600 degrees F, the following loads are those that the 0.030-inch-thick weld could not sustain at room temperature as reported in the 1985 evaluation: (●) axial force at 5400 pounds; (●) bending movement at 1400 inch-pounds; and (●) twisting torque at 5800 inch-pounds. Higher loads could be calculated for a greater thickness (i.e., 0.058 inch at 1900 degrees F).

- *Axial Tension Loads*. A review of the potential loading methods that could transmit loads from the in-core instrument string to the in-core nozzle concluded that only axial tension loads could be transmitted from the in-core instrument strings above the core support assembly to the in-core nozzles below the flow distributor. Based on this limited potential, the licensee’s material laboratory in Reading, Pennsylvania, experimentally determined the tensile load needed to break a new in-core instrument string. The three tests that were performed resulted in in-core string breakage at loads of 3800 to 3950 pounds. Therefore, the strings should break before resulting in damage to the in-core instrument nozzle weld.

  The application of an upward axial load to an in-core instrument string within the reactor vessel could result in a compressive load to the inside radius of the in-core piping and cause a bending movement to the pipe. The potential loads were determined to be low compared to loads in the original design. To minimize the potential damage to the nozzle welds, the vacuum equipment used in the lower head was designed to minimize the force that could be imparted to the guide tubes or nozzles.

- *Conclusion*. The licensee did not expect that in-core instrument tubes or nozzle welds would be damaged during defueling, such that a reactor coolant system leak that could not be isolated would occur. A licensee’s previous safety evaluation report (SER) [92] for heavy loads over the reactor vessel demonstrated that sufficient makeup water would be available following the simultaneous shearing off of all 52 in-core nozzles at the inside vessel wall. This evaluation concluded that the core debris would remain covered with borated water. As it was considered extremely unlikely that defueling operations could damage all 52 nozzles and tubes or both, the previous analysis provided a conservative limit on any leakage from in-core instrument tubes that could be experienced during defueling.

- ***NRC Review: Reactor Vessel Integrity***. (93) The NRC's safety evaluation stated that the use of the core boring equipment for core region defueling created a potential for imparting loads to the in-core instrumentation nozzles that could result in weld failure and a leak of reactor coolant that could not be isolated. In previous approvals for the use of this equipment for in-core sampling applications, the NRC imposed appropriate limitations to minimize the potential for failure of the in-core nozzle welds.

The licensee stated in its SER that a solid face drill bit would be used with the core boring equipment for bulk defueling. Such use was clarified in a previous SER (94) for the use of core stratification sample acquisition tool for defueling, which was submitted after the SER for bulk defueling, which detailed the technique and restrictions to be applied during core boring. The general protocol for core boring was previously addressed in the licensee's SER (95) for core stratification sample acquisition. The newer safety evaluation for core boring restricted core drilling operations to noninstrumented fuel assemblies at a depth above the lower grid support structure.

The NRC stated that core bore defueling activities could be safely conducted in the core region, subject to the following restrictions: (●) Maximum allowable drill depth would be limited to the top of the lower grid support structure. Such depth would be sufficient for core region defueling and would ensure that sufficient loads would not be imparted to the in-core nozzle and welds. (●) Drilling locations would be procedurally controlled to prevent the direct application of the drill to existing in-core instrument strings, unless that string and supporting guide tube were verified to have been cut. Such location control would rely on the theodolite system (for location measurement), as described in the newer evaluation of core boring, unless the NRC approved an alternative method.

The NRC's safety evaluation concluded that these precautions would reduce the potential for a leak that could not be isolated because of the failure of an in-core weld resulting from the proposed core bore defueling activities. In the unlikely event of such leakage, the licensee provided equipment and developed procedures to quickly identify the leak and establish subcriticality of the core.

8.6.5 Use of Core Bore Machine for Bulk Defueling

- ***Purpose***. To use the core stratification sample acquisition (core bore) tooling as a defueling tool so that other defueling tools could more effectively break up and remove the remaining core debris. The core bore tool used a solid-faced bit to perforate the hard crust region of the core, down to the lower grid support structure, at multiple locations. The defueling work platform orientation system was used to position the drill mechanism with restrictions.

- ***Evaluation: Reactor Vessel Integrity***. (96) The licensee's safety evaluation stated that the operation in the proposed mode was bounded by its previous safety evaluation report (97) for core stratification sample acquisition (core bore) except that drilling sites would be identified using the defueling work platform locating system as opposed to the theodolite system. The theodolite system would be used to establish boundaries for the initial drilling zone. Drilling operations for this initial phase would be restricted to the established zone that would not include locations containing in-core strings. Individual drilling sites within this zone would be found using the defueling platform locating system. Additional drilling zones would be established by using either the theodolite system or the work platform locating system, depending on the experience gained by drilling in the first zone. During this initial phase, it was possible to encounter an in-core string displaced from an adjoining element. However, the earlier safety evaluation for core boring concluded that no damage to an in-core string was credible since the drill had no mechanism to catch the in-core string and impart a load.

The proposed operation for bulk defueling would use the same mechanical limits imposed for core boring that was above the lower grid support structure during the earlier core stratification sample acquisition activity. Drilling at locations containing in-core instrument strings was being evaluated. Future correspondence would provide necessary technical information and seek appropriate NRC approvals for this activity.

- ***NRC Review***. (98) Editor's Note: The NRC documented its review of the licensee's proposal in the agency's safety evaluation report (99) for bulk defueling.

8.6.6 Lower Core Support Assembly Defueling

- ***Purpose***. To dismantle and defuel the lower core support assembly (LCSA) and to partially defuel the lower reactor vessel head. Structural material removed from the LCSA included the: (●) lower grid rib assembly; (●) lower grid forging; (●) distribution plate; and (●) in-core guide support plate. The lowest elliptical flow distributor and the gusseted in-core guide tubes would not be removed under this proposed activity. The use of the core bore machine, plasma arc cutting equipment, cavitating water jet, robot manipulators, automatic cutting equipment system, and other previously approved tools and equipment was included in this activity and associated safety evaluations.

- ***Evaluation: Reactor Vessel Integrity***. (100) The licensee's safety evaluation addressed two concerns: damage to the reactor vessel wall and damage to the in-core instrumentation nozzles.

  - *Reactor Vessel Wall*. The evaluation concluded that damage to the reactor vessel wall from the operation of burning devices inside the vessel would be precluded. The operation of such devices was physically limited to inside the confines of the core support structure and the elliptical flow distributor, where the torch was more than 1 foot away from the reactor vessel wall. Cutting operations would begin on the top of the LCSA and would sequentially cut through the lower grid rib assembly, lower grid flow distributor, lower grid forging, and in-core guide support plate. Therefore, the arc or flame of such burning devices, operating underwater, would always be operated at least 1 foot from the reactor vessel wall. Propagation of an arc through 1 foot of water was not possible; therefore, damage to the reactor vessel wall due to the operation of burning devices was precluded.

- *In-core Nozzles*. The only other major concern associated with reactor vessel integrity was the integrity of the in-core nozzles. Previous licensee and NRC correspondence related to the SERs [101, 102] for core bore operations established two possible in-core nozzle configurations as a result of the 1979 accident. In the worst case, the damage to the reactor vessel lower head would consist of an in-core nozzle melted to the inside diameter of the reactor vessel lower head, with a nozzle-to-vessel weld thickness of only 0.030 inch. The significance of this configuration was that if the weld experienced significant damage, the in-core nozzle above the weld would have melted. The other possible configuration was that the in-core nozzle was undamaged.

  Degradation of the in-core nozzles during the accident was determined to be highly unlikely. This was especially true of those in-core nozzles located under the peripheral nongusseted in-core guide tubes. The licensee's and the NRC's safety evaluation reports [103, 104] for the use of the core bore machine for dismantling the LCSA concluded that in-core nozzles in this area were unlikely to be degraded. However, because of the inability to visually inspect all the in-core nozzles, the evaluation assumed that on a worst case basis, the in-core nozzles under at least some of the gusseted in-core guide tubes experienced some degree of damage. Therefore, the safety evaluation concluded that care must be exercised when defueling the LCSA.

  Tools that could potentially impart excessive loads to the in-core instrument tube nozzles or damage the reactor vessel wall would be limited to use within the confines of the core support structure and the elliptical flow distributor. This limit would remain until most of the fuel within the lower LCSA was removed, after which procedural limitations would be applied. Mechanical cutting devices, such as the abrasive saw, grinding wheel, and impact hammer, were not of sufficient size or power to damage the reactor vessel wall; therefore, these devices did not create a safety issue.

  During the removal of fuel debris from the lower head, care would be taken to prevent excessive loads on exposed in-core nozzles. During the fuel removal process in the vicinity of an in-core nozzle, if observations indicated that the nozzle had suffered damage due to excessive temperatures, then work would be halted, and the situation evaluated to ensure that activities could continue within the bounds of this safety evaluation report.

- *Other Integrity Concerns*. Other reactor vessel integrity safety concerns (e.g., assessment of potential damage to in-core nozzles from pulling on in-core instrument strings) were bounded by the safety evaluation for core bore operations.

- ***NRC Review: Reactor Vessel Integrity***. [105] The NRC's safety evaluation stated that removal of any gusseted in-core guide tubes and the elliptical flow distributor was not included in the scope of the licensee's safety evaluation. These two structures formed part of the protection for heavy load drops inside and over the reactor vessel during LCSA defueling.

8.6.7 Completion of Lower Core Support Assembly and Lower Head Defueling

- ***Purpose***. To remove the elliptical flow distributor and gusseted in-core guide tubes of the lower core support assembly (LCSA) and to defuel the reactor vessel lower head (RVLH).

- ***Evaluation: Reactor Vessel Integrity (Nozzles)***. [106] The licensee's safety evaluation demonstrated that LCSA/RVLH defueling would have a low probability of impairing the integrity of the reactor vessel. This conclusion was based on results of previous evaluations and observations. The safety evaluation was comprehensive and is presented in its entirety below. Because the evaluation was lengthy, section numbers from the safety evaluation report (SER) are also provided for ease of reference.

  - *Introduction*. The safety evaluation (SER Section 4.11.1) of reactor vessel in-core nozzle integrity was based in part on two previous thermal-hydraulic evaluations: the Babcock & Wilcox report, "Evaluation of the Structural Integrity of the TMI-2 Reactor Vessel Lower Head, Final Report," dated June 1985 and the INEL report, [107,] "TMI-2 Reactor Vessel Lower Head Heatup Calculations" (EGG-TMI-7784). These studies demonstrated that the temperature required for deformation of the RVLH was well below the melting temperature for the in-core nozzle welds. Since no evidence of reactor vessel leakage was observed during or after the accident, reactor vessel failure did not occur, and consequently, the inside surface of the lower head clad material did not reach melting. Therefore, the licensee concluded that there was a high probability that in-core nozzle welds maintained their original integrity. The safety evaluations (documented in SER Sections 4.11.2 through 4.11.4) demonstrated that no information contradicted these conclusions of the earlier evaluations based on: (●) visual examination of in-core nozzles (discussed in SER Section 4.11.2), (●) verification of nozzles based on measured thermocouple junction cable lengths (discussed in SER Section 4.11.3), and (●) verification of nozzle integrity based on stub assembly removal (discussed in SER Section 4.11.4). Section 4.11.5 summarized the detailed information in SER Sections 4.11.1 through 4.11.4.

    The safety evaluation noted that none of the evaluations discussed in SER Sections 4.11.1 through 4.11.5, taken separately, completely confirmed nozzle integrity because of the inaccessibility of some of the nozzles. However, when taken together, the absence of negative findings and the variety of positive indications provided sufficient evidence to support a conclusion that nozzle welds were undamaged. Therefore, the evaluation concluded that lower reactor vessel head integrity was sufficient to withstand worst case bounded heavy load drops.

  - *Summary (SER Section 4.11.5)*. The evaluation of nozzle integrity (SER Sections 4.11.1 through 4.11.4) was summarized in this section. Based on this information, the licensee concluded that it was highly unlikely that the welds from the in-core nozzles to the reactor vessel were significantly degraded during the accident. Therefore, the evaluation concluded that LCSA/RVLH defueling could be conducted without impairing the integrity of the reactor vessel. This conclusion was based on the following observations:

- Several conservative thermal analyses of the RVLH response to the melted corium, both flowing and stationary, indicated little or no melting of the vessel wall would be expected (refer to SER Section 4.11.1).

- Stress analysis of the vessel head indicated that failure due to creep would occur before temperatures sufficient to melt in-core nozzle welds were achieved. Because no leakage from the vessel was observed, vessel head creep did not occur; therefore, the surface temperature of the vessel was well below melting (refer to SER Section 4.11.1).

- Gamma scan data implied that a nonfueled layer of resolidified material could have existed on the lower head inside surface; this layer would have protected the lower reactor vessel nozzle welds from melting (refer to SER Section 4.11.1).

- Current visual examinations of in-core nozzle welds at the outer periphery of the core revealed no weld damage (refer to SER Section 4.11.2).

- In-core instrument thermocouple resistance data indicated that there were 41 measurable thermocouple junctions at or above the RVLH. Visual inspections during fuel assembly stub removal indicated a high degree of correlation between in-core string length and thermocouple junction reformation measurements (refer to SER Section 4.11.3).

- Visual inspections during stub assembly removal also indicated that detectors for the 11 locations with open thermocouple junctions separated above the lower core support structure (refer to SER Section 4.11.4).

- Other reactor vessel integrity safety concerns (e.g., assessment of potential damage to in-core nozzles from pulling on in-core instrument strings) were bounded by the licensee's and the NRC's SERs [108, 109] for the use of core bore operations to facilitate defueling.

o *Precautions during LCSA/RVLH Defueling (SER Section 4.11.6)*. During the removal of fuel debris from the lower head, care would be taken to prevent excessive loading on exposed in-core nozzles. If during the process of removal of fuel in the vicinity of an in-core nozzle, observations indicated that a nozzle weld had suffered damage due to excessive temperatures, then work would be halted in the vicinity of that nozzle and the situation evaluated to determine if activities could continue within the scope of this SER.

o V*erification of In-core Nozzle Integrity Based on Thermal-Hydraulic Evaluation (SER Section 4.11.1)*. The safety evaluation concluded that the molten corium did not melt the lower head, even if a molten jet flowed into the lower head in only one location. Given that the RVLH did not melt, the licensee concluded that in-core nozzle welds maintained their integrity. This conclusion was based on the evaluation results discussed below.

- *Temperatures Less than 1600 degrees Fahrenheit (degrees F) Based on No Leakage*. As discussed in the Babcock & Wilcox report, increasing metal temperatures would decrease metal strength. At 1600 degrees F, the ultimate strength of the RVLH carbon steel material was only 13,400 pounds per square inch (psi). At an internal pressure of 2200 psi and a metal average temperature of 1600 degrees F, the 5-inch-thick lower head would either rupture or experience significant plastic deformation. Pressure transients to about 2200 psi were experienced in the reactor vessel after molten core material relocated into the lower head. Since the RVLH did not experience postaccident leakage, the evaluation concluded that the RVLH carbon steel shell did not creep; therefore, the shell did not attain average temperatures of 1600 degrees F.

- *Heat Conduction of the Welded Portion of the Nozzle*. The conservative thermal-hydraulic evaluation in the Babcock & Wilcox report demonstrated that if the temperature of a significant portion of the lower head reached or exceeded temperatures of 1600 degrees F, the in-core nozzle welds would not have reached their melting temperature of 2760 degrees F. The previous evaluation also noted that the upper part of the in-core instrument nozzle could reach temperatures well above its melting point if surrounded by hot corium and with no cooling, except for conduction down the nozzle and into the lower head. However, the half-inch or so of the nozzle above the inner surface of the lower head was close enough to the head that conduction of heat into the head provided sufficient cooling to keep this welded portion of the nozzle from melting.

- *Vessel Thermal Response with Lower Plenum Degraded Core Material*. The results from the INEL report predicted the temperature history of selected locations for the RVLH across six cases of core rubble material compositions. These cases assumed different thermal response characteristics of solid fuel material, fuel debris (fuel and clad), and control rod material in the lower head. The conclusions from the INEL report related to the thermal analysis of the lower head were repeated in the SER as follows:

    - The thermal response of the TMI-2 lower reactor vessel has been analyzed for three assumed lower plenum degraded core material configurations, (i.e., (a) a porous debris bed resting on the vessel head, (b) a debris bed resting on top of about 8 inches of consolidated molten fuel adjacent to the vessel, and (c) a porous debris bed resting on top of about 8 inches of assumed control rod material adjacent to the vessel). For each configuration, the vessel thermal response was calculated assuming the debris was both coolable and non-coolable.

    - The calculations show a wide range of vessel thermal response is possible based on the debris configuration and debris cooling assumptions. Vessel melting temperatures were predicted for two of the cases (Cases 3 and 4); however, for the relatively short transient (5400 seconds) very little melting was predicted. The most rapid heatup (resulting in the highest vessel wall temperatures) occurred for the case with assumed consolidated fuel adjacent to the vessel wall. For this case, temperatures in excess of 1100°K (1521 degrees F) were achieved in less than 20 minutes and these temperatures are expected to have resulted in creep rupture during the first hour after the major core relocation. Cooling of the porous debris resting on top of the consolidated molten material had little effect on the maximum vessel temperatures for this case.

- The calculations show for a porous debris bed that vessel wall temperatures would have been sufficiently low that creep rupture of the vessel would not be expected. In addition, a layer of control rod material adjacent to the vessel wall does provide an effective insulation to the wall at locations away from the wall/fuel debris interface.

Although the above conclusions showed that melting was predicted for the two worst-case assumptions vessel, the INEL report also clearly showed that vessel rupture due to creep would occur at vessel average wall temperatures well below melting. As an example, at an average wall temperature of 1050 degrees Kelvin (1430 degrees F) and a pressure of 10.0 megapascals (1450 pounds per square inch absolute pressure), vessel rupture due to creep would occur in about 10 minutes. The analysis in the Babcock & Wilcox report predicted a maximum temperature difference between inside vessel surface temperature and average wall temperature of 435 degrees F. Assuming this analysis to be correct, vessel rupture due to creep failure should have occurred at RVLH surface temperatures of less than 2000 degrees F, which was 650 degrees F less than clad melting temperature. The licensee concluded that since no evidence of reactor vessel leakage had been observed during or after the accident, reactor vessel failure did not occur, and consequently, the inside surface of the lower head clad material did not reach melting.

– *Insulation by Nonfuel Debris on Lower Head*. The licensee’s technical planning report TPO/TMI-175, “Analysis of Gamma Scanning of Incore Detector No. L-11 in Lower Reactor Vessel Head,” Rev. 5, dated June 1985, reported the possible presence of a nonfuel layer of material in the RVLH. This layer was expected to be about 9 inches high at the centerline of the reactor vessel. The material could be either control rod material (silver, indium, and cadmium) or control rod material mixed with stainless steel. The presence of this low melting point material would have protected the lower reactor vessel welds from melting from hot corium located above the layer of melted high-density material. This was also stated in the conclusion of the INEL report.

– *Vessel Thermal Response from Melt Jet Impingement*. INEL conducted a study [110] of the thermal response of the lower head when it would come in contact with a jet of falling melted core debris in a period of 75 seconds (depicted by nuclear instrumentation response during the TMI-2 accident). The conclusions from the INEL report that were related to the lower head analysis were repeated in the SER as follows:

- Thermal damage potential to the lower head was also assessed for the configuration of coherent jet impingement of relocating melt debris. For this assumed configuration, the thermal response of the lower head is largely dictated by the contact time and heat transfer characteristics at the jet impingement surface. Assuming a jet diameter equal to the flow area within a single undegraded fuel assembly, the time for melt relocation as a jet is estimated to be about 75 seconds. This estimate is consistent with source range monitor data, indicating that major core relocation occurred over a 1-minute period.

- Two limiting conditions were assumed with respect to jet impingement heat transfer characteristics. The first was for a weak jet with conduction-limited heat transfer. The second was for strong jet forces where turbulent mixing and mass transfer effects at the impact surface led to enhanced convection-controlled heat transfer process. For conduction-controlled heat transfer, surface ablation of the lower head by direct impingement is not inferred. This is due to the rather poor conductivity of the molten ceramic material and the high thermal capacity of the vessel head, which serves as an efficient and quick-response heat sink. However, calculational results for convection-controlled heat transfer indicate limited melt ablation at the liner surface. The calculated depth of penetration of the melt front is on the order of about 0.5 inch (versus a head thickness of 5.5 inches) for a jet impingement time on the order of 75 seconds. A direct jet impingement time of about 15–20 minutes is, however, calculated to be necessary for melt ablation of the vessel head 0.5 inch in thickness. It is, therefore, concluded that for a jet impingement time of 1 to 2 minutes (time associated with melt drainage to the lower plenum), little thermal damage to the lower vessel head would result.

– *Melt Jet Impingement Based on Visual Inspections*. Visual examinations in the reactor vessel and lower head indicated that the possibility of continuous jet impingement and consequent damage to an in-core nozzle was extremely remote, based on the following observations:

  - Each of the 175 fuel stub assembles removed from the reactor vessel had some intact zirconium fuel tubes still attached and had a semi-intact lower end fitting. Only at in-core locations R-6 and R-7 was it evident that significant amounts of molten corium could have passed through the space to the lower internals. The balance of the major core relocation to the lower internal area was most likely outside the nozzle area.

  - The large hole in the core baffle above grid positions R-6/R-7 provided evidence that the suspected corium flowpath was to the lower head through the core formers and was thus outside the area of most of the active core.

  - The in-core nozzle in position R-7, assembly No. 45, was observed to be standing but damaged at its upper end. This nozzle location, along with nozzle locations P-6, O-5, N-4, and M-3, was believed to have been in the flowpath of molten corium from the large hole in the baffle plates on the east side of the reactor vessel to the lower head. Resolidified material in R-6 and R-7 tends to confirm this flowpath. Therefore, based on the analysis in this section and Section 4.11.2, the licensee believed that these nozzle welds were intact.

- *Visual Examination of In-core Nozzles (SER Section 4.11.2)*. The licensee received NRC approval [111, 112] to disassemble and remove 15 of 52 in-core instrument guide tubes. This approval was based on the video inspection of in-core nozzles that revealed no observable damage of the in-core nozzle-to-reactor-vessel weld. This was based on visual examination of 12 nozzles.

  The licensee's previous SERs [113, 114] for use of the core bore for defueling operations concluded that significant in-core nozzle weld damage would be highly unlikely if the nozzle existed above the vessel wall. In the video examination of the lower head region, 12 nozzles or portions of nozzles were visually examined. These included nozzles at locations K-12, H-13, G-13, F-13, F-12, R-7, R-10, O-12, M-14, L-13, D-14, and C-13. Other than the limited damage to the tip of R-7, all other visible portions of these nozzles were undamaged. For five of these nozzles, portions of the weld were seen; all of these were also intact.

  In the same video inspections, the in-core instrument guide tubes were observed at 39 locations (including 12 where the nozzle was seen). For all of these locations, the visible portion of the guide tube was undamaged, except for thermal damage to the D-10 and R-7 guide tubes. Furthermore, even for the locations that showed limited thermal damage, the licensee believed that nozzle welds were intact. The thermal-hydraulic analysis (Section 4.11.1) indicated that even if the nozzles were immersed in molten corium, the original weld integrity would be maintained. Accordingly, the licensee concluded that the video evidence in the lower head showed no indication that any of the nozzle welds had been damaged.

- *Evaluation of In-core Nozzle Integrity Based on Thermocouple Junction and Self-Powered Neutron Detector Test Data (SER Section 4.11.3)*. An INEL study [115] obtained accurate loop resistance measurements of the in-core thermocouples including the extension cabling. The INEL report also provided in situ self-powered neutron detector test data for the in-core assemblies. These data were used to evaluate in-core nozzle integrity.

  – *In-core Thermocouples*. The actual loop resistance of the in-core thermocouples and extension cabling following the accident could be determined with the data from the INEL report. By comparing the postinstallation data with the present data, changes in the length of the thermocouples as a result of the accident could be identified. The data showed that of the closed loops measured, all of these measured lengths of the thermocouple junctions were above or at the elevation of the cladding of the lower reactor vessel head. The thermocouple lengths [116] were measured from the tangent of the cladding of the RVLH. The licensee believed that if the thermocouple juncture was still operative and at a location above the inside of the vessel wall, it was highly unlikely that the entire in-core nozzle had melted to a location below the juncture. The INEL report stated that 11 of the 52 thermocouples exhibited an open circuit and no location of the juncture was observed.

The junction locations of the 41 measurable thermocouples ranged from location E-11, which indicated a junction at the clad wall elevation, to O-14 where the junction appeared to be 4.5 feet below its original location above the top of the core. Therefore, the information from the INEL report further supported the conclusion that 41 in-core guide tube nozzles had undamaged welds joining them to the RVLH.

– *Self-Powered Neutron Detectors*. The results of in situ self-powered neutron detector (SPND) test data for the in-core assemblies included the 11 in-core locations with open thermocouple junction data. For three locations (H-13, O-6, and L-13), the in situ test results indicated that there was at least one intact SPND in the lower level of the core. This implied that the in-core assembly was intact up to an elevation corresponding to the lower portion of the core. For seven other in-core grid locations (K-12, F-13, E-4, F-3, M-3, P-6, and O-10), all seven SPNDs were found to have an open circuit. Since an operating in-core detector exhibited an open circuit by design, the open-circuit conditions were indicative of less severe failure than shorted conditions. Therefore, these locations also represented intact in-core strings. The remaining in-core location (G-11) had both open and shorted SPNDs.

○ *Verification of In-core Nozzle Integrity Based on Stub Assembly Removal (SER Section 4.11.4)*. During the removal of stub assemblies, trailing in-core strings were observed in a number of locations. In other locations, extensions of the in-core string were observed above the lower grid. These observations by themselves were not conclusive except in identifying the weakest point in the in-core string. However, a correlation between the instrument string break location and a new thermocouple junction location would tend to confirm the analysis of thermocouple junction data. These possible correlation assumptions were that no other damage had occurred to the in-core string during handling and that thermocouple junctions reformed at a lower elevation when they came in contact with molten corium.

Video examinations of 51 in-core instrument locations after fuel assembly stub removal revealed that the in-core string separated from the removed stub assembly at 37 locations. In addition, these separations were observed to extend above the instrument guide tube in the lower grid. In 14 other locations, the in-core instrument string broke within the lower core support assembly or the lower head area and was extracted with the stub assembly.

The break location of the instrument string was not a certain indicator of the status of nozzle weld integrity. The instrument string break point could have resulted from other causes (e.g., mechanical damage due to defueling operations or thermal distortion). However, the evaluation noted that there was a high degree of correlation between the video observed break points and the thermocouple length reduction data that supported the conclusion that nozzle welds were not damaged.

The video examinations were of special importance with respect to the 11 locations where in situ resistance tests [117] showed the thermocouple to have an open junction where no thermocouple length reduction data existed. Since video data for all of these locations indicated that instrument string separation occurred within the core area, the evaluation concluded that in-core damage did not extend to the lower head. Furthermore, the in-core nozzles at 3 of the 11 locations that presented an open junction (i.e., H-13, F-13, L-13) were visually observed (refer to Section 4.11.2 of the SER), thereby providing further evidence that nozzle welds at these locations maintained their integrity.

- ***Evaluation: Reactor Vessel Integrity (Melt Jet Impingement Update).*** (118) During the NRC's review of the licensee's SER (119) for the completion of LCSA and lower head defueling, the agency identified an inconsistency in the RVLH thermal response as a result of contact with a jet of molten materials falling into the lower head during the TMI-2 accident. The NRC's letter requested additional analysis to which the licensee responded.

- *The NRC's Comment*. (120) The NRC's letter stated that the licensee's SER submittal included an evaluation of the lower head thermal response as a result of contact with a jet of molten materials falling into the lower head during the TMI-2 accident. The conclusions in the licensee's SER were based on the INEL report (121) EGG-TMI-7811, "Thermal Interaction of Core Melt Debris with the TMI-2 Baffle, Core Former, and Lower Head Structures." Pages 60–63 of that report presented an evaluation that assumed that heat transfer between the corium jet and the RVLH was controlled by a *convection* process. However, the heatup of the lower head assumed a *conductive* process with transient temperatures being obtained using the Biot number. (d) Such an approximation could be appropriate if the corium jet occurred in the center of the reactor vessel, since any melted lower head material and the corium jet would accumulate in this region of the vessel. Since the TMI-2 corium jet occurred in the outer ring of the core, where there was significant curvature of the lower head, melted lower head material would be expected to be carried with the jet and settle in the bottom of the vessel. Therefore, lower head material would be continuously exposed to the hot corium jet, and melt ablation of the lower head would be controlled by a *convective* heat transfer process rather than the *conductive* process assumed in the analysis. This would result in an increased melt ablation of the lower head.

  The NRC asked the licensee to provide a revised analysis of the interaction between the corium jet and the lower head that accounted for this effect. If a decreased lower head thickness resulted, then the NRC requested that the licensee review its effect on other portions of its SER and revise the report as appropriate.

- *Licensee Response*. (122) The apparent concern with the potential for a decreased RVLH thickness due to corium jet ablation was whether the potential effect would be of such magnitude as to cause a loss of structural integrity or the RVLH. No leakage of the RVLH occurred; therefore, the licensee believed that the only safety concern related to the NRC's comment was whether a load drop on the ablated area could result in a structural integrity loss of the RVLH.

[d] Editor's Note: Biot number is an index of the ratio of the heat transfer resistances inside of a body and at the surface of a body.

Visual examination of the TMI-2 core indicated that the corium flow was primarily through the baffle plates in the east-southeast edge of the core and down through the core former plates below the baffle plate break. Consequently, the licensee indicated that it agreed with the NRC's premise that the jet stream could have fallen onto the RVLH, where the curvature of the head was steep enough to cause the corium to stream and the ablated head material to theoretically wash away faster than if it had fallen on a flat plate. However, the significance of this finding relative to the overall concern for lower head integrity, namely the potential of accidentally dropped objects penetrating the head and causing significant leakage, did not invalidate the licensee's conclusions.

The licensee provided the following rationale to support its position that the RVLH could sustain an accidental drop of the objects, based on the evaluation presented in Appendix A to the SER.

- *Melt Pathway Mostly Outside Core Periphery*. The large hole in the baffle plate, the corium buildup in the fuel assembly locations R-6 and R-7 in the lower grid, and the corium located at the bottom of in-core guide tube number 45 indicated that the majority of melted corium flowing to the lower head used a pathway mostly outside the circumference or periphery of the active core.

- *Melt Spread out around Core Periphery*. Visual examinations of the core support structure periphery, as viewed through the space between the bottom of the baffle plates and the top of the grid rib section and also between the bottom of the grid rib section and top of the flow distributor plate, showed large accumulations of column-shaped solidified core material at and beyond the outside diameter of the core. This evidence demonstrated that molten core material not only passed through the core formers to the lower head, but also this passage caused the molten material to spread out around the core periphery on its journey to the lower head. Visual examinations showed that at least 180 degrees of the core circumference contained large quantities of solidified material.

- *Melt Impingement from Many Small Jet Streams*. Experiments and mathematical evaluations showed [123] that corium jet streams that fell into water would break up and solidify more rapidly as their cross-sectional area decreased. Therefore, it would be expected that as the melted fuel cascaded down through the core former plates and flowed through the outer periphery of the LCSA, it would impinge or strike the RVLH in the form of many small jet streams as opposed to a single jet. Consequently, although the area of the vessel head ablation could be increased beyond that described in EGG-TMI-7811, the depth of penetration would likely be less or no more severe.

- *Melt Impingement Time for Melt Ablation*. EGG-TMI-7811 stated that if an ablation of one-half of the thickness of the RVLH occurred, loss of structural integrity of the vessel head would be expected. Since the RVLH did not experience any leakage during the TMI-2 accident, it was highly unlikely that the thickness of the RVLH was reduced by 50 percent. Further, EGG-TMI-7811 calculated a direct jet impingement time of 15 to 20 minutes for melt ablation of about half the thickness of the RVLH. While the licensee agreed that ablated material and molten corium would move away from the point of impact on a curved surface, time must be allowed for reactor vessel heatup after initial impact before reactor vessel material moved out of the impact area. It was expected that it would take 25 to 40 seconds before the reactor vessel surface would reach stainless-steel melting temperatures and flow out of the impact area. Since the maximum jet impingement time calculated in EGG-TMI-7811 was 75 seconds, the licensee concluded that any ablation of the RVLH due to melted corium was bounded by the analyses in EGG-TMI-7811.

– *Visual Examinations*. Visual examinations of areas located directly under resolidified material in the LCSA, including that on the north side, did not show any evidence of erosion of the RVLH.

– *Potential Drop Locations*. Most potential jet impact locations in the lower head, regardless of jet stream size, occurred outside the circumference or outer boundary of the active core. Consequently, any object that might fall into the reactor vessel would not be expected to strike the potentially ablated area. Therefore, head integrity would not be compromised.

- ***Evaluation: Reactor Vessel Integrity (Burning and Cutting)***. [124] The licensee's safety evaluation noted that the operation of burning devices inside the vessel was previously evaluated by the licensee [125, 126] and reviewed by the NRC [127] for the LCSA defueling. During initial LCSA dismantlement, the operation of such devices was physically limited to inside the confines of the core support structure and the elliptical flow distributor where the torch was more than 1 foot away from the reactor vessel wall. As discussed in the initial SER [128] for the LCSA defueling, cutting operations were expected to begin on the top of the LCSA and sequentially cut through the lower grid rib assembly, lower grid flow distributor, lower grid forging, and in-core guide support plate to the elliptical flow distributor. Thus, the LCSA structure precluded cutting of the elliptical flow distributor until the upper layers were removed. The elliptical flow distributor, which was more than 1 foot from the reactor vessel wall, would be cut only after considerable experience was gained by using the plasma arc torch elsewhere in the reactor vessel. The arc or flame of such burning devices, operating underwater, would always be operated at least 1 foot from the reactor vessel wall. Because of rapid dissipation of the arc energy (i.e., within a few inches of water), propagation of an arc through 1 foot of water was not possible. Therefore, damage to the reactor vessel wall due to the operation of burning devices was precluded even when cutting the elliptical flow distributor.

- ***NRC Review: Reactor Vessel Integrity (Load Drop)***. [129] The NRC's safety evaluation noted that the agency had previously reviewed a progressive series of submittals from the licensee regarding the defueling. Most of the equipment, techniques, and safety issues in the subject licensee's SER were previously reviewed. The principal consideration in the SER for the completion of defueling was the removal of a portion of the elliptical flow distributor. This presented the potential interaction of defueling equipment and dropped loads with the in-core instrument penetrations and lower reactor vessel head.

- *Impact Area*. Observations made during previous defueling efforts had shown little damage to the in-core instrument penetrations and none to the lower head. Since many of the penetrations and much of the lower head were hidden under core debris, the potential for damage could not be precluded. Thus, the potential area of interaction could be intact or partially degraded. In addition, adequate forces could be generated from defueling equipment or a dropped load to shear an intact penetration if applied horizontally or obliquely. The potential for damage and thinning of the lower head due to jet impingement and ablation by molten material during the accident was limited to the area beneath fuel assemblies R6 and R7 and the area outside the core baffle plates.

- *Mitigation*. In the unlikely event of a complete shear of a penetration, an annular gap would exist between the in-core instrument string and the lower head. The maximum leakage through this annular gap would be 0.4 gallon per minute for each sheared penetration. This leakage rate would be well within the licensee's capability to make up water to the reactor coolant system using gravity feed or pumping. If an unspecified mechanism provided adequate force to push the instrument string through the lower head, a 1-inch-diameter hole and a leak of 120 gallons per minute could result. Active pumping of borated water would be required to maintain the reactor vessel level. Maintaining the reactor vessel level would not be required to maintain subcriticality or to protect the health and safety of the public. However, radiation and airborne activity could limit access to the containment building, and fuel debris could be flushed to the reactor vessel cavity.

- *Conclusion*. The NRC concluded that the proposed activities could be accomplished without significant risk to the health and safety of the public, provided that they complied with the limitations stated in the licensee's and the NRC's SERs. The NRC imposed a restriction on activities near the area of potential ablation of the lower head to preclude the creation of a leakage path larger than 1 inch, in order to keep fuel particle size and total mass within the bounds of the licensee's criticality analysis. Further, the NRC required that the portion of the elliptical flow distributor be left intact over the area of potential ablation to protect the portion of the lower head immediately below from potential load impacts. If later visual inspection verified that the lower head in this area was undamaged or that erosion was less than 0.5 inch deep, this restriction would be removed.

8.6.8 Upper Core Support Assembly Defueling

- ***Purpose***. To cut and move the baffle plates and to defuel the upper core support assembly (UCSA). This evaluation addressed the following activities: (●) cutting the baffle plates for later removal; (●) removing bolts from the baffle plates; (●) removing the baffle plates; and (●) removing core debris from the baffle plates and core former plates.

- ***Evaluation: Reactor Vessel Integrity***. [130] The licensee's safety evaluation stated that the reactor vessel integrity concerns during UCSA defueling were generally bounded by the evaluation provided in its safety evaluation report (SER) [131] for the completion of lower core support assembly (LCSA) and lower head defueling. Based on the information from the report, the licensee believed that UCSA defueling could be conducted without impairing the integrity of the reactor vessel.

Further, the licensee's safety evaluation concluded that damage to the reactor vessel wall due to the operation of burning devices would be precluded. Operation of burning devices inside the vessel was evaluated by the licensee in its SER [132,133] for LCSA defueling, SER [134] for criticality safety assessment for using the plasma arc torch to cut the LCSA, and SER [135] for completion of LCSA and lower head defueling. UCSA burning and cutting operations were limited to inside the core support structure where the torch was at least 15 inches from the reactor vessel wall. The proposed cutting operations were expected to begin at the top of eight or more locations on the baffle plates and cut the length of the plate. These cuts would be made after considerable experience had been gained using the plasma arc torch elsewhere in the reactor vessel. The arc of flame of such burning devices would be operated underwater, at least 15 inches from the reactor vessel wall. Because of rapid dissipation of the arc energy, propagation of an arc through the 15 inches (two 2-inch-thick steel plates and 11 inches of water) was not possible.

- ***NRC Review***. [136] Editor's Note: The NRC's safety evaluation report did not specifically address this evaluation topic.

8.7 Evaluations for Waste Management (NA)

8.7.1 EPICOR II (NA)

8.7.2 Submerged Demineralizer System (NA)

8.7.2.1 Submerged Demineralizer System Operations (NA)

8.7.2.2 Submerged Demineralizer System Liner Recombiner and Vacuum Outgassing System (NA)

8.8 Endnotes

Reference citations are exact filenames of documents on the Digital Versatile Discs (DVDs) to NUREG/KM-0001, Supplement 1, [137] "Three Mile Island Accident of 1979 Knowledge Management Digest: Recovery and Cleanup," issued June 2016, DVD document filenames start with a full date (YYY-MM-DD) or end with a partial date (YYYY-DD). Citations for references that are not on a DVD begin with the author organization or name(s), with the document title in quotation marks. Contents of the DVDs are available at the Idaho National Laboratory Web site (https://tmi2kml.inl.gov).

[1] USNRC, "Control of Heavy Loads at Nuclear Power Plants," NUREG-0612, July 1980 [Available at nrc.gov]

[2] USNRC, "Control of Heavy Loads at Nuclear Power Plants," NUREG-0612, July 1980 [Available at nrc.gov]

[3] *U.S. Code of Federal Regulations*, "Reactor Site Criteria," Part 100, Chapter I, Title 10, "Energy"

[4] (1986-06-11) GPU Safety Evaluation, Core Stratification Sample Acquisition, Rev. 4

[5] (1986-05-05) NRC Safety Evaluation, Core Stratification Sample Acquisition SER, Rev. 1 (re 08-30-1985, 12-31-1985)

[6] American Society of Mechanical Engineers, *Boiler and Pressure Vessel Code*, Section III, Division 1, Appendix XVII, "Design of Linear Type Supports by Linear Elastic and Plastic Analysis," New York, NY [Revision not provided]

[7] (1989-08-18) GPU Safety Evaluation, Remove Metallurgical Samples from Reactor Vessel SER

[8] (1989-10-20) GPU Safety Evaluation, Remove Metallurgical Samples from Vessel, Rev. 1 (re 08-18-1989) (effective pages)

[9] (1986-07-11) GPU, Extended Core Stratification Sample Acquisition Activity (re 06-23-1986)

[10] (1989-11-28) NRC Safety Evaluation, Reactor Vessel Lower Head Metallurgical Sampling (re 08-18-1989)

[11] (1988-06-27) GPU Safety Evaluation, Completion of Lower Core Support and Lower Head Defueling, Rev. 1

[12] (1988-12-01) NRC Safety Evaluation, Lower Core Support and Lower Head Defueling (re 06-06-1988)

[13] (1986-03-28) GPU Safety Evaluation, Reactor Building Decontamination and Dose Reduction Activities for 1986, Rev. 0

[14] (1984-02-03) NRC Safety Evaluation, Containment Decontamination and Dose Reduction (re 09-29, 01-23-1983)

[15] (1982-11-22) GPU, Inform Plans to Refill RCS

[16] (1982-12-14) GPU, Changes in Plan to Vent Pressurizer During RCS Refill (re 11-22-1982)

[17] (1983-02-18) GPU Safety Evaluation, Polar Crane Load Test

[18] (1983-11-18) NRC Safety Evaluation, Reactor Building Polar Crane Load Test

[19] (1983-09-29) GPU, Plans for First Pass Stud Detensioning for Head Removal

[20] (1984-01-30) GPU Response to NRC, First Pass Stud Detensioning SER (re 09-29, 12-06-1983)

[21] Electric Power Research Institute, "Scoping Thermal Analysis of the Three Mile Island Unit 2 Pressure Boundary," EPRI -NP-3211, July 1983, Palo Alto, CA

[22] (1984-02-17) NRC Safety Evaluation, First Pass Vessel Stud Detensioning (re 09-29, 12-06-1983, 01-30-1984)

[23] (1984-03-09) GPU Safety Evaluation, Head Removal, Rev. 5

[24] (1984-03-09) GPU Safety Evaluation, Head Removal, Rev. 5

[25] (1984-07-17) NRC Safety Evaluation, Reactor Pressure Vessel Head Lift (re -3-09-1984)

[26] USNRC, "Control of Heavy Loads at Nuclear Power Plants," NUREG-0612, July 1980 [Available at nrc.gov]

[27] (1986-06-02) GPU Safety Evaluation, Heavy Load Handling Inside Containment, Rev. 3

[28] (1986-06-02) GPU Safety Evaluation, Heavy Load Handling Inside Containment, Rev. 3

[29] (1985-05-02) NRC Safety Evaluation, Heavy Load Handling Over the TMI-2 Reactor Vessel (re 04-19-1985)

[30] (1985-11-05) NRC Safety Evaluation, Defueling Canisters (re various letters)

[31] (1985-04-19) GPU Safety Evaluation, Heavy Load Handling Over Reactor Vessel SER

[32] (1985-05-02) NRC Safety Evaluation, Heavy Load Handling Over the TMI-2 Reactor Vessel (re 04-19-1985)
[33] (1984-06-18) GPU Safety Evaluation, Preparatory Activities for Plenum Assembly Removal
[34] American Society of Mechanical Engineers, *Boiler and Pressure Vessel Code*, 1974 Edition, Section III, "Rules for Construction of Nuclear Facility Components," New York, NY
[35] (1984-09-14) NRC Safety Evaluation, Plenum Assembly Removal (re various letters)
[36] (1984-07-17) NRC Safety Evaluation, Reactor Pressure Vessel Head Lift (re -3-09-1984)
[37] (1985-05-01) GPU Safety Evaluation, Plenum Removal and Transfer, Rev. 3
[38] (1984-09-14) NRC Safety Evaluation, Plenum Assembly Removal (re various letters)
[39] (1984-05-31) GPU Safety Evaluation, Internals Indexing Fixture Processing System, Rev. 1
[40] (1984-07-24) NRC Safety Evaluation, Internals Index Fixture Water Processing System (re 05-31, 07-16-1984)
[41] (1987-05-21) GPU Safety Analysis, Defueling Water System Cross-Connect to Reactor Vessel Cleanup System, Rev. 0
[42] (1987-06-11) NRC Review, DWCS Cross-Connect to Reactor Vessel Cleanup System (re 05-21-1987)
[43] (1985-08-06) NRC Safety Evaluation, Defueling Water Cleanup System (re various letters)
[44] (1986-12-15) GPU Safety Evaluation, Addition of Coagulants to the Reactor Coolant System, Rev. 0
[45] (1986-12-31) GPU, Use of Different Coagulants (re 12-15-1986)
[46] (1987-03-27) NRC Review, Criticality Safety Evaluation for Coagulants (re 12-11-1986, 02-20, 01-05-1987)
[47] (1986-05-05) GPU Safety Evaluation, Addition of a Biocide to the TMI-2 Reactor Coolant System, Rev. 0
[48] (1986-05-06) NRC Review, Addition of a Biocide to the TMI-2 Reactor Coolant System (re 05-05-1986)
[49] (1986-06-02) GPU, Testing of Core Region Defueling Techniques
[50] (1986-05-15) GPU Safety Evaluation, Defueling SER, Rev. 10
[51] (1986-06-04) NRC Safety Evaluation, Heavy Duty Tong Tool, Spade Bucket, Hydraulic Impact Chisel (re various letters)
[52] (1985-11-12) NRC Safety Evaluation, Early Defueling of the TMI-2 Reactor Vessel (10-10-1985)
[53] (1985-10-10) GPU Safety Evaluation, Early Defueling of the TMI-2 Reactor Vessel, Rev. 4
[54] (1985-11-12) NRC Safety Evaluation, Early Defueling of the TMI-2 Reactor Vessel (10-10-1985)
[55] (1986-01-28) GPU, Fines-Debris Vacuum System Modifications (re various letters)
[56] (1985-10-10) GPU Safety Evaluation, Early Defueling of the TMI-2 Reactor Vessel, Rev. 4
[57] (1986-02-21) NRC Review, Modification of Fines and Debris Vacuum System (re various letters)
[58] (1985-08-08) Order Amendment (also exemption from 10CFR50 AppA)
[59] (1985-05-02) NRC Safety Evaluation, Heavy Load Handling Over the TMI-2 Reactor Vessel (re 04-19-1985)
[60] (1986-03-26) GPU, Plan to Use Hydraulic Shredder to Reduce Core Debris
[61] (1985-04-19) GPU Safety Evaluation, Heavy Load Handling Over Reactor Vessel SER
[62] (1986-04-16) NRC Review, Use of Hydraulic Shredder
[63] (1986-08-27) GPU, Plan to Use Plasma Arc Torch to Cut Upper End Fittings (re 08-18-1986)
[64] (1987-08-20) NRC Review, Use of Plasma Arc Torch to Cut Upper End Fittings (re various letters)

[65] (1987-11-30) GPU Criticality Assessment for Using Plasma Arc Torch to Cut Lower Core Support Assembly, Rev. 1

[66] (1988-04-01) NRC Safety Evaluation, Lower Core Support Assembly Defueling (re 01-18-1988)

[67] (1987-12-28) GPU, Use of Core Bore Machine for Dismantling the Lower Core Support Assembly

[68] (1986-09-19) GPU, Core Bore Operations

[69] (1986-10-16) NRC Safety Evaluation, Core Bore Operations (re 09-19-1986)

[70] (1988-01-08) NRC Review, Use of Core Bore Machine for Dismantling Lower Core Support Assembly (re 12-03, 12-28-1987)

[71] (1986-10-16) NRC Safety Evaluation, Core Bore Operations (re 09-19-1986)

[72] (1985-05-02) NRC Safety Evaluation, Heavy Load Handling Over the TMI-2 Reactor Vessel (re 04-19-1985)

[73] (1987-01-15) GPU Safety Analysis, Pressurizer Spray Line Defueling System, Rev. 2

[74] (1986-12-22) NRC Review, Proposal for Defueling the Pressurizer Spray Line (re 08-06-1985, 11-26-1986)

[75] (1986-03-14) GPU Safety Evaluation, Decontamination Using Ultrahigh Pressure Water Flush, Rev. 0

[76] (1985-09-11) GPU Safety Evaluation, Heavy Load Handling Inside Containment, Rev. 2

[77] (1986-07-16) NRC Safety Evaluation, Decontamination using Ultrahigh Pressure Water Flush (re 03-14-1986)

[78] (1985-08-08) Order Amendment (also exemption from 10CFR50 AppA)

[79] (1985-10-24) GPU, Commencement of Preliminary Defueling Operations

[80] (1985-10-29) NRC Safety Evaluation, Commencement of TMI-2 Preliminary Defueling Operations (re 10-24-1985)

[81] (1985-05-02) NRC Safety Evaluation, Heavy Load Handling Over the TMI-2 Reactor Vessel (re 04-19-1985)

[82] (1985-08-08) Order Amendment (also exemption from 10CFR50 AppA)

[83] (1985-10-10) GPU Safety Evaluation, Early Defueling of the TMI-2 Reactor Vessel, Rev. 4

[84] (1986-05-15) GPU Safety Evaluation, Defueling SER, Rev. 10

[85] (1986-05-20) GPU Safety Evaluation, Defueling SER, Rev. 10 (Correction)

[86] (1985-11-12) NRC Safety Evaluation, Early Defueling of the TMI-2 Reactor Vessel (10-10-1985)

[87] (1985-04-19) GPU Safety Evaluation, Heavy Load Handling Over Reactor Vessel SER

[88] (1985-09-11) GPU Safety Evaluation, Heavy Load Handling Inside Containment, Rev. 2

[89] (1985-10-29) NRC Safety Evaluation, Commencement of TMI-2 Preliminary Defueling Operations (re 10-24-1985)

[90] (1986-05-15) GPU Safety Evaluation, Defueling SER, Rev. 10

[91] (1986-05-20) GPU Safety Evaluation, Defueling SER, Rev. 10 (Correction)

[92] (1985-04-19) GPU Safety Evaluation, Heavy Load Handling Over Reactor Vessel SER

[93] (1986-07-24) NRC Review, Core Region Bulk Defueling Operations Limited to the Core Region (re 05-15, 07-23-1986)

[94] (1986-07-23) GPU, Plan to Use Core Bore Tooling as Defueling Tool

[95] (1986-05-15) GPU Safety Evaluation, Defueling SER, Rev. 10

[96] (1986-07-23) GPU, Plan to Use Core Bore Tooling as Defueling Tool

[97] (1986-06-11) GPU Safety Evaluation, Core Stratification Sample Acquisition, Rev. 4

[98] (1986-07-24) NRC Review, Core Region Bulk Defueling Operations Limited to the Core Region (re 05-15, 07-23-1986)

[99] (1986-07-24) NRC Review, Core Region Bulk Defueling Operations Limited to the Core Region (re 05-15, 07-23-1986)
[100] (1988-01-18) GPU Safety Evaluation, Lower Core Support Assembly Defueling, Rev. 2
[101] (1986-09-19) GPU, Core Bore Operations
[102] (1986-10-16) NRC Safety Evaluation, Core Bore Operations (re 09-19-1986)
[103] (1987-12-28) GPU, Use of Core Bore Machine for Dismantling the Lower Core Support Assembly
[104] (1988-01-08) NRC Review, Use of Core Bore Machine for Dismantling Lower Core Support Assembly (re 12-03, 12-28-1987)
[105] (1988-04-01) NRC Safety Evaluation, Lower Core Support Assembly Defueling (re 01-18-1988)
[106] (1988-06-06) GPU Safety Evaluation, Completion of Lower Core Support and Lower Head Defueling, Rev. 0
[107] EGG-TMI-7784, TMI-2 REACTOR VESSEL LOWER HEAD HEATUP CALCULATIONS (1987-08)
[108] (1986-09-19) GPU, Core Bore Operations
[109] (1986-10-16) NRC Safety Evaluation, Core Bore Operations (re 09-19-1986)
[110] EGG-TMI-7811, THERMAL INTERACTION OF CORE MELT DEBRIS WITHIN REACTOR VESSEL (1987-09)
[111] (1987-12-28) GPU, Use of Core Bore Machine for Dismantling the Lower Core Support Assembly
[112] (1988-01-08) NRC Review, Use of Core Bore Machine for Dismantling Lower Core Support Assembly (re 12-03, 12-28-1987)
[113] (1987-12-28) GPU, Use of Core Bore Machine for Dismantling the Lower Core Support Assembly
[114] (1986-09-19) GPU, Core Bore Operations
[115] GEND-INF-031-VOL-2, TMI-2 IN-CORE INSTRUMENT DAMAGE, AN UPDATE (1984-04)
[116] GEND-INF-031-VOL-2, TMI-2 IN-CORE INSTRUMENT DAMAGE, AN UPDATE (1984-04)
[117] GEND-INF-031-VOL-2, TMI-2 IN-CORE INSTRUMENT DAMAGE, AN UPDATE (1984-04)
[118] (1988-09-09) GPU Response to NRC, Lower Head Defueling SER (re 08-11, 06-06-1988)
[119] (1988-06-06) GPU Safety Evaluation, Completion of Lower Core Support and Lower Head Defueling, Rev. 0
[120] (1988-08-11) NRC Review, Corium Jet Assessment for Evaluation of SER on Lower Head Defueling (re 06-06-1988)
[121] EGG-TMI-7811, THERMAL INTERACTION OF CORE MELT DEBRIS WITHIN REACTOR VESSEL (1987-09)
[122] (1988-09-09) GPU Response to NRC, Lower Head Defueling SER (re 08-11, 06-06-1988)
[123] Spencer, B.W., et al., “Effect of Boiling Regime on Melt Stream Breakup in Water,” Argonne National Laboratory, 1986 [Available at osti.gov]
[124] (1988-06-06) GPU Safety Evaluation, Completion of Lower Core Support and Lower Head Defueling, Rev. 0
[125] (1988-01-18) GPU Safety Evaluation, Lower Core Support Assembly Defueling, Rev. 2
[126] (1987-11-30) GPU Criticality Assessment for Using Plasma Arc Torch to Cut Lower Core Support Assembly, Rev. 1
[127] (1988-01-08) NRC Review, Use of Core Bore Machine for Dismantling Lower Core Support Assembly (re 12-03, 12-28-1987)
[128] (1988-01-18) GPU Safety Evaluation, Lower Core Support Assembly Defueling, Rev. 2
[129] (1988-12-01) NRC Safety Evaluation, Lower Core Support and Lower Head Defueling (re 06-06-1988)

[130] (1988-09-15) GPU Safety Evaluation, Completion of Upper Support Core Assembly Defueling, Rev. 0

[131] (1988-06-06) GPU Safety Evaluation, Completion of Lower Core Support and Lower Head Defueling, Rev. 0

[132] (1988-01-18) GPU Safety Evaluation, Lower Core Support Assembly Defueling, Rev. 2

[133] (1988-04-01) NRC Safety Evaluation, Lower Core Support Assembly Defueling (re 01-18-1988)

[134] (1987-11-30) GPU Criticality Assessment for Using Plasma Arc Torch to Cut Lower Core Support Assembly, Rev. 1

[135] (1988-06-06) GPU Safety Evaluation, Completion of Lower Core Support and Lower Head Defueling, Rev. 0

[136] (1989-04-04) NRC Safety Evaluation, Completion of Upper Core Support Assembly Defueling (re 9-15-1988)

[137] USNRC, "Three Mile Island Accident of 1979 Knowledge Management Digest: Recovery and Cleanup," NUREG/KM-0001, Supplement 1, June 2016 [Available at nrc.gov]

9 OCCUPATIONAL EXPOSURE SAFETY EVALUATIONS

TABLE OF CONTENTS

Note: “NA” (not applicable) means the licensee and the NRC determined that the safety topic was not important for the activity.

9.1 Introduction

9.1.1 Background

The safety topic of occupational radiation exposure covers essentially the entire fuel cycle, ending with storage or disposal. This chapter focuses on the evaluations and the application of radiation protection during postaccident TMI-2 cleanup. The safety areas of radiation protection, ALARA, and shielding are closely related to occupational exposure and are also presented in this chapter.

Occupational exposure to ionizing radiation was a safety consideration for all cleanup activities. The safety evaluations typically included assessments of occupational exposures from external and internal radiation sources. The evaluations of cleanup activities usually entailed: (●) identification of radiation sources and source terms; (●) estimation or measurement of exposure rates; (●) estimation of the collective time spent (person-hours) in radiation fields for each task or subtask; (●) calculation of collective exposure (person-rem) to complete the activity; and (●) characterization of uncertainties of the estimates. Each evaluation would generally include an assessment of dose reduction from the applications of radiation protection and ALARA measures.

Postaccident radiological conditions at TMI-2 were substantially different from those normally encountered at commercial operating nuclear plants because of the magnitude and specific mix of the radionuclide contamination. Radiation protection controls unique to TMI-2 included control of hot particles from fuel fines and skin protection from high-energy beta radiation. In addition, many locations inside the containment building and auxiliary and fuel handling building contained extremely high radiation areas from the collection of and contamination from accident-generated water.

Although worker activities at TMI-2 were quite different from those at operating power plants, the collective dose to workers at TMI-2 following the accident and from subsequent recovery and cleanup activities was lower than the average doses experienced at operating reactors. This chapter summarizes the licensee's and the NRC's safety evaluations associated with occupational exposure and controls.

9.1.2 Chapter Contents

This chapter presents occupational exposure safety evaluations of the postaccident TMI-2 and defueling operations. It describes the many studies performed to give the reader an understanding of the thinking of the analysts at the time, the expectations and the reality, the uncertainties in the data, and the measurement and mitigation methods. It also presents a high-level timeline of cleanup activities.

The evaluations presented in this chapter ensured that all activities that could result in excessive radiation exposures to workers due to internal and external doses were addressed and consequences evaluated; controls were maintained in accordance with the requirements of the plant's license, technical specifications, procedures, and applicable regulatory requirements; and adequate contingencies were developed for normal conditions.

Practically every cleanup activity addressed the concerns of occupational exposures, including keeping doses ALARA and providing adequate radiation protection. Occupational exposure evaluations presented in this chapter include dose from normal operations. Evaluations of doses to workers and offsite populations from accidents and abnormal releases are provided in NUREG/KM Chapter 10 on radiological release evaluations.

Section 2 summarizes the key studies used to support safety evaluations. The following sections present the safety evaluation for each applicable cleanup activity or system. Section 8 lists endnotes showing references cited throughout this chapter.

9.2 Key Studies

Planning, investigations, and reviews significantly influenced the success of radiation protection and ALARA programs at TMI-2. This section summarizes several reports that documented these activities and were frequently referenced in the licensee's and the NRC's safety evaluations of cleanup activities. This list of reports is only a sampling. In addition, the summaries provided below are not complete. The interested reader should refer to the reports on this topic that are provided on the DVDs in Supplement 1 [(1)] of this NUREG/KM.

9.2.1 TMI-2 Radiation Protection Program Report of the Special Panel

(USNRC, NUREG-0640, December 1979)

This early report [(2, a)] documented the findings and recommendations of a special panel that was appointed by the NRC to review the radiation protection program at TMI-2. Following the accident at TMI-2, the licensee faced extraordinary radiation safety problems. Both the technical and management requirements in these first few weeks were substantial, and certainly unparalleled in the history of the U.S. nuclear power program. The problems were associated with emergency activities necessary to ensure that the reactor was placed in a safe-shutdown condition. In the first few weeks after the accident, many entries into high-radiation areas, frequently involving high-level concentrations of airborne radioactivity, were made to mitigate airborne releases and to provide storage for highly radioactive liquids. Additionally, many personnel were hired to meet increased needs, imposing unusual demands related to coordination and integration of these people into the radiation safety program.

The NRC became increasingly concerned during the months following the accident with the licensee's ability to adequately manage the program for worker radiation safety during the recovery of TMI-2. There was evidence of a lack of management commitment conveyed throughout the workforce that radiation protection needed to be an integral part of the recovery effort. Unplanned exposures above the NRC guidelines reinforced this concern. At meetings on

[a] Editor's Note: Operating experience is an effective driver for improvements in any program or activity. The lessons identified from this investigation and others formed the framework of the subsequent successful radiological protection program at TMI-2. This report is mentioned here as a reminder of the importance of effective quality assurance and corrective action programs.

July 13 and 18, 1979, senior NRC officials formally identified to the licensee's senior management a number of significant problems in the radiation safety program. As a result of these meetings, the NRC received commitments from the licensee to upgrade its radiation safety program according to a specified schedule. However, by mid-September, the licensee was unable to meet these commitments. The continuing uncertainty as to the adequacy of the radiation safety program at TMI prompted concern from others, including members of the U.S. Congress, officials of the Commonwealth of Pennsylvania, and the NRC Commissioners.

In September 1979, the NRC created a special panel to provide an independent review of the radiation protection program at TMI. The panel was charged with evaluating the capability of the existing and planned radiation safety program to maintain radiation exposure to personnel ALARA. To meet the schedule, the panel concentrated its efforts on the most pressing problem, program management. The panel realized that the licensee had many technical problems that needed to be solved before proceeding with recovery activities. However, because of time constraints, only a limited evaluation of some technical problems was included in the panel's report. The conclusions of the panel included the following: (●) The present radiation safety program had substantial deficiencies and required significant corrective action to support major recovery activities. (●) During the panel's review, the licensee's management demonstrated a strong commitment to upgrading the radiation safety program to ensure that radiation exposure to all employees would be ALARA. (●) The radiation safety program was capable of supporting limited work activities with the continuation of existing management controls. (●) Upgrading of the radiation safety program for the major recovery activities was not complete, so the panel could not judge the capability of this future program.

On the basis of its investigation, the panel recommended the following: (●) The licensee should be permitted to continue limited radiological recovery operations, provided that the recently established administrative controls and positive management attitude toward the radiation safety program were maintained. (●) The licensee should not perform major radiological recovery efforts until it upgraded its radiation safety program. (●) The adequacy of the upgraded program should be independently addressed before the initiation of major recovery activities. (●) The licensee should provide a management plan and firm schedule, as well as demonstrate substantial progress toward resolving the existing management and technical deficiencies in the radiation program.

9.2.2 Programmatic Environmental Impact Statement, Supplement 1
(USNRC, NUREG-0683, October 1984)

- ***Background***. The NRC's PEIS was an important document for the licensee and the NRC in their safety evaluations of cleanup activities. The report was cited in all safety evaluation reports and NRC approval letters. In 1981, the NRC Commissioners endorsed the PEIS in their policy statement that concluded that the PEIS satisfied the NRC's obligations under the National Environmental Policy Act. The policy statement also stated that, as the licensee proposed specific major decontamination activities, the NRC would determine whether these proposals and their predicted impacts, fell within the scope of those already assessed in the PEIS. With the exception of the disposition of processed accident-generated water (the Commissioners wanted to decide this issue later), the staff was allowed to act on each major cleanup activity without the Commissioners' approval if the activity and associated impacts fell within the scope of those assessed in the PEIS. [(3)]

This supplement [(4)] was required because cleanup could entail substantially more occupational radiation dose to the cleanup workforce than originally anticipated in 1980/1981. The occupational dose estimates were updated, based on previous cleanup experience, recent known radiological conditions, and better ideas on how to proceed with the cleanup, including defueling. Alternative cleanup methods were considered, as required by the National Environmental Policy Act; however, the alternative methods noted in the supplement either did not result in appreciable dose savings or were not known to be technically feasible.

- ***Purpose and Scope***. The purpose of Supplement 1 to the PEIS was to reevaluate the impact of the radiation dose to workers based on experience and data obtained since the PEIS was issued in early 1981. Since the issuance of the PEIS, the licensee had proposed many activities (e.g., cleanup of accident-generated water, reactor and auxiliary building decontamination, reactor vessel underhead characterization). The NRC evaluated these activities and determined that they fell within the scope of the activities assessed in the original PEIS. Completion of these activities resulted in considerable progress toward finishing the cleanup, along with supplying new information about conditions in the containment building and in the auxiliary and fuel handling building and about the effectiveness of various decontamination activities. One conclusion of the original PEIS was that the most significant environmental impact associated with the cleanup would result from the radiation dose received by the entire workforce from cleanup activities. That collective dose was estimated in the original PEIS to be in the range of 2000 to 8000 person-rem.

Cleanup activities conducted through May 11, 1984, resulted in about 2000 person-rem based on the results of self-reading dosimeters. Individual worker doses were based on the results of thermoluminescent dosimeters, which were more accurate and showed somewhat lower readings. Although this occupational dose was still within the predicted range, there was substantial uncertainty about future occupational exposures, primarily because the most difficult work remained to be done and, in certain areas, dose rates did not decline as projected. Based on the prior cleanup experience at TMI-2, it appeared that the entire cleanup could result in doses in excess of the 8000 person-rem previously estimated. Therefore, the NRC prepared this supplement to update the estimates of radiation dose and assess the associated environmental impacts. The doses for waste-related tasks used in this supplement were taken directly from the original PEIS. These doses were expected to contribute little to the total dose from cleanup.

This document, like the impact statement it supplements, was programmatic in nature. That is, the action being considered was the assessment of the cleanup that was subject to NRC approval. To accurately predict the impact of the occupational radiation dose from cleanup, the most probable sequences and methods for cleanup were evaluated. The most likely course of action, presented in the supplement as "the current cleanup plan," differed in sequence from the most likely course of action at the time the original PEIS was prepared. At that time, the licensee planned to begin cleanup in the containment building with an extensive decontamination of the building and equipment. Although progress was made on building and equipment decontamination, much additional work still remained. Rather than complete building and equipment decontamination before reactor disassembly and defueling as originally planned, the licensee indicated its intention to remove the damaged reactor fuel as soon as possible. Therefore, defueling before complete building cleanup was the predominant feature of the current cleanup plan, which was presented and evaluated in the supplement (refer to Section 2.2 of the supplement).

- ***Cleanup Plan***. The cleanup option chosen by the licensee and evaluated by the NRC in the PEIS supplement was for dose reduction followed by defueling and decontamination. The licensee's program for cleanup of the containment building assumed that extensive decontamination of the containment building would significantly reduce the radiation levels before reactor disassembly and defueling. This sequence was revised for several reasons. First, the containment building decontamination up to that time was less effective in reducing dose rates than was originally anticipated. Second, the presence of the damaged fuel in the reactor core constituted some risk, primarily to workers in the containment building (the risks were from uncertainties in the core configuration and the remote possibility of a boron dilution incident potentially leading to criticality of the core). Third, the information that would be obtained from laboratory examination of the damaged core would be valuable to the design of planned facilities and could also benefit the continued safe operation of other nuclear power facilities. Therefore, to avoid further delay of the core removal, the licensee adopted a revised approach to cleanup.

The revised cleanup program entailed the same milestones as the initial schedule, but the sequence of tasks was altered as follows: (●) dose reduction would continue during reactor disassembly; (●) reactor disassembly and defueling would begin; (●) primary system decontamination would follow defueling; (●) containment building and equipment cleanup would proceed, as resources allowed, with completion following that of other activities; and (●) cleanup of the auxiliary and fuel handling building would continue concurrently with containment building work until completed. The milestones and activities for which the supplement provided an occupational exposure evaluation are summarized below.

- *Dose Reduction*. The purpose of the dose reduction program was to reduce the radiation dose rates in the occupied portions of the containment building before and during reactor disassembly and defueling. These activities, which included the installation of temporary shielding and the removal of certain equipment, were well along and helped reduce the average transit dose for each worker entering the building on the 305-foot elevation (entry level) and traveling to the 347-foot elevation (operating level) and back from 40 millirem to 14 millirem. Dose reduction would continue to rely on shielding; additional source identification; and the removal, decontamination, or shielding of floor surfaces, cable trays, air coolers, and other sources of exposure. Dose reduction activities would also decrease airborne radioactive contamination and the recontamination of cleaned surfaces.

- *Reactor Disassembly and Defueling*. Reactor disassembly and defueling activities included: (●) removal of the reactor pressure vessel head; (●) installation of high-volume cleanup systems for the water in the reactor vessel and fuel transfer canal (FTC); (●) refurbishment of the FTC in the containment building and of the fuel storage pool in the auxiliary and fuel handling building; (●) removal of the reactor vessel upper internals (plenum); (●) removal of the reactor fuel, followed by its placement in containers and transfer to the fuel storage pool; (●) removal of the reactor vessel lower internals (core support assembly); and (●) removal of the remaining debris from the reactor vessel.

  – *Reactor Vessel Head Removal*. When preparations for reactor vessel head removal were completed, the reactor vessel head would be lifted with the polar crane to gain access to the reactor vessel internals and the fuel. The head and attached service structure would be placed on the storage stand with shielding surrounding the structure. If dose rates or contamination were high, the transfer canal could be filled to facilitate head lift. The internals indexing fixture and a cover would then be installed on top of the reactor vessel to facilitate water shielding of the plenum and to provide a work platform for plenum inspection activities.

  – *Installation of High-Volume Water Cleanup Systems*. The filter for the system servicing the reactor vessel would be designed to fit in modified fuel canisters and would be located in the FTC for shielding. The ion exchange columns would be shielded underwater in the transfer canal pool or placed in a shielded cask inside or outside of containment.

  – *Plenum Removal*. Plenum removal was not ordinarily a high-dose job; however, if intact plenum removal at TMI-2 was not possible, the removal could be a high-dose job. The additional dose contribution from the plenum and reactor coolant could be fairly small, depending on the depth of water coverage and the effectiveness of the water cleanup systems.

  – *Fuel Removal*. Although the fuel and reactor core material were highly radioactive, the depth of water over the core should shield workers from all debris, with the exception of dissolved or very finely divided debris that became dispersed in the coolant. The reactor water cleanup system was expected to remove this material and provide cleaned coolant in the vicinity of defueling workers. Defueling would require that workers spend considerable time in the containment building where they would receive radiation doses from many sources. Because of the time required for defueling, these sources would be a relatively large contributor to the radiation dose for cleanup.

  – *Lower Internals Removal*. Similar to defueling the reactor core region, the depth of water over the lower internals should shield workers. The reactor water cleanup system would remove dissolved or very finely divided debris that became dispersed in the coolant.

- *Containment Building and Equipment Cleanup*. The containment building cleanup plan would involve the identification of the most significant contributor to radiation dose and airborne contamination. The second step would be the decontamination or removal of that source. These two steps would be repeated for other important sources, until dose rate objectives were met. These two steps would be repeated for other important sources, until dose rate objectives were met. Recontamination by particulate radioactive material from the air, or from activities involving equipment removal or decontamination, in the adjacent areas was a concern. Dose rates in the containment building (from equipment and surfaces) would depend on the effectiveness of the cleanup actions. Cleanup was expected to require a many person-hours, and dose rates would decrease slowly as cleanup progressed, because removing a single large source would have a much greater effect on dose rates (per worker-hour expended) than removing numerous smaller sources.

- ***Occupational Radiation Dose Results (Cleanup Plan)***. To determine the occupational radiation dose associated with the cleanup plan, a team of nuclear operations and decontamination specialists evaluated the work to be performed and the dose required for each task. Each task was evaluated assuming that tasks would be performed in the sequence described and that occupational radiation doses would be maintained ALARA by the proper planning and execution of each task. The information and data required for accurate estimates would become available only during the progress of cleanup (e.g., conditions inside the reactor, characterization of contamination). Because of this uncertainty, the radiation dose estimate was presented as a range. The upper and lower ends of the estimated range represented the corresponding extremes of conditions based on an evaluation of the information presently available.

  - *Results*. The PEIS supplement listed the estimated range of occupational radiation doses for cleanup performed according to the cleanup plan. Occupational radiation dose estimations for the cleanup plan by task included: (●) dose reduction program: 2000 to 5100 person-rem; (●) reactor disassembly and defueling: 2600 to 51,000 person-rem; (●) primary system decontamination: 56 to 970 person-rem; (●) containment building and equipment cleanup: 5900 to 21,000 person-rem; (●) auxiliary and fuel handling building cleanup: 500 to 1400 person-rem; (●) utility and system maintenance: 100 to 200 person-rem; (●) waste management and transportation: 97 to 485 person-rem; and (●) dose to date: 2000 person-rem. The total dose estimate ranged from 13,000 to 46,000 person-rem. The dose estimate for waste management tasks was taken from the original PEIS.

  - *Observations*. Observations on the revised estimated doses included the following:

    – *Dose Reduction Task*. The occupational dose incurred during performance of the dose reduction task would effectively reduce the radiation doses to workers performing subsequent tasks. Eliminating this task would effectively increase the doses for later tasks.

    – *Reactor Disassembly and Defueling Task*. The range of estimated doses for completing reactor disassembly and defueling (2600 to 15,000 person-rem) was wide because of many uncertainties involving the removal of the reactor internals and fuel and the effectiveness of the water cleanup systems. The plenum could be removed intact, or an extensive effort could be needed to section and remove the plenum in pieces. The time required to transfer the fuel to canisters was likewise uncertain. If the fuel was not fused, fewer person-hours and a lower dose would be expected. However, if much of the fuel was fused, the dose would be much higher. The transfer canal would contain a myriad of small particulate sources of radiation that would be removed by the water cleanup system during defueling. If these sources were kept well underwater and transferred to fuel canisters by the water cleanup system, dose rates would be low. However, if a significant portion of these particulates formed a film on the surface of the water in the transfer canal, the average dose rate for the workers could be much higher.

- *Primary System Decontamination Task*. The licensee had identified the processes for primary system decontamination at the time of this estimate. The occupational dose required would be a function of the number and type of dead legs (sample lines and other areas of restricted flow) that workers must flush, the number of repeat processes that must be performed, the occurrence of spills resulting from leaks in the system, and the waste-handling method used.

- *Containment Building and Equipment Cleanup Task*. Cleanup of the containment building and equipment would result in an estimated 5900 to 21,000 person-rem of occupational radiation dose. As much as 80 percent of this dose was associated with cleanup of the 282-foot elevation (basement level). This estimate assumed that considerable decontamination of this elevation was performed from the 305-foot elevation (middle entry level) through floor penetrations before entry into the 282-foot elevation. As an alternative, immersion decontamination by filling the basement with water or other decontamination solutions and processing the water on either a batch or a continuous basis was considered but was not evaluated because of limited knowledge of its effectiveness. Extensive use of robotics on the 282-foot level would also reduce the dose to workers. The robotic option was explored further as Alternative 3 (defueling followed by delayed cleanup using robotics).

- *Auxiliary and Fuel Handling Building Cleanup Task*. Final cleanup of cubicles and systems in the auxiliary and fuel handling building, including the processing of decontamination waste from system and tank cleanup, was estimated to require between 500 and 1400 person-rem.

- *Utility and System Maintenance Task*. The maintenance of utilities, communication systems, and other essential services during the cleanup was expected to require an additional 100 to 200 person-rem, depending on the frequency of breakdowns and the duration of the cleanup effort.

- *Dose to Date*. About 2000 person-rem was incurred during cleanup operations through May 11, 1984. In the opinion of the NRC, the cleanup might be completed at the low estimate of 13,000 person-rem if it went well. However, even if additional problems were to arise, cleanup should be completed at less than the high estimate of 46,000 person-rem.

- ***Conclusion***. The NRC's conclusions from its reevaluation of occupational exposure are summarized below.

  - *Occupational Doses*. All options for the TMI-2 cleanup evaluated in this supplement involved occupational radiation doses higher than those predicted more than 3 years before the original PEIS. The revised estimates were based on increased knowledge of the conditions inside the containment building and of the effectiveness of decontamination and dose reduction efforts.

  - *Cost-Benefit*. The costs of the cleanup, in terms of environmental impacts, were in the radiation exposures and potential health effects among the cleanup workers. Despite the possible increase in radiation exposures to the workers, the benefits of cleanup, especially reactor disassembly and defueling, still exceeded the drawbacks. The major benefit of the cleanup would be eliminating the continuing risk of potential uncontrolled releases of radioactivity to the environment from damaged fuel or from radioactive contamination. These radiation sources were distributed throughout the primary system, the containment building, and the auxiliary and fuel handling building. In the NRC's judgment, the conclusion of the original PEIS that cleanup of the TMI-2 facilities should proceed as expeditiously as reasonably possible to reduce the potential for uncontrolled releases of radioactive materials to the environment remained valid, at least through the defueling stage.

  - *Other Benefits.* Another benefit of cleanup was the additional knowledge gained that would be useful for reducing the risks and consequences from possible future accidents at nuclear power plants. Further, the information obtained could be applied to current and planned nuclear power facilities in a variety of areas, including: (●) plant and equipment layout and design; (●) accident mitigation system design; (●) instrument location and design; (●) radioactive waste processing system design; (●) surface coatings for contamination control; and (●) mitigation of fission product releases from severe accidents.

  - *Robotics*. The only means identified in this supplement for substantially reducing the occupational dose was the extensive use of robotic technology. Under any cleanup plan that could use this technology, the feasibility of completing the cleanup would depend on developments in robotics that were uncertain at the time. Because the highest dose was associated with the containment building and equipment cleanup, an adaptation of this approach could be considered following defueling or when there were sufficient developments in robotic technology. An early decision to use robotics was not necessary as long as the licensee defueled the reactor before containment building cleanup.

  - *Dose Limits*. The occupational radiation dose to an individual worker would be limited to less than 3 rem per quarter in accordance with 10 CFR Part 20, "Standards for Protection against Radiation." [b, 5] Based on operating experience and the licensee's more stringent limits, most workers would receive radiation doses substantially below the limit.

- *Consequences*. Decontamination workers at the plant would receive a total collective radiation dose estimated at between 13,000 and 46,000 person-rem for the whole cleanup program. These ranges were broad because of uncertainties about the plant conditions and the work that would be needed to decontaminate the containment building and its contents. The consequences of this exposure were evaluated for their potential to cause premature cancer deaths among the exposed workers and to cause genetic abnormalities in their offspring.

- *Cleanup Plan*. The current plan provided the most likely path for early fuel removal. Extensive building cleanup before defueling, or the modification of defueling methods, would cause substantial, unwarranted delays in fuel removal, with attendant risks.

- *Dose Reduction*. The dose reduction program had substantial potential for lowering the total radiation dose to workers during the cleanup. ALARA considerations dictated that this effort should continue to receive a significant commitment of funds and managerial emphasis. Containment building cleanup concurrent with defueling would also be expected to reduce the occupational dose by removing sources of radiation exposure from the workplace.

- *Conclusion*. Other conclusions of the PEIS that did not pertain to occupational radiation dose remained valid. The NRC concluded that the cleanup should proceed as expeditiously as possible while ensuring the health and safety of the workers and the public. All work performed as part of the cleanup was to be done in a manner that kept occupational doses ALARA.

9.2.3 Corporation Radiation Protection Plan
(GPU Nuclear, Various Revisions)

The licensee's radiation protection plan had undergone several revisions during the cleanup period. The recovery technical specifications required the NRC's review and approval of the early revisions. The title and scope were changed in late 1982 to include Three Mile Island Unit 1 and Oyster Creek Nuclear Generating Station. The 18-page plan provided policy guidance for the licensee's radiation protection at TMI-2 and included the: (●) foundation for the radiation control program; (●) ALARA program; (●) responsibilities of workers and supervisors; (●) system of audits and reviews; (●) radiological controls training; (●) external exposure

[b] Editor's Note: Requirements in section 10 CFR 20.101, "Radiation Dose Standards for Individuals in Restricted Areas," at the time of TMI-2 cleanup permitted a worker to receive a total occupational dose to the whole body up to 3 rem during any calendar quarter, as long as the dose when added to the accumulated dose to the whole body did not exceed 5 x (N-18) rem, where "N" equals the worker's age. In addition, the licensee must determine the worker's lifetime accumulated occupational exposure with recorded exposure records. This quarterly limit was subsequently removed from the regulations during the complete revision of Part 20 that became effected on June 20, 1991. All new sections were renumbered with a "1" in front of the old number, for example, 20.1101.

controls; (●) internal exposure controls; (●) control of radioactive contamination; (●) control of radioactive materials; and (●) radiological controls organization.

The licensee's radiation protection plan was referenced in many safety evaluations of proposed cleanup activities. Revision 1 of the corporate radiation protection plan was in effect during the initial defueling operations. Select sections of this plan and the revision history are summarized below.

Editor's Note: The summary below is somewhat abbreviated and excludes sections on the foundation of the radiation control program, system of audits and reviews, and the radiological controls organization. The reader should refer to the original document provided on the DVD to Supplement 1 [(6)] of the NUREG/KM for the complete plan.

- ***Objectives***. The radiation protection plan described the philosophies and basic policy guidelines for the radiological control programs at TMI-2. The plan stated the following objectives: (●) Minimize individual exposure to radiation and radioactive material so that risks were consistent with those commonly accepted in the daily lives of plant workers. (●) Prevent any significant internal exposure. (●) Minimize collective radiation exposure. (●) Minimize contamination of personnel, areas, and equipment. (●) Minimize the production of solid radioactive waste. (●) Minimize exposure to the public.

- ***Revision History***. The summary timeline for the licensee's postaccident radiation protection plan submittal and revisions included the following events: (●) The licensee submitted its postaccident radiation protection plan for NRC review and approval in November 1979. (●) Revision 1 of the plan [(7)] was submitted to the NRC in December 1979 after receipt of substantial comments [(8)] from the NRC on the original plan. (●) Revision 2, [(9)]submitted in July 1980, incorporated NRC comments on Revision 1. The NRC provided comments [(10)] on Revision 2 in March 1981. (●) In December 1982, the existing plan was replaced with the corporate radiation protection plan. Revision 0 of the corporate plan [(11)] did not contain substantial changes; the plan was approved [(12)] with comments the following month. (●) Revision 1 of the corporate plan [(13)] was submitted to the NRC in December 1985. Changes to the plan included a new section outlining the licensee's ALARA policy. (●) In April 1986, the NRC approved [(14)] Revision 1 with comments. (●) In May 1988, the NRC approved [(15)] a proposed change to the technical specification that deleted the requirement for NRC approval of the radiation protection plan. Removal of this requirement was based on the acceptable past performance of the licensee in the area of radiation protection. Following defueling, the proposed TMI-2 possession only license required the licensee to submit and implement a radiation protection plan. (●) Revision 4 of the corporate plan [(16)] was submitted in January 1993, and subsequent revisions followed.

- ***ALARA Program***. The plan provided a policy to ensure that risks from ionizing radiation exposures during recovery and cleanup activities were maintained ALARA. Elements of the ALARA program included: (●) responsibilities of management, task supervisors, radiological control technicians, and individual workers; (●) implementation of a radiation protection program to ensure compliance with regulatory requirements and the ALARA objective; (●) procedures that, at a minimum, established requirements for pre-job planning, recordkeeping, use of special equipment, post-job evaluation, and other requirements necessary to achieve the ALARA objective; (●) post-job evaluations of those work activities with significant radiological consequence to serve as the basis for future job planning, procedure, or equipment modification to achieve ALARA; (●) incorporation of ALARA concepts into the design and construction of new systems or facilities, or the modification of existing systems or facilities, that could result in radiological concerns; and (●) training and retraining programs in the principles of radiological controls, the risk of low-level radiation exposure, and the ALARA level for all employees who could be involved in radiological activities.

- ***Responsibilities of Workers***. The plan listed the following rules to be followed by individual workers to minimize radiological problems: (●) Obey stop-work and evacuate orders. (●) Obey posted, oral, and written radiological control instructions and procedures. (●) Wear thermoluminescent dosimeters and self-reading dosimeters where required. (●) Keep track of worker radiation exposure status to ensure that exposure limits are not exceeded. (●) Remain in as low a radiation area as practicable to accomplish work. (●) Do not loiter in radiation areas. (●) Do not smoke, eat, drink, or chew in radiologically controlled areas without permission. (●) Wear anticontamination clothing and respiratory protection properly and as required. (●) Remove anticontamination clothing and respiratory protection properly to minimize spread of contamination. (●) Frisk or be frisked for contamination when leaving a contaminated area or a radiological control point. (●) Report contamination. (●) Minimize the spread of potential contamination spills and promptly report the spill. (●) Do not unnecessarily touch a contaminated surface or allow clothing, tools, or other equipment to do so. (●) Place contaminated tools, equipment, and solid waste on disposable surfaces (e.g., sheet plastic) when not in use and inside plastic bags when work was finished. (●) Limit the amount of material that has to be decontaminated or disposed of as radioactive waste. (●) Report faulty or alarming radiation protection equipment. (●) Report the presence of open wounds before work in contaminated areas; exit immediately if a wound occurs. (●) Report the medical administration of radiopharmaceuticals before arriving at the site. (●) Ensure a mentally alert and physically sound condition for performing assigned work. (●) Ensure that your activities do not create radiological problems for others and be alert for the possibilities that activities of others may change the radiological conditions to which you are exposed. (●) Supervisors must recognize their responsibility to be at the worksite to ensure that radiological control practices and procedures are enforced.

- ***External Exposure Control***. The licensee controlled external radiation exposures with the following practices and requirements: (●) Worker’s annual exposure was limited to 5 rem. (●) Collective (person-rem) exposure goals for major work were established. (●) Work involving radiation exposure was preplanned. (●) Major exposure jobs required that radiological controls be incorporated in the design, written instructions be prepared, pre-job briefings be conducted before commencing work, and post-job debriefings be conducted for lessons learned. (●) Each person entering a radiologically controlled area would be provided with a primary dosimetry device (thermoluminescent dosimeter) capable of measuring the worker’s whole-body exposure. Those entering a radiation or high-radiation area would be provided with a self-indicating, dose-integrating device. (●) Personnel access to radiologically controlled areas (e.g., high-radiation areas) would be defined and controlled according to radiological control procedures. (●) Radiological surveys would be conducted at regular intervals for airborne activity, removable surface contamination, and external radiation. Surveys would be performed to monitor the suitability of control measures, evaluate the need for additional controls, and evaluate trends for ALARA purposes. Surveys outside of radiologically controlled areas would be taken to ensure the effective control of radioactive material. (●) Unusual radiological conditions would be immediately reported to management. (●) Records of radiological surveys would be maintained. (●) Radiation survey instruments would be calibrated.

- ***Internal Exposure Control***. The licensee controlled internal radiation exposures with the following practices and requirements: (●) Exposure from internal radioactivity would be limited to no more than 1/10 of the quarterly allowable internal exposure (i.e., 52 maximum permissible concentration-hours per quarter). (●) Engineering and personnel access control would be applied to the maximum extent practicable, and respiratory protective equipment would be used when no other controls were practicable with ALARA considerations. (●) Airborne radioactivity would be measured regularly in areas where personnel could be exposed. The air the person was breathing would be sampled to supplement periodic measurements during work that had the potential to generate significant airborne radioactivity exposure to individuals. (●) Internal radioactivity would be measured before assignment and at least annually for each person who worked in radiological controlled areas. (●) Internal radioactivity would be measured promptly for each person who received significant radioactive contamination on the skin and for anyone who was suspected of inhaling sufficient radioactivity to cause measurable internal radioactivity.

- ***Contamination Control***. Radioactive surface contamination would be controlled to minimize possible inhalation or ingestion of radioactivity and to minimize buildup of radioactivity in the environment. Specific guidance included the following: (●) Measures would be taken to contain radioactivity and to minimize the number of areas contaminated in order to minimize personnel radiation exposure, to simplify subsequent personnel and facility decontamination, and to minimize the need to rely on protective clothing. (●) “Beta/gamma contamination” was defined as the amount of radioactive material that would produce greater than 100 counts per minute above background using a count rate meter with a pancake Geiger-Muller detector probe. Other instrumentation of equal or greater sensitivity could be used. (●) Procedures would be maintained to identify and control radioactive contamination of personnel, areas, equipment, and tools. (●) During planning, training, and working, emphasis would be placed on minimizing the occurrences of radioactive surface contamination on skin or on areas not controlled for radioactive surface contamination. (●) Occurrences of skin contamination would be reviewed and documented.

- ***Radioactive Materials Control***. A radioactive material control system was established to ensure that radioactive material was not lost or misplaced in a location where personnel could be unknowingly exposed to radiation and to prevent the uncontrolled spread of radioactivity to areas where the public might be affected. This system included the following requirements: (●) The number of areas where radioactive materials were stored would be minimized. (●) New radioactive material storage areas would be approved before use. (●) Numbers of radioactive items and the amount of radioactivity in storage would be minimized to the extent practicable. (●) All items having a potential for loose surface contamination would be surveyed as they were removed from radiologically controlled areas. (●) Radioactive materials that were removed from the protected security area or removed from a restricted area outside the plant that was protected by security would be controlled in accordance with an accountability procedure. This procedure would ensure that materials were not lost, improperly handled during transfer, or subject to unauthorized removal. This accountability procedure would also require inventory of radioactive materials that remained outside such areas. (●) Each incoming or outgoing shipment of radioactive material would be handled in strict compliance with written procedures. (●) Loss of unaccounted for radioactive material would be reviewed to determine the potential radiation exposure personnel might receive, to correct deficiencies, and to improve control of radioactive materials.

9.2.4 Handling of Small Particles of Fuel Debris

(GPU Nuclear, June 1986)

In a letter [17] to the NRC, the licensee documented the corrective actions taken for techniques used to survey and handle small particles of core debris found on the defueling work platform. Also included was a description of the immediate and continuing actions taken by the licensee to ensure a proper response by defueling crews, including radiological control technicians and command center personnel.

This letter was in response to two incidents that brought attention to potential hazards in handling debris particles on the defueling work platform. During the first incident, the radiological control technician did not take proper radiation surveys before picking up the particle while wearing heavy rubber lineman's gloves. However, the extremity dose was estimated to be low at 12 millirem. In the next incident, the radiological control technician performed the proper surveys (contact gamma radiation dose rate of 6 rem per hour and a beta radiation dose rate of 20 rem per hour at a distance of 6 inches) and properly coordinated actions with the defueling coordination center. All actions were appropriate and consistent with established procedures. The technician removed the pea-sized particle back to the reactor vessel with a length of adhesive tape while wearing heavy rubber lineman's gloves. The extremity dose was estimated to be 187 millirem. In both cases, the technicians stopped work and required the defueling crew on the defueling work platform to maintain a distance from the debris, thus minimizing crew exposures. Wearing heavy-duty beta radiation gloves was considered to be an effective dose reduction action. Although neither incident resulted in significant additional dose contributions to the involved individuals, the licensee concluded that measures should be taken to ensure the proper response to such incidents in the future.

In recognition of the potentially serious hazards in handling small, highly radioactive particles, the licensee took the actions described below for subsequent defueling operations.

Tools. Tools for handling (i.e., scooping up) small particles of radioactive debris were staged on the defueling work platform. These tools would eliminate the need to pick up radioactive debris by hand. Other remote techniques, such as the use of adhesive tape, would continue to be a viable option. A lead-shielded cask (with 1.5-inch-thick walls) was placed on the platform to receive and store small debris particles that could not be conveniently returned to the reactor vessel.

Work Briefings. Pre-job briefings for containment building entries were amended to include discussions of the methods for handling and disposing of small particles of radioactive core debris. These briefings stressed the need for accurate dose measurements and use of techniques to minimize dose to the body and extremities. All members of the defueling work crews received these pre-job briefings, which would continue throughout the defueling process. Also, supervisors were briefed on the two incidents, the corrective and preventive actions, and their specific responsibilities for instructing and supervising field technicians in the proper techniques for handling radioactive debris particles.

Training. The issues pertinent to the incidents were incorporated into training programs and discussed at awareness meetings. The technician involved in the first incident was retrained and counseled.

9.2.5 Final Report, Three Mile Island Unit 2 Safety Advisory Board
(GPU Nuclear, March 1990)

This final report [18] of the TMI 2 Safety Advisory Board constituted a summary of the board activities from its establishment in March 1981 through its final meeting in December 1989. The licensee established the advisory board to provide an independent appraisal of the recovery program that gave particular emphasis to the assurance of public and worker health and safety. The board met every 3 months and reviewed many aspects of recovery activities, including: (•) radiological protection; (•) regulations; (•) nuclear criticality; (•) safety; (•) risk assessment; (•) project organization; (•) project financing; (•) project procedures; (•) technical planning; and (•) public communications. Additionally, the advisory board regularly expressed its views to the NRC and to the Board of Directors of the licensee's parent company.

- ***Observations***. The board closely advised and monitored the radiation protection program and made numerous recommendations regarding its role and practices in the performance of the cleanup. In its final report, the board noted the following observations during its evaluations of worker safety.

  - *Exposures*. No radiation injury to any worker occurred. No worker was exposed to levels of whole-body or internal radiation that exceeded the regulatory limits of the NRC. Worker exposures were remarkably low, representing a collective dose equivalent of about 60 person-sievert (6000 person-rem) or about 13 percent of the conservative estimate made in the 1984 supplement to the NRC's PEIS. The radiation protection program was, in large measure, responsible for the licensee's excellent record in occupational exposures.

  - *Organization*. The licensee's radiological controls division, which was established in 1980, reported to the President of GPU Nuclear. This reporting level ensured that the program received appropriate attention, authority, and independence.

- *Balanced Approach to Safety*. A balanced perspective was the consistent theme of positions taken by the board throughout the cleanup. The board recommended an approach that minimized potential risks to workers. The attributes to this approach included: (●) identifying and reducing hazards; (●) instituting appropriate administrative controls; (●) taking a reasonable and practical approach to interpreting regulations; and (●) maintaining a balanced view of worker safety in order to reduce all potential occupational hazards. This proved to be an effective practice throughout the entire cleanup program.

- *Excessive Controls*. The board reported a major concern with the excessive regulatory controls and selective licensee restraints exerted on cleanup operations, particularly in the control of skin contamination and the sometimes unnecessary use of respiratory protection. Part of the excessive conservatism could be traced to the original PEIS's use of bounding values rather than providing guidance. The collective dose equivalent limits in the original PEIS were unrealistically low when compared to the experience of other nuclear plants under normal operating conditions. The maximum permissible dose to individual workers at any facility or activity was limited by 10 CFR Part 20. The report concluded that higher levels of collective dose equivalent could be allowed when larger numbers of highly trained workers were recruited for recovery operations. Further, higher individual exposures could be permitted if they were still within NRC limits to reduce the number of workers to be trained, thus reducing costs and improving efficiency. The board was satisfied with the realistic evaluation of projected personnel exposures given in the supplement to the PEIS in 1984, although by the end of the cleanup, the original 1981 estimate proved to be more accurate, and the scope of work actually conducted was considerably less than projected in the original PEIS estimate.

- *Airborne Contamination*. The board felt that the licensee did not address the concerns of airborne contamination in the containment building with sufficient urgency nor did it consider the contamination a matter of high priority. The board recommended a greater effort to reduce surface and airborne contamination levels in order to reduce the use of respirators and cumbersome protective clothing in the containment building. In the opinion of the board, this unnecessary equipment reduced productivity, increased the level of radiation exposure because of the lowered efficiency, and increased the probability of industrial safety risks resulting from heat stress or restricted movement. The tradeoffs between protection from radioactive contamination and industrial hazards had to be continually evaluated. For many years, the board argued in favor of the reduced use of respirators when radiological conditions permitted. This could be accompanied by increased training and education of workers about all potential health risks, including those of internal versus external radioactive contamination.

- *Training*. Detailed planning for operations and extensive training, especially the use of mockups, were essential to ensuring worker health and safety. The board strongly endorsed the continuous training of workers in the use of defueling tools and in general safe work practices. Contingency plans and readiness review meetings before all major operations proved to be a vital component of effective and successful completion of each difficult task. Extensive use of mockups was an important part of preparation for the successful execution and completion of the radioactive decontamination tasks, especially in the hostile environment of a high-beta-radiation field, in addition to that of a high-gamma field. The close support provided by radiological control personnel ensured maximum protection from both beta and gamma radiation.

- *Robotics*. The board supported the use of a fully robotic defueling system, but the licensee rejected the system as it was untested, time consuming, and expensive. The use of some robotic and remotely operated equipment was effective in supporting defueling operations, such as the remotely operated plasma arc torch that was used to cut the lower core support assembly. Decontamination by robotic means represented only a small portion of the total decontamination effort.

9.3 Data Collection Activities

9.3.1 Axial Power Shaping Rod Insertion Test

- ***Purpose***. To individually move the eight axial power shaping rod assemblies in the core to provide insight into the extent of core and upper plenum damage. This early insight allowed time to factor this information into plans for subsequent inspections for the head, upper plenum, and core removal.

- ***Evaluation: Occupational Exposure***. (19) The licensee's safety evaluation stated that the radiological aspects of this test were limited to the radiation exposure received when attaching the test instrumentation to the control rod mechanisms and when moving the video cameras. The actual test would be conducted from outside the containment building. This test would generate no effluents. The standard licensee radiological control procedures would be used for this test. The licensee estimated that this test would result in an exposure of less than 5.0 person-rem.

- ***NRC Review***. (20) Editor's Note: The NRC's safety evaluation report did not specifically address this evaluation topic.

9.3.2 Camera Insertion through Reactor Vessel Leadscrew Opening

- ***Purpose***. To insert a camera into the reactor vessel through a control rod drive mechanism leadscrew opening to provide the first visual assessment of conditions inside the reactor vessel. This activity was known as "Quick Look."

- ***Evaluation***: ***Occupational Exposure***. (21) The licensee's safety evaluation stated that the total exposure for Quick Look was estimated to be 50 to 150 person-rem. The activities included in this estimate were those associated with: (●) decontamination and contamination control; (●) rigging; (●) reactor coolant system (RCS) venting; (●) installation of temporary power and lighting; and (●) removal of a leadscrew and inspection inside the reactor vessel.

The evaluation estimated that about 300 in-containment person-hours would be associated with the covered activities. These person-hours would be spent in radiation fields that varied from 0.15 to 30 roentgens per hour. The calculated person-rem for all activities was 100 person-rem. Because of the uncertainty in the dose rates and person-hours, the person-rem for the activities could vary ± 50 percent.

The quantity of krypton-85 contained within the RCS and available for release to the containment during RCS venting was estimated to be up to 30 curies. This estimate was based on RCS sample data before RCS processing. Of the estimated total krypton-85 inventory, about 7 curies were estimated to be in the free gas volume. To minimize the occupational exposure from krypton-85 during the venting operation, the following requirements were factored into the RCS venting procedure. The discharge point would be at least 10 feet from personnel and directed away from personnel. External exposures of personnel would be monitored by personnel dosimetry. In addition, the air would be sampled during venting and samples analyzed for krypton-85, particulates, and tritium.

The release of the krypton-85 to the containment could require continuous purging of the containment for a set period of time after the venting operation to reduce the krypton-85 concentration in the containment atmosphere to ALARA. The exact length of time for the required purging would depend on the amount of krypton-85 released to the containment but could be less than 24 hours based on an assumption that 10 equivalent air changes would be more than sufficient for flushing the containment atmosphere.

During RCS venting operations, there was a possibility of releasing additional particulates and tritium to the containment atmosphere. The level of the RCS water contamination was less than the contamination level of the water that was removed from the containment sump. Therefore, no significant increase in in-containment particulates and tritium airborne contamination would be expected during RCS venting.

- ***NRC Review: Occupational Exposure***. [22] The NRC's safety evaluation noted that the licensee's estimation of occupational exposure resulting from Quick Look would range from 50 to 150 person-rem based on about 300 in-containment building person-hours. Based on the NRC site office's recent evaluation of the average exposure rates expected from performing work in the containment building (200 millirem per hour), the NRC estimated that the work in the containment would result in about 60 person-rem of occupational exposure.

9.3.3 Reactor Vessel Underhead Characterization (Radiation Characterization)

- ***Purpose***. To characterize radiation under the reactor vessel head to ensure that adequate radiological protection measures would be taken to keep radiation exposures ALARA during head removal. To achieve this objective, measurements were necessary with the reactor vessel head still in place.

- ***Evaluation: Occupational Exposure (External Dose)***. [23] The licensee's safety evaluation stated that all individuals entering the containment building were monitored for external exposures in accordance with radiological control procedures to ensure that personnel exposures were maintained within 10 CFR Part 20 dose equivalent guidelines. Administrative control points in accordance with the procedures were used to ensure that specified dose limits were not exceeded. Extremity monitoring followed existing procedures. The underhead characterization required that the reactor coolant system (RCS) water level be lowered to the 321.5-foot elevation, which was about 1 foot below the top of the plenum. Lowering of the water in the reactor vessel would remove the shielding provided by the water and raise the potential for an increase in radiation levels in the area of the reactor vessel head and service structure.

  - *Radiation Sources*. The safety evaluation included the following sources of radiation: (●) reactor vessel head; (●) service structure; and (●) control rod drive mechanism (CRDM) penetration.

    – *Reactor Vessel Head*. The increase in dose rates around the reactor vessel head was expected to be relatively small. This assumption was based on the reactor vessel head having a minimum thickness of about 6.75 inches, which provided an attenuation factor of about 7×10^{-4} for radiation sources from within the reactor vessel. Dose rates around the outside of the reactor vessel head were about 150 to 300 millirem per hour. These doses were due largely to contamination of surfaces external to the reactor vessel. The contributions from the deposited material and liquid inside the vessel were negligible. Estimates of dose rates on the underside surface of the reactor vessel head, with the RCS water level lowered, ranged from 200 to 600 rem per hour. These levels would result in an increase of about 140 to 420 millirem per hour on the outside of the reactor vessel head.

    – *Service Structure*. The increase in dose rate on the service structure platform resulting from lowering the RCS level was much less than the increase adjacent to the vessel head. This was due to the combined effects of extra distance and shielding. The top of the service structure was more than 20 feet above the vessel head. This accounted for a substantial reduction in the degree of dose rate increase. Also, the service structure housed 69 CRDMs in a tightly spaced configuration. The CRDMs constricted the pathway of radiation emanating from the vessel head so that only a fraction of the space above the head was available for streaming. The result was that the dose rate increase on the service structure platform resulting from a lower RCS level was considered to be acceptably low.

    – *CRDM Penetration*. The increase in radiation levels over a removed CRDM was estimated using the results from Quick Scan I [c] performed in December 1982. Since the CRDM penetration through the reactor vessel head and service structure would act as a

[c] Editor's Note: Quick Scan I was part of the reactor vessel underhead characterization program. An ionization chamber was lowered into the top of the reactor vessel through two access openings that were created when the CRDM lead screws wre removed for the quick look video surveys. The underhead radiation surveys provided data just above the top of the plenum at two locations.

collimator, only a portion of the surface below the reactor vessel head would contribute to the dose rate over the removed CRDM. The analysis was performed assuming a conservative plateout source of 25,000 microcuries per square centimeter of cesium-137 on the top of the plenum. The result of the analysis was an increase in the dose rate above background levels over the removed CRDM of less than 20 millirem per hour. Contributions from vented CRDMs would be negligible compared to this dose rate. The general area dose rate on top of the service structure was measured at 50 to 150 millirem per hour. The increase in the general area dose rate on the top of the service structure of less than 20 millirem per hour would be relatively small.

  – *Conclusion*. Based on the above, the safety evaluation report stated that potential increases in general area radiation levels resulting from lowering the RCS water level and removing a CRDM were considered acceptable. Measurements would be taken for varying water level conditions to assess the actual conditions with respect to the removed CRDM.

- o *Total Exposure*. The total exposure for the underhead radiation characterization was estimated to be 16 person-rem. This was based on the scope defined in the licensee's safety evaluation. The estimate assumed that the majority of the person-hours were in a dose field of about 100 millirem per hour, and about 7.4 person-hours were in higher dose fields during the actual CRDM lift, the cutting of the leadscrew support tube, and the retrieval of the camera. This estimate would be reduced by about 1 person-rem should the fallback procedure, detailed in the safety evaluation report, be used. Because of the uncertainty in both the person-hour estimate and the dose, the evaluation estimated that the total exposure could vary by up to ± 50 percent. Considering the uncertainties associated with the person-rem estimate, 8 to 24 person-rem was selected for use as the estimate for the performance of the underhead radiation characterization.

- ***Evaluation: Occupational Exposure (Internal Dose)***. [24] The licensee's safety evaluation stated that the ongoing decontamination of the containment was effective in reducing the airborne activity. Since the underhead characterization program would not increase the airborne activity level in the containment, the safe and manageable condition of airborne activity would persist. As specified by radiological control procedures, analyses of expected airborne activity levels would be performed to select appropriate respiratory protective devices for personnel entering the containment. These devices were used to protect against particulate radioactivity. Other forms of radioactivity, such as noble gases and tritium, were not expected to pose difficulties. Data from air samples of the containment indicated a mean tritium airborne concentration of 1.0×10^{-6} microcurie per cubic centimeter, which was equivalent to 0.20 maximum permissible concentration-hour per hour.

- ***Evaluation: Radiation Protection/ALARA***. [25] The licensee's safety evaluation stated that the objective of minimizing occupational exposure was a major goal in the planning and preparation for all activities in the containment. The actions toward meeting this objective included the following: (●) Protective clothing and respirators would be used as necessary to reduce the potential for external contamination and internal exposure of personnel.

(●) Exposures while performing individual tasks were maintained ALARA by a detailed radiological review by radiological engineering and mockup training of work crews. This training would approximate the actual work situation as closely as possible for each task using appropriate equipment, protective clothing, and respiratory protection. (●) Extensive planning of tasks to be conducted in a radiation field and training of personnel would be used to reduce the time needed to complete a task. (●) Photographs and the containment video system would be used extensively to familiarize personnel with the work area. (●) Higher radiation areas would be identified to personnel and shielded where practicable. (●) Work would be structured to avoid higher radiation areas to the extent possible.

In addition, all personnel assigned to the various tasks would be trained to respond to unforeseen radiation levels. Should a radiation level exceed a prescribed limit, personnel would be trained to stop present activities and proceed to the fallback procedure. Should a radiation level limit still be exceeded, while performing the fallback procedure, the CRDM would be lowered back to its original position, and all subsequent tasks would be stopped. Practice sessions would be conducted as necessary to ensure that personnel understand their assignments before entering the containment. Planning and training were proven methods of ensuring that personnel were properly prepared to conduct the assigned task expeditiously.

- ***NRC Review: Occupational Exposure***. [26] The NRC's safety evaluation considered radiological exposures to workers and the general public for the unique conditions related to lowering the reactor water level. This activity would be the first time that large surfaces were exposed inside the reactor vessel. The NRC concluded that the licensee's dose rate estimates and the estimate for the total personnel exposure (15–16 person-rem) for the underhead characterization study were reasonable, based on the information available at the time. The proposed dose rate monitoring program, which would be in effect while the reactor water level was being lowered, would provide an adequate early indication of any dose rates higher than those projected.

9.3.4 Reactor Vessel Underhead Characterization (Core Sampling)

- ***Purpose***. To obtain up to six specimens of the core debris using specially designed tools. These tools would be lowered into the reactor to extract samples of the loose debris and to load the samples into small, shielded casks for offsite shipment and analysis. The analysis of the samples would: (●) identify the composition of the particulate core debris; (●) determine particle size; (●) determine fission product content; (●) determine fission product leachability from the debris; (●) analyze the drying properties of the debris; and (●) determine whether pyrophoric materials existed in the core debris. This examination was a follow-on activity to the underhead characterization study.

- ***Background***. Three shipping casks [d] were considered [27] for shipping the core debris samples to the laboratory. The selected cask was the modified and recertified Model CNS 1-13C Type B shipping cask, with the sample in a Specification 2R container. An earlier version of this cask was used to ship spent submerged demineralizer system liners. [28]

- ***Evaluation***: ***Occupational Exposure***. [29] The licensee's safety evaluation considered radiological controls and exposure rates for the sample retrieval operation, as well as for an abnormal contingency.

  - *Radiological Controls*. All individuals entering the containment building would be monitored for external exposures in accordance with radiological control procedures to ensure that personnel exposures were maintained within 10 CFR Part 20 dose equivalent guidelines. Administrative control points in accordance with the procedures would be used to ensure that specified dose limits were not exceeded. Extremity monitoring would be performed in accordance with existing procedures. Personnel radiation exposure would be monitored throughout the execution of this task. The task would be performed with the airlock doors secured to maintain containment building isolation.

    The samples would be obtained after access to the reactor vessel was available but not simultaneously with other reactor-related operations. With the reactor coolant level lowered, the greatest potential for increased radiation levels would be the working area of the service structure. As described in the licensee's safety evaluation report [30] for underhead characterization of the reactor vessel head, the increase in the dose rate above normal background at the top of the service structure was expected to be about 20 millirem per hour.

  - *Exposure Rates (Normal Operations)*. Preliminary calculations were performed to determine estimated exposure rates during the grab sample task. Exposure rate calculations assumed that the fuel in the sample was exposed during reactor operation to the core average neutron flux. Two cases were analyzed: assuming no cesium leaching and assuming 40-percent cesium leaching. Calculations further assumed that the sampler was full of debris (39.3 cubic centimeters) with a volume void fraction of 0.3. The sample was assumed to consist of the following (material/volume fraction/percent): (●) fuel/0.303/70.3; (●) Zircaloy-4/0.102/23.7; (●) stainless steel/0.003/0.7; and (●) control rod material/0.023/5.3.

    The steel thickness for shielding consisted of 2.25 inches (5.715 centimeters) of steel cask wall and 0.065 inch (0.165 centimeter) of sampler wall. A computer code was used to determine exposure rates at various distances from the cask. The following were the results of these calculations for the two cases at 1 foot, 2 feet, and 3 feet: (●) 100-percent cesium present: 395 milliroentgens per hour, 120 milliroentgens per hour, and 57 milliroentgens per

[d] Editor's Note: While large shipping containers of radioactive materials may often be referred to as "shipping casks," the proper term for such containers, when loaded with contents and in their transportation configuration, is "package." See 10 CFR Part 71, "Packaging and Transportation of Radioactive Material," Section 71.4, "Definitions."

hour, respectively; and (●) 60-percent cesium present: 252 milliroentgens per hour, 77 milliroentgens per hour, and 36 milliroentgens per hour, respectively.

The contribution to the radiation levels noted above from activation of materials within the core was found to be negligible. [e] Therefore, assuming four teams of six workers to stage and unstage the equipment and three teams of four workers to retrieve the samples, the analysis estimated that the total exposure would be 8.4 person-rem. Because of the uncertainties associated with this estimate, 10 person-rem ± 5 person-rem had been selected as the estimate for the performance of the debris sample program.

- *Exposure Rates (Abnormal Operations)*. The evaluation further assumed that, in the course of moving the sample from the debris bed to the cask at the top of the empty control rod drive mechanism (CRDM) guide tube, the sample would become stuck in the guide tube. The cask had a steel skirt that would fit over the temporary manipulator support tube (installed on the top CRDM nozzle flange to support and guide tools into the head volume) and extend down from the top of the tube to the upper CRDM seismic plate. To eliminate any shielding contribution from the cask and skirt, the evaluation postulated that the specimen was stuck 2 feet below the CRDM support structure grating. The radiation field 2 feet from an unshielded debris sample was calculated to be 1.15 roentgens per hour.

  The safety evaluation concluded that if the sample become stuck, the operators would be protected by distance, and procedures for this event would instruct them to step away from the sample position. Procedures would also specify putting the sample back into the core cavity should the radiation field exceed 75 roentgens per hour, as measured at the sampler cask support with the sample at approximately the same elevation as the detector. The radiation from the specimen would be closely monitored as the sample was raised into the cask.

- ***Evaluation: ALARA***. [31] The licensee's safety evaluation considered programmatic ALARA practices for performing the task. The core debris samples were considered necessary at this time to provide data that would influence defueling tooling and water cleanup systems. This influence would significantly reduce exposure to workers during the plenum/fuel removal phases of the overall cleanup program, as demonstrated by the necessary modifications for the reactor vessel head removal task based on the data obtained from the first "Quick Scan" program, [32] which was conducted through a removed CRDM in December 1982. Quick Scan consisted of a simple vertical scan of the underhead region with an ionization chamber. The measured radiation levels were higher than previously assumed. The Quick Scan results concluded that activity levels measured in the upper plenum were most likely a result of cesium deposition on all vertical and horizontal surfaces. Calculations showed that activity levels were unlikely to be solely the result of debris on the horizontal upper surface of the plenum.

----

[e] Editor's note: At the time of the accident, TMI-2 had been in commercial operation for only 3 months; therefore, activation products, such as cobolt-60, were almost nonexistent.

- ***NRC Review: Occupational Exposures***. [33] The NRC's safety evaluation stated that projected occupational exposure for the debris sampling task, 8.4 person-rem, would be commensurate with the expected benefit to the accident recovery program. After a review of the sampling procedure, all reasonable measures would be taken to minimize the exposure to workers in accordance with the principles of keeping radiation exposures ALARA.

9.3.5 Core Stratification Sample Acquisition

- ***Purpose***. To perform the activities associated with: (●) installation, operation, and removal of the core boring machine; (●) acquisition of the core samples; (●) transfer of the samples from the machine to the defueling canisters; and (●) viewing of the lower vessel region through the bored holes.

- ***Evaluation: Occupational Exposure (External Dose)***. [34] The licensee's safety evaluation stated that all individuals entering the containment building would be monitored for external exposures in accordance with radiological control procedures to ensure that personnel exposures were maintained ALARA and within 10 CFR Part 20 dose equivalent limits. Administrative control points in accordance with radiological control procedures would be used to ensure that specified dose limits were not exceeded. Extremity monitoring would be performed as needed in accordance with existing procedures. The radiological controls department personnel would continuously monitor dose rates in the containment building during the sample acquisition and supporting activities.

  - *Exposure Estimates*. The collective personnel radiation exposure to workers during the core boring and sample transfer operations and also during the supporting activities in the containment building was estimated. This estimate was based on projected person-hour requirements and containment building exposure rates associated with these activities. The average dose rate during staging, assembly, disassembly, and removal was taken as 50 millirem per hour. The average dose rate during core drilling operations and video inspection at five locations was estimated to be 25 millirem per hour. The collective dose and associated exposure time were estimated as follows: (●) 17.3 person-rem for assembly and disassembly (346 hours); (●) 10.4 person-rem for drill handling (416 hours); and (●) 15.6 person-rem for core drilling (624 hours).

  - *Additional Exposure*. Person-rem for radiological control support was not included in the exposure estimates noted above. From a review of historical data, the evaluation assumed that person-rem for the radiological control group would be 20 percent of that accumulated by other groups in the containment building. Based on this, the estimate for radiological control support was 8.7 person-rem, and the total for all groups was estimated at 52 person-rem.

  - *Uncertainties*. Because of the uncertainty in the person-hour estimate and the radiological conditions that would exist during the operation, the evaluation estimated that the total exposure could vary by up to ± 30 percent. Considering these uncertainties, 35 to

70 person-rem was selected as the estimate for the performance of the activities in the scope of this safety evaluation, including radiological control support.

- ***Evaluation***: ***Occupational Exposure (Internal Dose)***. [35] The licensee's safety evaluation stated that all individuals entering the containment building would be monitored for internal radiation exposures according to established procedures. This monitoring would be done by either periodic whole-body counting or bioassay or both. All exposures to airborne radioactivity would be maintained ALARA and within the limits established in 10 Part CFR 20. Airborne radioactivity in work areas would be monitored according to established procedures. Air sampling for particulates would be performed using devices such as breathing zone air samplers and grab samples. Tritium grab samples would be taken, as required, according to established procedures.

Respiratory protection would be used to minimize the uptake and deposition of airborne radioactivity in the body. The use of respiratory protection devices resulted in overall dose savings (internal and external) by reducing uptakes of radioactive materials; however, if the devices impeded work, total dose could increase by causing an elevated external dose. Current radiation protection guidance as expressed in International Commission on Radiological Protection Publication 26, [36] "Recommendations of the ICRP," considered both external and internal sources of exposure and recommended minimizing the sum of these sources.

For soluble cesium-137, the internal dose was 2.5 millirem (received over several years; effective half-life 70 days) for each hour of exposure at maximum permissible concentration [f] (MPC). For soluble strontium/yttrium-90, the bone dose was about 15 millirem (received over 50 years; effective half-life 6400 days) for each hour of exposure at MPC. Even if there was no overall savings in the total dose due to elimination of a respirator for a given task (that is, the increased internal dose exactly offsets the decreased external dose), the fact that the internal dose was calculated on a 50-year dose commitment, whereas external dose was deposited instantly, means that the rate of dose deposition was generally reduced.

The radiological controls department, via the prework radiological review process, determined if the use of respiratory devices for a task was necessary to achieve the ALARA objective. This review examined: (●) current radiological conditions in the work area; (●) the potential of the task or other concurrent tasks to perturb the radiological conditions; and (●) when available, results of previous airborne activity measurements in the work area for similar tasks.

- ***Evaluation: Radiation Protection/ALARA***. [37] The licensee's safety evaluation stated that minimizing occupational exposure was a major objective in the planning and preparation for all activities in the containment. The actions planned to meet this objective would minimize the time personnel must work in radiation fields, maximize the distance between personnel and radiation sources to the extent practicable, and use shielding where appropriate to meet the ALARA objective. These actions included the following: (●) Protective clothing and respirators would be used as necessary to reduce the potential for external contamination and internal exposure of

[f] Editor's Note: Derived air concentration (DAC) was instituted in 1994 to take the place of the maximum permissible concentrations (MPCs) in 10 CFR Part 20.

personnel. (●) Execution of individual tasks would meet the ALARA objective through a detailed radiological review by radiological engineering and mockup training where appropriate. (●) Training of workers on a mockup would familiarize the worker with tasks to be performed. This training would result in less time and personnel exposure in the containment building. (●) Equipment would be designed to keep radiation exposures ALARA by minimizing assemblies and simplifying operations inside the containment building.

- ***Evaluation***: ***Shielding***. [38] The licensee's safety evaluation stated that when the 0.375-inch-thick stainless-steel core barrel that contained the sample was removed from the water, lead shielding with a thickness of about 3 inches would be used in the core sample transfer cask to minimize direct radiation levels. A radiological analysis performed for different source terms predicted exposure rates on the outside surface of the cask of 15 millirem per hour for cesium-leached fuel. To prevent streaming, two temporary shielding inserts would be placed at the top of the transfer cask around the core barrel during sample transfer. The cask incorporated a water manifold to supply borated rinse water around the circumference of casing or drill tubes for decontamination as they were withdrawn from the reactor.

- ***NRC Review***. [39] Editor's Note: The NRC's safety evaluation report did not specifically address the evaluation topics.

9.3.6 Use of Metal Disintegration Machining System to Cut Reactor Vessel Lower Head Wall Samples

- ***Purpose***. To remove metallurgical samples from the inner surface of the lower reactor vessel head using the metal disintegration machining system. An international research program used these samples to study core melt and reactor vessel interactions. This program was conducted after the reactor vessel was defueled.

- ***Evaluation: Radiation Protection/ALARA***. [40, 41] The licensee's safety evaluation concluded that the radiological considerations associated with these activities were bounded by the licensee's safety evaluation report [42] for bulk defueling. However, special precautions would be taken to minimize exposure of operating personnel during transport of the samples and nozzles from the reactor vessel to temporary storage in the containment building. Methods used to reduce personnel exposure included cleaning of the samples and nozzles of fuel debris before removal and use of shielded canisters. The samples were expected to be radioactive because of soluble fission products. The samples and nozzles to be removed were expected to be less radioactive than the lower grid assembly or support plates; therefore, these articles represented less of a radiation hazard. Previous defueling activities demonstrated the adequacy of the proposed personnel exposure control practices. The overall estimated occupational exposure to complete the sampling project was 53 person-rem over 6000 person-hours.

- ***NRC Review***. [43] Editor's Note: The NRC's safety evaluation report did not specifically address this evaluation topic.

9.4 Pre-Defueling Preparations

9.4.1 Containment Building Decontamination and Dose Reduction Activities

- ***Purpose***. To conduct decontamination and dose reduction activities in the containment building at elevation levels 305 feet (entry level) and above. Planned activities included: (●) flushing containment building surfaces with deborated water; (●) scrubbing selected components of the polar crane; (●) conducting hands-on decontamination of vertical surfaces; (●) decontaminating the air coolers; (●) flushing the elevator pit; (●) cleaning floor drains; (●) shielding the seismic gap and penetration; (●) decontaminating and shielding hot spots; (●) decontaminating missile shielding; (●) shielding reactor vessel head surface structure; (●) removing concrete and paint; (●) decontaminating cable trays; (●) decontaminating equipment; (●) remote flushing the containment building basement; and (●) testing remote decontamination technology.

- ***Evaluation: Occupational Exposure (External Dose)***. [44] The licensee's safety evaluation stated that all individuals entering the containment building were to be monitored for external exposures in accordance with radiological control procedures to ensure that personnel exposures were maintained within 10 CFR Part 20 dose equivalent limits. In accordance with procedures, administrative dose limits were used to ensure that 10 CFR Part 20 dose limits were not exceeded. Extremity monitoring was performed as needed in accordance with existing procedures.

When calculating the occupational exposures, the analysis assumed that 1400 in-containment person-hours were required to support the ongoing containment decontamination and dose reduction program (during 1985) and that radiation dose rates and airborne activity levels in the containment remained constant throughout the remaining activities in the containment building. The total exposure from ongoing decontamination activities was estimated at 130 person-rem. This was based on general decontamination activities and included: (●) area preparations; (●) decontamination activities; (●) cleanup operations; (●) periodic sampling; (●) health physics support; (●) equipment installation; and (●) any other activity necessary to support decontamination and dose reduction operations. The safety evaluation report presented the dose estimates.

Because of the uncertainty in the dose rates and person-hour estimates, the person-rem for the activities was estimated to vary by ± 50 percent. Considering the uncertainties associated with the person-rem estimate, 65 to 195 person-rem was selected as the estimate for the ongoing decontamination and dose reduction operations described in the safety evaluation report.

- ***Evaluation: Occupational Exposure (Internal Dose)***. [45] The licensee's safety evaluation stated that personnel entering the containment building were protected against the inhalation of gaseous or particulate radioactivity as necessary in accordance with radiological control procedures. As specified by Regulatory Guide 8.15, "Acceptable Programs for Respiratory

Protection,” [46] analyses of expected airborne contamination levels were performed to select appropriate respiratory protective devices. Air sampling for particulate activity was performed using devices such as lapel samplers and methods such as grab samples. Tritium air samples were taken unless deemed unnecessary by radiological control personnel.

An estimate of the airborne radioactivity to be encountered by the workers performing decontamination activities was derived from the average breathing zone sampler results of workers participating in the decontamination. Estimated maximum permissible concentration (MPC)-hour was 0.01 MPC-hour per hour with air purifiers (protection factor 1000) based on conservative concentrations from the estimated average breathing zone sampler. Tritium levels were not expected to pose concerns, based on bioassay results from workers participating in the decontamination

- ***Evaluation: Radiation Protection/ALARA***. [47] The licensee’s safety evaluation stated that minimizing occupational exposure was a major goal in the planning and preparation for all activities in the containment. The actions taken to meet this objective included the following: (●) Protective clothing and respirators were used as necessary to reduce the potential for external contamination and internal exposure of personnel. (●) Techniques and sequence of operations were developed to achieve the greatest decontamination at minimum person-hour and person-rem expenditures in the containment. (●) Detailed radiological review by radiological engineering and very substantial mockup training of work crews were used in the execution of individual decontamination tasks. This training approximated the actual work situation as closely as possible for each task and used appropriate equipment, protective clothing, and respiratory protection. (●) Extensive planning of tasks to be conducted in a radiation field was used to reduce the time needed to complete a task. (●) Training aids were used extensively to familiarize personnel with the work area. (●) Higher radiation areas were identified to personnel, and the work was structured to avoid these areas to the extent practical. (●) Practice sessions were conducted as necessary to ensure that personnel understood their assignments before entering the containment.

In addition, potential improvements in operational technique were fed back into future work packages and mockup trainings, in a manner consistent with the development of work activities. If the observed techniques definitively demonstrated major operational problems or the ineffectiveness of a particular decontamination technique, the decontamination activities would be altered to reflect this feedback. The evaluation of the adequacy of a particular decontamination technique weighed several operational factors such as person-rem and person-hour expenditure, personnel safety, operational complexities, and training requirements.

- ***Evaluation: Shielding***. [48] The licensee’s safety evaluation noted that temporary shielding could be used for dose reduction where decontamination was difficult or ineffectual or for ALARA purposes. Shielding attached to or interfacing directly with systems or structures that were safety related would be evaluated to ensure that the safety functions provided by these components were not adversely affected by this shielding. The placement of temporary shielding on piping would be controlled to ensure that the pipe loading limits were not exceeded.

Other shielding structures were reviewed to assess their potential interface with safety-related components, including the potential collapse of large shielding structures.

- ***NRC Review: Occupational Exposure***. [49] The NRC's safety evaluation noted that the licensee established the cumulative occupational dose expected to be incurred during extensive decontamination of the containment building to be 180 to 535 person-rem. The NRC's evaluation agreed with the licensee's estimate of cumulative occupational dose expected to be incurred during the extended decontamination of the containment building. This estimate was based on measured radiation levels in the containment building and estimated cumulative occupancy time by personnel performing the decontamination, as well as their dose data obtained from previous entries into the containment building. This estimated occupational dose was a small fraction of the occupational dose discussed in the PEIS for activities related to containment building decontamination and dose reduction. The NRC's safety evaluation concluded that corresponding potential health effects were also well within the scope of those given in the PEIS.

To calculate the total expected exposure for the ongoing containment decontamination and dose reduction activities over the suggested 16-month period, the licensee assumed the planned in-containment person-hours to support those decontamination and dose reduction activities to be 3000 person-hours. The NRC stated its understanding that the 3000 person-hours assumption was for estimating the range of possible occupational doses and that there was some uncertainty in this estimate.

- ***NRC Review: Radiation Protection/ALARA***. [50] The NRC reviewed the proposed plans and engineering features aimed at reducing occupational doses expected to be incurred during containment building decontamination and dose reduction efforts. The NRC found that the plans and features provided adequate assurance that these activities would be conducted consistent with the principle of maintaining ALARA radiation doses. The NRC's safety evaluation stated that it would continue to closely monitor the licensee's overall ALARA program.

9.4.2 Reactor Coolant System Refill (NA)

9.4.3 Reactor Vessel Head Removal Operations

9.4.3.1 Polar Crane Load Test

- ***Purpose***. To conduct load testing of the polar crane in preparation for reactor vessel head lift and removal. The polar crane load test required four major sequential activities: (1) relocation of the internals indexing fixture; (2) assembly of test load; (3) load test; and (4) disassembly of test load.

- ***Evaluation: Occupational Exposure***. [51] The licensee's safety evaluation considered occupational exposure during load test activities. Radiological considerations for the protection of workers from radiation exposures included the following:

- *Exposure Estimates*. The total exposure for the polar crane load test was estimated to be 30 to 90 person-rem. This was based on the scope defined in the safety evaluation report, which estimated 540 person-hours inside the containment building. Because of the uncertainty in the dose rates and person-hours, the person-rem for the activities was estimated to vary by ± 50 percent. Considering the uncertainties associated with the person-rem estimate, 30 to 90 person-rem was selected as the estimate for the polar crane load test.

- *External Exposure Controls*. All individuals entering the containment building were monitored for external exposure in accordance with radiological control procedures, to ensure that personnel exposures were maintained within 10 CFR Part 20 dose equivalent guidelines. Administrative control points in accordance with procedures were used to ensure that specified dose limits were not exceeded. Extremity monitoring was performed in accordance with existing procedures.

- *Internal Exposure Controls*. Personnel entering the containment building were protected against the inhalation of particulate radioactivity in accordance with radiological control procedures. As specified by these procedures, analyses of expected airborne contamination levels would be performed to select appropriate respiratory protective devices. Air sampling for particulate activity was performed using devices such as lapel samplers and methods such as grab samples. Tritium air samples would be taken as required.

- ***Evaluation: Shielding***. (52) The licensee's safety evaluation considered the impact on radiation shielding due to a load drop. Radiation shielding was composed of major contributions from steel in the reactor vessel, concrete and steel in the primary shield, steam generator D-rings, [g] and concrete and steel in the containment wall. No planned activity or any consequences of a credible unplanned occurrence associated with the load test had the potential to degrade major shielding components to a degree that the shielding would not be functional. This was based on engineering judgment resulting from a comparison of load/target relative weights and strengths, as well as a review of load paths and credible impact orientations. For example, it was not considered credible that a missile shield composed of concrete and rebar could penetrate the 8-inch-thick steel reactor vessel head when the shield fell from the lift height used in the load test.

- ***NRC Review: Occupational Exposure***. (53) Based on the scope of work that defined the activities in the containment for the requalification of the polar crane, the licensee's evaluation estimated that the load test would require about 270 person-hours. The bulk of the work would take place on the operating floor of the containment building where the average exposure field was about 110 millirem per hour. Therefore, conducting the load test would result in an expected occupational exposure of about 30 person-rem. The NRC's safety evaluation considered this estimate to be reasonable and concurred that the occupational exposure

[g] D-rings were the shield enclosures around the steam generator compartments; they were so named because of their shape.

resulting from this effort would be somewhat less than 50 person-rem. The proposed activity and associated environmental impacts were well within the impacts previously assessed in the PEIS.

9.4.3.2 First-Pass Stud Detensioning for Head Removal

- ***Purpose***. To perform the first-pass detensioning of the 60 reactor vessel studs and the removal of up to 5 reactor vessel studs to check for stuck nuts and to examine the condition of the removed studs.

- ***Evaluation: Occupational Exposure***. [54,55] The licensee's safety evaluation concluded that the first-pass detensioning and removal of up to five studs would result in an occupational exposure of 15 person-rem. This estimate was approximate and was based on an estimated 151 person-hours expended during six entries. No additional exposure was included for removing stuck studs if any were encountered.

- ***NRC Review: Occupational Exposure***. [56] With regard to the occupational exposure likely to be incurred to complete the first-pass stud detensioning activities, the licensee estimated a total exposure of 15 person-rem based on 151 in-containment person-hours. The NRC's safety evaluation concluded that the licensee's estimate was low because the estimate did not provide for the contingency of dealing with stuck studs and nuts. More than 5 years had passed since the previous detensioning of the studs, and the studs had surface rust and corrosion. The NRC believed that there was substantial potential for additional work during stud removal. The agency estimated that about 75 in-containment person-hours would be required to remove up to five studs and nuts that could not be manipulated using the standard techniques. Accordingly, the NRC estimated that first-pass stud detensioning activities could result in as much as about 23 person-rem of occupational exposure. This estimate was well within the range of impacts previously assessed in the PEIS.

9.4.3.3 Reactor Vessel Head Removal Operations

- ***Purpose***. To prepare for reactor vessel head removal to perform the lift and transfer of the head and conduct the activities necessary to place and maintain the reactor coolant system in a stable configuration for the next phase of the recovery effort.

- ***Evaluation: Occupational Exposure (External Dose)***. [57] The licensee's safety evaluation considered the estimation of collective radiation exposure to workers during the head removal evolution. The estimate was based on the projected person-hour requirements and containment building exposure rate for each phase of activities associated with head removal. These estimates were based on assumptions and uncertainties provided in the safety evaluation.

    - *Exposure Estimates*. The person-rem estimates and associated person-hour estimates for the head removal program included: (●) perform head lift preparations: 188 person-rem

(1170 person-hours); (●) prepare reactor vessel for head removal: 90 person-rem (525 person-hours); and (●) lift head and store: 210 person-rem (865 person-hours). Totals were 488 person-rem and 2560 person-hours.

The projected person-hour estimates were based on the time that workers were in the containment building. The estimates did not include person-rem for radiological control support. From a review of historical data, the evaluation assumed that person-rem for the radiological control group would be 20 percent of that accumulated by other groups in the containment building. Based on this, the estimate for radiological control support was 98 person-rem for the head removal program.

- *Assumptions*. The person-rem estimates were calculated using the following assumptions about expected radiological conditions in the containment building: (●) average general area dose rates for elevations at 347 feet (operating level) and 305 feet (entry level) were 100 millirem per hour and 300 millirem per hour, respectively; (●) before head lift, the average general area dose rate around the head and service structure was 200 millirem per hour; (●) during actual head removal activities, the general area dose rate for individuals involved with the lift was 500 millirem per hour; and (●) dose rate around the head and service structure was 200 millirem per hour with the head on the storage stand.

- *Uncertainties*. Because of the uncertainty in the person-hour estimate and the radiological conditions that would exist during the head lift activities, the evaluation estimated that the total exposure could vary by ± 30 percent. Considering these uncertainties, 410 to 761 person-rem was selected as the estimate for the head lift and transfer, along with activities in preparation for and directly supporting head removal, including radiological control support. These collective exposure estimates were developed for the entire head removal program. Detailed exposure estimates would be developed on a task-by-task basis as a normal part of ALARA review of work in the containment. This would ensure that each activity was performed with a focus on person-rem.

- ***Evaluation: Occupational Exposure (Internal Dose)***. [58] The licensee's safety evaluation stated that an estimate of airborne radioactivity to be encountered by individuals performing head removal activities was derived from the personnel breathing zone air samples taken in the containment building during June and July 1983 for radioactive particulates. Tritium levels were from tritium grab samples taken in the containment building from October 1, 1982, to August 8, 1983. Results were provided for containment building concentration and worker maximum permissible concentration (MPC), both in units of microcuries per cubic centimeter: cesium-137, $2.9x10^{-8}$ (concentration) and $1x10^{-8}$ (MPC); cesium-134, $2.0x10^{-9}$ (concentration) and $1x10^{-8}$ (MPC); strontium-90, $2.5x10^{-9}$ (concentration) and $1x10^{-9}$ (MPC); and tritium, $4.1x10^{-7}$ (concentration) and $5x10^{-6}$ (MPC). These levels were acceptable for workers since respiratory protection with particulate protection factors of 1000 were normally used, and stay times in the containment building were controlled.

- ***Evaluation: Radiation Protection/ALARA***. [59] The licensee's safety evaluation considered ALARA measures for radiation controls, monitoring, and exposure reduction.

- *External Radiation Control*. All individuals entering the containment building were monitored for external radiation exposures according to established radiological control procedures. All external radiation exposures would be maintained within the dose equivalent limits established in 10 CFR Part 20. All personnel exposures would be maintained ALARA. The licensee established lower administrative dose limits in its procedures to help maintain exposures ALARA and to ensure that the 10 CFR Part 20 limits were not exceeded. Extremity monitoring would be performed, as needed, according to existing radiological control procedures.

- *External Radiation Monitors*. Radiation exposure rates inside the containment building would be monitored during head lift and transfer. The radiological controls department would determine requirements for radiation monitoring for personnel protection. A multichannel radiation monitoring system would be installed before the removal of the reactor vessel head. The system consisted of multiple area radiation monitors that would be positioned at strategic locations throughout the containment building. Such locations could include multiple detector points around the vessel flange and several locations along the load path of the head. Monitors would be positioned at the anticipated personnel work and access areas during the head lift and transfer activities. The monitoring system readouts would be at a console area located on top of the pressurizer missile shield. Radiological controls department personnel would use this system and/or other instrumentation to continuously monitor dose rates in the containment building during head lift and transfer. After head removal, radiation monitors could be positioned on the internals indexing fixture (IIF) cover to monitor the radiation dose rate on the work platform.

- *Internal Radiation Control*. All individuals entering the containment building would be monitored for internal radiation exposures according to established procedures. This monitoring could be done by either periodic whole-body counting or bioassay, or both. All exposures to airborne radioactivity would be maintained ALARA and within the limits established in 10 CFR Part 20. Airborne radioactivity in work areas would be monitored according to established procedures. Air sampling for particulates would be performed using devices such as lapel samples and grab samples. Tritium grab samples would be taken as required according to established procedures.

  Airborne particulate activity could increase after the head was removed from the vessel. To minimize the increase of airborne particulates, the following precautions would be taken: (●) Contamination control devices could be used to ensure that airborne radioactivity did not increase to unacceptable levels. The need for these controls would be based on data obtained during the underhead characterization program. If it was determined that special controls were necessary, devices such as tents and enclosures, water sprays, or ventilation systems could be used. (●) The head would be bagged on the storage stand to control airborne radioactivity, as required by radiological conditions. (●) The IIF would be installed on the vessel as soon as possible after head removal. Once filled with water, this cylindrical fixture would inhibit the release of airborne activity from the exposed contaminated surfaces of the upper plenum. A water cleanup system would be used to minimize radioactivity dissolved in the water in the fixture. (●) During head removal, grab samples would be taken periodically to monitor containment building atmosphere.

- *Exposure Reduction*. Minimizing occupational exposure was a major goal in the planning and preparation for all activities in the containment. During the planning stages of the head removal program, the principles of ALARA were considered. Such considerations included the following:

  – *Performance Enhancements*. In studying the alternatives for removal of the reactor vessel head, ALARA was considered, and specific actions were taken on tools and equipment to enhance the performance of certain operations. Also, operational sequences were reviewed and changed to allow performance of work in lower radiation areas. Performance enhancements discussed in the safety evaluation report included: (●) modifications to the hydraulic stud tensioners to improve the rate of detensioning; (●) new stud handling tools to reduce time to unthread the studs; and (●) installation of stud hole seal plugs and stud hole corrosion inhibitor before head lift at lower radiation levels.

  – *Protective Clothing and Respirators*. Protective clothing and respirators would be used as required by personnel to reduce the potential for external contamination and internal exposure.

  – *Mockup Training*. Exposure during execution of individual tasks was maintained ALARA through a detailed radiological review by radiological engineering and mockup training. The need for mockup training would be determined on a case-by-case basis. The degree of difficulty and newness of the operation would influence the need for mockup training and the level of its detail.

  – *Planning and Training*. Extensive planning of tasks to be conducted in a radiation field and training of personnel would be used to reduce the time needed to complete a task. Photographs and the containment building video system would be used extensively to familiarize personnel with the work area. The higher radiation areas were identified to personnel, and the work was structured to avoid these areas to the extent practical. Practice sessions would be held as necessary to ensure that personnel understood their assignments before entering the containment. Planning and training were proven methods of ensuring that personnel were properly prepared to conduct the assigned task expeditiously.

  – *Distance and Shielding*. Sources of high radiation that could not be avoided during head lift activities would be shielded based on the principles of ALARA. Examples of high-radiation sources for which shielding could be provided included the reactor vessel head, the service structure, and the upper plenum. In addition, the head storage stand would be shielded to reduce dose rates around the head storage area after the transfer of the head as needed.

During head removal, the top portions of the control rod guide assemblies and the plenum cover plate would be exposed and were expected to be strong radiation sources. To avoid these radiation sources, all activities during and immediately following the actual removal of the head would be performed from either the 347-foot elevation (operating level) or from the top of the D-rings, when practical. Local shielding at the vessel could be used if personnel access to the refueling canal floor was necessary. If required, shadow shields could also be used for personnel on the upper elevations.

After head removal, the IIF would be installed on the vessel flange. The installation of the fixture, if required, would be done remotely to avoid excessive radiation exposure. The fixture would be filled with water to provide shielding for the upper plenum radiation source. A cover capable of supporting lead shielding would also be installed to provide a work platform on the fixture. These measures would reduce the radiation source that was uncovered by removal of the head.

– *Water Processing*. Water processing would be used to reduce radioactivity dissolved in the water and reduce the general area exposure rate above the IIF cover plate.

– *Staffing*. The number of personnel in the containment during and after head removal would be minimized.

– *Data Collection*. Data from the underhead characterization program would be used to conservatively predict with some certainty the dose rates that would be present at worker locations after the lift of the head. Based on these predictions, shielding could be installed to protect workers directly involved with the head lift and transfer operations.

– *Remote Cameras*. A system of multiple video cameras would be used to monitor the actual lifting of the head and the installation of the head on the storage stand. This would eliminate the need for personnel in areas of potentially high radiation. The video monitor would be set in the console area on top of the pressurizer missile shield. Shielding could be used around the console area to ensure low radiation fields for personnel monitoring the lift and transfer of the head.

- ***NRC Review: Occupational Exposure***. [60] The NRC's safety evaluation considered individual worker exposure and collective worker exposure.

o *Individual Worker Exposure*. Reactor vessel head removal activities involved manual manipulations around significant sources of radiation (e.g., the underside of the head and exposed plenum) that presented the potential for individual worker overexposure. The licensee established measures (e.g., use of shadow shielding and distance from known sources) to reduce worker exposure and to minimize the potential for overexposure. Based on administrative procedures and controls that were enforced in 1984, the radiation exposures to any individual cleanup worker would be kept below the regulatory limits for occupational radiation doses.

– *Head Lift*. The majority of head removal activities would be performed in gamma radiation fields of 7.5 to 150 millirem per hour. The highest gamma radiation fields where workers would be stationed were not expected to exceed 300 millirem per hour. The NRC concluded that the maximum radiation fields under expected and abnormal conditions would not exceed the worker dose limits specified in the regulations.

  Throughout the entire head removal evolution, there would be adequate real-time monitoring instrumentation to allow for continuous monitoring and evaluation of the occupied areas. Preentry planning and continuous personnel monitoring dosimetry would minimize the potential for overexposure. Should unexpected difficulties occur, such as leakage from the IIF after placement on the reactor vessel flange and subsequent filling, contingency plans existed (e.g., remote clamping of the IIF) to eliminate the need for personnel to access high dose rate areas.

– *Plenum Assembly*. Based on data collected during the underhead characterization study, the expected dose rates after head removal were estimated. The calculated dose rate in the refueling canal at a distance of about 5 feet from the inside diameter of the vessel flange was about 10 roentgens per hour. However, following head lift, the installation of the IIF and subsequent filling with water would be performed remotely, so the worker exposure to those relatively high radiation fields would not be necessary.

– *IIF Flange Seal*. The NRC reviewed the design features for IIF sealing, which were intended to produce a tight seal without the need for personnel access into the refueling canal area. To provide a watertight seal, a soft gasket would be placed on the IIF and seated on the reactor vessel flange during IIF installation. The gasket seating surface on the vessel flange was outboard of the second flange O-ring and inboard of the mating point between the vessel flange and head mating surfaces. This surface, which was expected to be clean, would be inspected before positioning of the IIF. The IIF flange surface would be inspected to ensure that there were no dents or surface marks that would cause leakage. A mockup test was performed that simulated the installation and water filling of the IIF on the vessel flange. These inspections and tests provided assurance that no significant leakage should occur under the weight of the IIF alone. Hold-down “dogs” (at least 10) would be bolted around the IIF to ensure both alignment and a tight gasket seal on the vessel flange. As a contingency, if modifications were needed on the IIF and gasket, the refueling canal could be partially flooded to ensure leaktightness.

  The NRC concluded that there was reasonable assurance that the IIF could be installed and filled with water remotely, allowing the water to shield the strong radiation source from the upper plenum surfaces, with little risk of worker overexposure.

– *Conclusion*. Based on the above radiological considerations, the NRC concluded that during head removal activities, radiation exposures to cleanup workers could be kept at levels below those limits of section 10 CFR 20.101.

- *Collective Worker Exposure*. The NRC review estimated the total occupational radiation exposure for all the tasks associated with the head removal program.

  – *Head Lift*. The review estimated that the activities related to head removal would result in a collective occupational dose ranging from 300 to 1100 person-rem. The NRC expected the actual head lift to result in about 20 percent of the collective dose; no other single task was expected to account for more than 10 percent of the maximum expected occupational dose. The head removal activities consisted of numerous tasks including: (●) pre-head-lift preparations; (●) final detensioning of the reactor vessel head studs; (●) the actual head lift and transfer to the head storage stand; (●) the installation of the IIF; (●) the installation of shielding; and (●) various radiation health physics surveillance and monitoring activities. Some of these activities, such as the installation of shielding on the IIF cover and the activities around the head storage stand and reactor vessel service structure, would have a long-term dose rate reduction effect for maintaining occupational doses ALARA.

  – *Worst Case*. In comparison with the licensee's estimate of occupational dose (410 to 760 person-rem), the NRC's estimate encompassed a larger range. The low estimate was somewhat below the licensee's low estimate because ongoing containment building dose reduction activities could significantly reduce ambient radiation levels before head lift. While finding that the licensee's overall estimates were fairly realistic, the NRC noted that it was prudent for this estimate to include the pessimistic worst case upper range. This high estimate covered the possibility of operational difficulties in the critical time between head lift and IIF installation and filling and the possible need for more cleanup or for additional shield placement following head lift. In addition, this high estimate accounted for some isolated "hot spot" locations where the NRC calculated higher dose rates compared to the licensee's calculations. Although worker occupancy at those locations was not expected, the high estimate included the possible placement of additional shielding and the need for workers to be in the vicinity.

  – *Conclusion*. The PEIS estimated the occupational exposure to be incurred by cleanup workers to be 2000 to 8000 person-rem. Actual occupational exposures for cleanup activities up to that date (1993 person-rem as of May 11, 1984) plus those projected to occur during head removal fell well within the estimated range of the PEIS.

• ***NRC Review: Radiation Protection/ALARA***. [61] The NRC's safety evaluation stated that the licensee was required to maintain occupational exposures ALARA throughout all cleanup operations, including during reactor vessel head removal activities. Accordingly, the NRC reviewed the head removal program to ensure compliance with the ALARA principle. Before, during, and after head lift, workers would be exposed to radiation from sources within the head and the reactor vessel and from other sources within the building. The measures to reduce doses for the following activities were evaluated: (●) head lift preparation; (●) head lift and IIF installation; and (●) the impact of the filled IIF and stored reactor vessel head on work in the containment building following head lift.

- *Dose Reduction (Background Radiation)*. The background radiation in the containment building accounted for a substantial portion of the collective occupational dose. An effective ALARA program required an ongoing dose rate reduction effort to reduce the background radiation levels in the containment building. Since fall 1982, the licensee had in place an aggressive dose reduction program. This program consisted mainly of identifying, removing, shielding, and/or decontaminating discrete radiation sources. Area radiation levels and transit dose incurred by workers in the building decreased substantially.

- *Distance and Shielding (Head Lift)*. Head lift would expose workers not only to present radiation sources but to highly contaminated new sources, such as the underside of the reactor vessel head, the leadscrews that would be parked in the reactor vessel head service structure, and the plenum. Accordingly, the licensee prepared to ensure that doses were ALARA during head removal. Distance and shadow shielding would be used to minimize the doses to workers who must guide the head lift and the installation of the IIF. The workers were stationed on the tops of the D-rings and would not be immediately adjacent to the head. Shielding was placed on the service structure to reduce the dose rate from the lead screws. Water columns were also used to shield workers from the underhead radiation sources after placement of the head on its storage stand.

- *Shielding (Head Storage)*. After the reactor vessel head was placed on the head storage stand and the IIF was installed, the reactor vessel and head radiation sources had the potential to impact work in the containment building. The major new sources of radiation on the 347-foot elevation (operating level) of the containment building were the leadscrews in the parked position inside the head service structure, the reflected radiation shine from the underhead surfaces with the head supported 4 feet off the floor on the storage stand, and the exposed plenum. The leadscrew source was shielded by the control rod drive mechanism motor tubes, stators, service structure barrel, service structure shield assemblies (lead blankets of 44 pounds per square foot that extended from the head flange up to the monorail support beams), and the 12-foot high and about 2-foot-thick water [(h)] shield columns. The water columns also shielded the shine from the underhead surfaces.

- *Shielding (Plenum)*. The exposed plenum source would be shielded by the IIF and its cover. The IIF would be installed semiremotely from the polar crane and D-rings and filled with water (4 to 5 feet) to shield the exposed plenum, which was a strong radiation source. As the reactor coolant level was raised in the IIF, the coolant was processed to ensure that it did not significantly contribute to doses. Dose rates on the IIF from sources in the vessel were expected to be about 5 to 15 millirem per hour with the IIF fully filled. This dose rate was further reduced by the incorporation of about 1 inch of lead shielding in the IIF cover.

- *Stay Time Reduction*. The licensee's ALARA efforts were directed not only at minimizing radiation fields, but also at minimizing the accumulated stay time of personnel in radiation areas for the required tasks. For example, new stud handling tools with air pressure drives

[h] Editor's Note: Water was eventually replaced with sand for easier maintenance of the shield columns.

were procured for use in unthreading the studs, reducing the time required for operation by a factor of about 3.

The licensee also had an extensive program to ensure that workers were adequately prepared to expeditiously conduct the tasks in the containment building. Methods to reduce stay time involved preplanning, training, and mockup exercises before execution of the tasks and supervision by video whenever possible.

- *Conclusion*. Based on reviews of the licensee's plans and programs, the NRC concluded that there was adequate assurance that performance of the head removal activities would be consistent with the principle of maintaining doses to workers at ALARA levels.

9.4.4 Heavy Load Handling inside Containment (NA)

9.4.5 Heavy Load Handling over the Reactor Vessel

- ***Purpose***. To permit the handling of heavy loads (greater than 2400 pounds including rigging weight) in the vicinity of the reactor vessel. This included any load handling activity that could result in a heavy load drop onto or into the vessel either directly or indirectly (as the result of the collapse of structures or equipment installed over the vessel).

- ***Evaluation: Occupational Exposure***. (62) The licensee's safety evaluation stated that radiological considerations for heavy load handling over the reactor vessel were not significantly different than for other activities performed in the containment building, excluding the potential releases due to a load drop. Radiological considerations such as occupational external and internal exposures and special radiological precautions would be addressed in documents generated for specific tasks, including procedures, unit work instructions, and radiological review documentation.

- ***NRC Review***. (63) Editor's Note: The NRC's safety evaluation report did not specifically address this evaluation topic.

9.4.6 Plenum Assembly Removal Preparatory Activities

- ***Purpose***. To confirm the adequacy of plenum removal equipment and techniques, the preparatory activities included: (●) video inspection of potential interference areas; (●) video inspection of the core void space; (●) video inspection of the axial power shaping rod (APSR) assemblies; (●) measurement of the loss-of-coolant accident restraint boss gaps; (●) measurement of the elevations of the APSR assemblies; (●) cleaning of the plenum and potential interference areas; (●) separation of unsupported fuel assembly end fittings; and (●) movement of the APSR assemblies.

- ***Evaluation: Occupational Exposure (External Dose)***. (64) The licensee's safety evaluation addressed collective exposure and sources of exposure.

- *Collective Exposure*. The collective personnel radiation exposure to workers during the inspection activities was estimated. The estimate was based on projected person-hour requirements and containment building exposure rates associated with the inspection activities.

  - *Exposure Estimates*. The estimate of 105 person-rem was taken from a previous analysis based on 1035 in-containment person-hours. This estimate did not include person-rem for radiological control support. From a review of historical data, the analysis assumed that person-rem for the radiological control group would be 20 percent of that accumulated by other groups in the containment building. Based on this, the estimate for radiological control support was 21 person-rem, and the total for all groups was estimated at 126 person-rem.

  - *Uncertainty*. Because of the uncertainty in the person-hour estimate and the radiological conditions that would exist during the inspection activities, the evaluation estimated that the total exposure could vary by up to ± 30 percent. Considering these uncertainties, 90 to 165 person-rem was selected as the estimate for the preparatory activities for plenum assembly removal, including radiological control support. Detailed exposure estimates would be developed on a task-by-task basis as a normal part of ALARA review for work in the containment building to ensure that each activity was performed with a focus on person-rem.

- *Radiation Source*. Long-handled manual impact tools would be used to attempt to dislodge stuck APSR assemblies. As with the fuel assembly end fittings, the intent would be to knock the APSR assemblies into the core region so that plenum removal was not restricted. If the APSR assemblies could not be knocked into the core region, an attempt would be made to withdraw them into the plenum using a coupling tool and crane. The expected dose contribution from withdrawn APSR assemblies would be minimal.

• ***Evaluation: Occupational Exposure (Internal Dose)***. [65] The licensee's estimate of airborne radioactivity encountered by individuals performing plenum removal preparatory activities was derived from the personnel breathing zone air sample for radioactive particulates and from tritium grab samples. These data were taken in the containment building during the first quarter of 1984. Results were provided for containment building concentration and worker maximum permissible concentration (MPC), both in units of microcuries per cubic centimeter: (●) cesium-137, 8.3×10^{-8} (concentration), 1×10^{-8} (MPC); (●) cesium-134, 3.9×10^{-9} (concentration), 1×10^{-8} (MPC); (●) strontium-90, 2.5×0^{-9} (concentration), 1×10^{-9} (MPC); and (●) tritium, 1.0×10^{-7} (concentration), 5×10^{-6} (MPC).

The planned activities were not expected to increase the tritium or particulate levels inside the containment building. An additional release of tritium to the containment building atmosphere due to evaporation of the reactor coolant was not expected to increase the tritium level in the containment building atmosphere because of the low tritium concentration in the reactor coolant of about 0.03 microcurie per milliliter. The release of additional particulates to the containment building atmosphere could result from water droplets falling from retrieved tools and becoming airborne or the drying out of residue upon the inner diameter of the internals indexing fixture (IIF). The speed of the upward airflow associated with the evaporative process of the water was expected to be small so that the water droplets, liberated from the tools, would fall back into the reactor vessel water rather than becoming airborne. Also, the presence of water vapor in the air space above the water would minimize the drying out of any residue on the walls of the IIF and the associated particulate release.

- ***Evaluation: Radiation Protection/ALARA***. [66] The licensee's safety evaluation considered external and internal exposure controls, as well as exposure reduction measures.

  - *External Exposure Control*. All individuals entering the containment building would be monitored for external exposures in accordance with radiological control procedures to ensure that personnel exposures were maintained ALARA and within 10 CFR Part 20 dose equivalent limits. Administrative control points in accordance with the procedures would be used to ensure that specified dose limits were not exceeded. Extremity monitoring would be performed as needed in accordance with existing procedures. Personnel from the radiological controls department would continuously monitor dose rates in the containment building during these preparatory activities for plenum assembly removal.

  - *Internal Exposure Control*. All individuals entering the containment building would be monitored for internal radiation exposures according to established procedures. This monitoring would be done either by periodic whole-body counting or bioassay, or both. All exposures to airborne radioactivity would be maintained ALARA and within the limits established in 10 CFR Part 20. Airborne radioactivity in work areas would be monitored according to established procedures. Air sampling for particulates would be performed using devices such as breathing zone air samplers and grab samples. Tritium grab samples would be taken, as required, according to established procedures.

  - *Exposure Reduction*. The objective of minimizing occupational exposure to ALARA levels was a major goal in the planning and preparation for all activities in the containment building. These measures would minimize the time personnel must work in radiation fields, maximize the distance between personnel and radiation sources to the extent practicable, and use shielding where appropriate to meet the ALARA objective. Measures could include the following:

    – *Protective Clothing and Respirators*. Protective clothing and respirators would be used as necessary to reduce the potential for external contamination and internal exposure of personnel.

    – *Mockup Training*. Exposures while executing individual tasks were maintained ALARA by a detailed radiological review and mockup training. The need for the mockup training would be determined on a case-by-case basis. The degree of difficulty and newness of the operation would influence the need for mockup training and the level of its detail. This training would approximate the actual work situation for each task as closely as possible, using appropriate equipment, protective clothing, and respiratory protection.

– *Tool Design*. Inspection tooling was designed with the intent of keeping radiation exposures ALARA. These design features included the following: (●) Counterbalances would be used to allow one operator to raise or lower tooling with no heavy exertion. (●) The camera positioning cable and draw wire would be routed inside the tool handles to prevent tangling or severing of the cable or the draw wire. (●) There would be individual tool handles for each tool end to avoid interchanging tool handles inside containment. (●) Tools would be assembled outside of containment except for attaching tool ends to tool handles and the final routing of the cable. (●) Locking the tool handle to the tool end and the cable routing with one pin was an easy onetime action. (●) Flow holes existed in the tool ends to allow flooding of the hollow tooling so that buoyancy effects were minimized, and radiation streaming was prevented. (●) For ease of decontamination, the tooling was fabricated of stainless steel and aluminum with smoothed inside and outside surfaces and no sharp edges or protrusions. (●) Flushing and draining holes were provided with flushing capability from the top of the tooling.

– *Planning and Training*. Extensive planning of tasks to be conducted in a radiation field and training of personnel would be used to reduce the time needed to complete a task. Photographs and the containment building video system would be used extensively to familiarize personnel with the work area. The higher radiation areas were identified to personnel, and the work was structured to avoid these areas to the extent practical. Practice sessions would be held as necessary to ensure that personnel understood their assignments before entering the containment. Planning and training were proven methods of ensuring that personnel were properly prepared to expeditiously conduct the assigned task.

- ***NRC Review***. [67, 68] Editor's Note: The NRC issued two safety evaluation reports (SERs); the first SER covered the first five activities (see "Purpose" above), and the second SER covered the remaining three activities. The first SER was brief and did not specifically address this chapter's safety topic. However, the first safety evaluation stated that the first five activities were previously addressed in the NRC's SERs [69, 70] for Quick Look video inspection of the reactor core and for the reactor vessel underhead characterization. The NRC concluded that the licensee's experience with conducting core and plenum video inspections and other in-vessel activities (e.g., radiation measurements, reactor coolant sampling) demonstrated that these were benign activities (i.e., environmental impacts were very small), which posed little risk to the onsite workers or offsite public. The NRC further stated that the corresponding plenum inspection activities did not warrant further review.

- ***NRC Review: Occupational Exposure***. [71] The NRC's second safety evaluation of the three remaining activities concluded that the licensee's occupational exposure estimates for plenum removal preparatory activities fell within the scope of those previously assessed in the PEIS.

The NRC's conclusion was based on the following: (●) Dose rates in the containment building did not change appreciably following the head lift and the placement of the IIF and shield cover.

(●) Continued dose reduction efforts (i.e., scrabbling of floor surfaces) were underway to further reduce exposure. (●) The licensee established that the total collective exposure to workers during the proposed activities would be in the range of 30 to 165 person-rem. This estimate was based on 1035 in-containment person-hours and did not take credit for the recent dose reduction efforts (i.e., scrabbling).

- ***NRC Review: Radiation Protection/ALARA***. [72] The NRC's second safety evaluation of the remaining activities noted that the licensee would implement appropriate measures to keep worker exposures ALARA during plenum removal preparatory activities.

These measures would include the following: (●) Shielding of the contaminated plenum was provided by 1-inch-thick lead plates on the IIF work platform and by 5 feet of water in the IIF. (●) The IIF processing system would be operated to limit activity levels in the reactor coolant, thereby limiting the dose contribution of that source. (●) The IIF ventilation/filtration system would be available to remove any airborne activity in the IIF air space resulting from the proposed activities. (●) Detailed planning and personnel training would be conducted to reduce the time needed for completion of identified tasks. (●) Mockup training would be used extensively, and the actual work conditions would be closely modeled to familiarize workers with their assigned tasks. (●) Practice sessions would be held to instruct workers in the use of the long-handled tools, which were designed to allow easy operation. (●) Higher radiation areas would be identified, and work would be planned to avoid those areas as much as possible. (●) Use of respirators would be reviewed for each task to ensure that worker exposures were kept ALARA. This review would include examinations of current radiological conditions, the potential for perturbation of those conditions, and previous airborne activity measures. (●) Detailed exposure estimates would be developed on a task-by-task basis as a normal part of ALARA review of work in the containment to ensure that each activity was performed while minimizing worker exposure. (●) Dose rates in the containment building would be continuously monitored during the proposed activities; administrative control points would be established to ensure that specified dose limits would not be exceeded.

9.4.7 Plenum Assembly Removal

- ***Purpose***. To remove the plenum assembly from the reactor vessel to gain access to the core region for defueling. The plenum was moved to the deep end of the fuel transfer canal, which contained reactor coolant for shielding.

- ***Evaluation: Occupational Exposure (External Dose)***. [73] The licensee's safety evaluation considered collective exposure and dose rate estimates.

  - *Collective Exposure*. The safety evaluation estimated the collective radiation exposure to workers during the lift and transfer activities of the plenum assembly and during the support activities in the containment building. The estimate was based on projected person-hour requirements and containment building exposure rates associated with these activities.

- *Results*. The collective dose was estimated to be 30 person-rem. This value was based on 500 in-containment person-hours. Person-rem for radiological control support was not included in the estimate noted above. From a review of historical data, the evaluation assumed that person-rem for the radiological control group would be 20 percent of that accumulated by other groups in the containment building. Based on this, the estimate for radiological control support was 6 person-rem, and the total for all groups was estimated at 36 person-rem.

- *Uncertainty*. Because of the uncertainty in the person-hour estimate and the radiological conditions that would exist during the inspection, lift, and transfer activities, the evaluation estimated that the total exposure could vary by up to ± 30 percent. Considering these uncertainties, 25 to 50 person-rem was selected to be used as the estimate for the performance of the activities within the scope of this safety evaluation report, including radiological control support. Detailed exposure estimates would be developed on a task-by-task basis as a normal part of ALARA review of work in the containment building. This would ensure that each activity was performed to reduce collective worker exposure.

o *Dose Rates*. Conservative calculations estimated dose rates in air as a function of distance from the side of the plenum assembly. Radiation measurements showed that actual dose rates were less than those calculated. However, the following results from the conservative calculations (dose rate at a distance from the plenum assembly) were used for contingency planning: (●) 120 roentgens per hour at 3 feet; (●) 80 roentgens per hour at 6 feet; (●) 16 roentgens per hour at 20 feet; and (●) 0.86 roentgen per hour at 100 feet.

- ***Evaluation: Occupational Exposure (Internal Dose)***. [74] The licensee's safety evaluation stated that an estimate of airborne radioactivity encountered by individuals performing the initial lift activities was derived from the personnel breathing zone air samples for radioactive particulates and tritium grab samples taken in the containment building before, during, and after reactor vessel head removal. These containment building concentrations in microcuries per cubic centimeter were: (●) cesium-137 of 2.5×10^{-9}; (●) cesium-134 of 6.8×10^{-11}; (●) strontium-90 of 2.3×10^{-11}; and (●) tritium of 1.1×10^{-7}.

For soluble cesium-137, the internal dose was 2.5 millirem (received over several years; effective half-life is 70 days) for each hour of exposure at the maximum permissible concentration (MPC). For soluble strontium/yttrium-90, the bone dose was about 15 millirem (received over 50 years; effective half-life is 6400 days) for each hour of exposure at MPC. Even if there was no overall savings in the total dose due to elimination of a respirator for a given task (that is, the increased internal dose exactly offset the decreased external dose), the fact that the internal dose was calculated on a 50-year dose commitment, whereas external dose was deposited instantly, means that the rate of dose deposition was reduced overall.

The planned activities were not expected to increase the tritium or particulate levels inside the containment building. An additional release of tritium to the containment building atmosphere due to evaporation of the reactor coolant was not expected to increase the tritium level in the containment building atmosphere because of the low tritium concentration in the reactor coolant (about 0.03 microcurie per milliliter). The release of additional particulates to the containment building atmosphere could result from water droplets falling from the plenum assembly and potentially drying out. Because of the short time that the plenum assembly would be exposed, the quantity of particulates potentially becoming airborne was not expected to increase the particulate concentration in the containment building atmosphere.

- ***Evaluation: Radiation Protection/ALARA***. [75] The licensee's safety evaluation considered external and internal exposure controls, as well as exposure reduction measures.

- *External Exposure Control*. All individuals entering the containment building would be monitored for external exposures in accordance with radiological control procedures to ensure that personnel exposures were maintained ALARA and within 10 CFR Part 20 dose equivalent limits. Administrative control points, in accordance with the procedures, would be used to ensure that specified dose limits were not exceeded. Extremity monitoring would be performed as needed in accordance with existing procedures. Personnel from the radiological controls department would continuously monitor dose rates in the containment building during the plenum assembly lift and support activities.

  The increase in radiation dose field inside the containment building resulting from the dry lift and transfer of the plenum assembly was not expected to increase the estimated personnel exposure. This was because a minimum number of personnel would be inside the containment building during the lift and transfer operation. In addition, those personnel would control and monitor the lift and transfer operation from within the lead curtain shielded area above the pressurizer missile shields. A direct line-of-sight path between the plenum assembly and workers in the containment building was planned only when the plenum assembly was submerged in water.

- *Internal Exposure Control*. All individuals entering the containment building would be monitored for internal radiation exposures according to established procedures. This monitoring would be done by either periodic whole-body counting or bioassay, or both. All exposures to airborne radioactivity would be maintained ALARA and within the limits established in 10 CFR Part 20. Airborne radioactivity in work areas would be monitored according to established procedures. Air sampling for particulates would be performed using devices such as breathing zone air samplers and grab samples. Tritium grab samples would be taken as required by established procedures.

  To minimize the uptake and deposition of airborne radioactivity in the body, respiratory protection would be used. By reducing uptakes of radioactive materials, the use of respiratory protection devices could result in overall dose savings (internal and external); however, if the respiratory gear impeded work, total dose could increase by causing an elevated external dose. Current radiation protection guidance as expressed in International Commission on Radiological Protection Publication 26 considered both external and internal sources of dose and recommended minimizing their sum.

The radiological controls department, via the prework radiological review process, would determine if the use of respiratory devices for a task was necessary to achieve the ALARA objective. This review examined: (●) current radiological conditions in the work area; (●) the potential of the task or other concurrent tasks to perturb the radiological conditions; and (●) when available, results of previous airborne activity measurements in the work area for similar tasks.

- *Exposure Reduction*. Minimizing occupational exposure was a major goal in the planning and preparation for all activities in the containment building. These measures would minimize the time personnel must work in radiation fields, maximize the distance between personnel and radiation sources to the extent practicable, and use shielding where appropriate to meet the ALARA objective. Measures could include the following:

  - *Protective Clothing and Respirators*. Protective clothing and respirators would be used as necessary to reduce the potential for external contamination and internal exposure of personnel.

  - *Mockup Training*. Exposures while executing individual tasks were maintained ALARA through a detailed radiological review and mockup training. The need for the mockup training would be determined on a case-by-case basis. A mockup would be used to simulate the three-point lift and transfer, including a simulated plenum cover with appropriate lift ribs, a simulated internals handling fixture (tripod), and the actual pendant assemblies. Extensive training of workers on the mockup would familiarize them with the tasks to be performed. This training would result in less time in the containment building and reduced worker exposure.

  - *Equipment Design*. Equipment was designed with the intent of keeping radiation exposures ALARA by minimizing assembly in the containment building, amplifying operation, and having remote operation capability. The lifting arm assemblies attached to the bottom of the pendant assembly were self-latching and, once unloaded, could be detached remotely. The load positioners could be remotely operated from about 200 feet.

  - *Remote Cameras*. The extensive use of video equipment to monitor the lift and the levelness of the lift allowed workers not to be in the line-of-sight of the plenum assembly while the plenum was being transferred from the reactor vessel to the deep end of the fuel transfer canal.

  - *Distance and Shielding*. Saving time spent in the containment building was also realized by simplifying the communication and control required during the lifting operations with a central area for monitoring and control. This central area was the same shielded area located above the pressurizer missile shields that was used during the reactor vessel head lift operations.

- ***NRC Review: Occupational Exposure***. [76] The NRC's safety evaluation considered collective exposure and dose rate.

  - *Collective Exposure*. The licensee estimated a collective worker exposure of 25 to 50 person-rem for the plenum lift and transfer. This estimate was consistent with the NRC's estimated worker exposure for this activity as detailed in the PEIS, and the NRC considered the estimate to be reasonable.

  - *Dose Rate (Stuck Plenum Assembly)*. The NRC and the licensee calculated the dose rates at various distances from the unshielded plenum assembly. If the plenum assembly became stuck during the transfer, dose calculations indicated that personnel would be able to exit the containment building without significantly large exposures (about 10 millirem per person depending on the plenum assembly stuck position). In addition, calculations indicated that at locations such as the polar crane, the spider lift device would permit personnel access for corrective actions (about 50 millirem per person depending on the stuck position and time required for corrective action). The licensee would verify dose estimates by actual surveys. If potential doses to personnel were estimated to be significantly higher than anticipated, the option was available to flood the fuel transfer canal to the normal refueling level in order to shield the plenum assembly.

  - *Conclusion*. Based on this review, the NRC concurred with the licensee's estimate of worker exposure during plenum assembly lift and transfer and found that the measures adopted to maintain worker exposures ALARA were acceptable.

- ***NRC Review: Radiation Protection/ALARA***. [77] The NRC's safety evaluation concluded that the licensee would implement appropriate measures to keep worker exposures ALARA during plenum lift and transfer activities.

The ALARA measures would include the following: (●) use of administrative control points to limit worker doses and continuous monitoring of containment building dose rates during plenum assembly lift and transfer; (●) radiological reviews before execution of individual tasks to determine requirements for protective clothing, respirators, and shielding; (●) mockup testing of various activities to familiarize workers with specific tasks in order to reduce their time spent in radiation fields; (●) design of plenum assembly lift rigging to permit remote operation of the load positioners and to simplify operation of the lifting arm assembly latching/unlatching devices; (●) submergence of the plenum assembly under several feet of water during latching before the plenum assembly lift and unlatching following the transfer; (●) rigging personnel working at the internals indexing fixture platform area with dose rates close to the ambient radiation levels in the containment building (less than 35 millirem per hour); (●) remote performance of the actual lift and transfer, with the plenum assembly out of water coverage; and (●) control and monitoring of the lift and transfer operation from within the lead curtain shield area.

9.4.8 Makeup and Purification Demineralizer Resin Sampling

- ***Purpose***. To obtain resin samples from the two makeup and purification demineralizers. Resin samples were required to characterize the present resin conditions for the development of a technically sound resin removal and disposal program.

- ***Evaluation: Occupational Exposure***. [78] The licensee's safety evaluation stated that portable shielding and distance from the loaded sampler would be used to minimize exposures to personnel. The small mechanical probe required several operations to obtain the necessary sample quantity and would provide some radiation control for each operation. Estimates of the dose rates expected on the sample bottle shield were 530 millirem per hour using a source term of 5000 microcuries per gram for specific activity of the resin and a 100-gram sample limit.

- ***NRC Review***. Editor's Note: The NRC's safety evaluation was not located.

9.4.9 Makeup and Purification Demineralizer Cesium Elution

- ***Purpose***. To remove most of the radioactivity from the resins while they were in the demineralizers to the extent that standard resin sluice procedures could complete the task. The scope of this evaluation included only the first phase of a three-phase process for disposition of the makeup and purification of resins. This first phase included the rinse and elution of the demineralizer resins. The latter two phases would address the sluicing, removal, solidification or other packaging, and disposal of these resins. Separate safety evaluations would address the latter phases.

- ***Evaluation: Radiation Protection/ALARA***. [79] The licensee's safety evaluation stated that all personnel performing work used every means available to maintain their radiation exposure ALARA.

ALARA measures included the following: (●) Radiation control personnel would monitor work areas as required and provide dose rate information to aid individuals in performance of tasks, as far as radiological work practices were concerned. (●) Extensive planning of tasks to be conducted in a radiation field and training of personnel could reduce the time needed to complete a task. (●) Higher radiation areas would be identified to personnel and shielded, where practicable. (●) Work was structured to avoid these areas to the extent possible. This included locating the control panel behind an existing shield wall. (●) The chemical addition skid would be located in the valve alley corridor to minimize exposure during the chemical mixing process. (●) Worker accessibility for processing of the purification demineralizers would be limited to the gas analyzer room located immediately north of the demineralizer cubicles. The gas analyzer room was decontaminated to the extent that processing personnel would not require respirators. (●) Portable radiation monitors would be used to detect radiation levels in the demineralizer cubicles, in the filtering process equipment, and in general work areas. Radiological information and relevant data from within the demineralizer cubicles would be obtained remotely.

In addition, demineralizer rinse effluent would be diluted. Curie levels for initial rinse of the demineralizers were anticipated to be as high as 1350 microcuries per milliliter based on sample results from Oak Ridge National Laboratory. Since the levels required for reasonable management through normal waste processing (i.e., to the neutralizer tanks and the submerged demineralizer system) was 70 microcuries per milliliter, the demineralizer rinse effluent would be diluted as it passed through the eductor or pump. The first batch would be diluted by 20:1, and subsequent batches would be adjusted to maintain levels so it would not exceed 70 microcuries per milliliter.

- ***Evaluation: Shielding***. [80] The licensee's safety evaluation noted that the system used for elution consisted of four equipment skids that would interface with existing plant systems. They included: (●) the chemical mix tank and fill pump skid; (●) the dilution eductor and transfer pump skid (one skid for each demineralizer), and (●) the back flushable stainless-steel filter skid. Various structures, components, and piping/hose segments were shielded to reduce radiation exposure.

  - *Piping*. Process flow piping into the gas analyzer room would be shielded to maintain the workstation levels at or below 2.5 milliroentgens per hour. The lines from the chemical mix tank into the demineralizer would not be shielded since only processed water would be in those lines. Inadvertent backflow could occur only with the simultaneous failure of two check valves with other valve leak/failures.

  - *Eductor/Transfer Pump Skids*. Both dilution eductor and transfer pump skids would be shielded to maintain workstation radiation levels at or below 2.5 milliroentgens per hour. The reach rods of the process flow valves penetrated the shielding. These skids were supported by structural support members that were anchored to the demineralizer cubicle walls and floor. These structural supports also held shielding for interconnecting piping.

  - *Back Flushable Filter Skid*. The back flushable filters utilized a steel shielding design. The system's design provided backflushing for the filters with nitrogen and rinsing with processed water to flush the collected radionuclides back to the demineralizer vessels.

  - *Filter Effluents*. Effluents from the filters were directed through a shielded water-meter assembly and then to shielded piping that led to the valve alley corridor. The effluent would then be delivered to the neutralizer tanks using rubber hosing from the corridor. The hose would be prefabricated in 50-foot sections with factory-installed, locking quick-disconnect fittings to facilitate disposal. The fittings would be wrapped in plastic using existing plant contamination control practices. This hose would be shielded to maintain acceptable radiation levels.

  - *Neutralizer Tanks*. The final destination of the process flow was the neutralizer tanks. The tanks were shielded by concrete walls and were access controlled by radiological control procedures. Introduction of 70-microcurie-per-milliliter water in the neutralizer tanks would not exceed the design criteria of the shielding walls of that cubicle.

- ***NRC Review: Occupational Exposure***. [81] The total collective occupational dose for this operation was estimated to be 4 person-rem, which included system installation, operator training, and operation.

- ***NRC Review: Radiation Protection/ALARA***. [82] The NRC reviewed all radiological controls and concluded that the following ALARA measures were adequate: (●) Adequate monitors and controls existed to ensure acceptable radiation levels in all accessible areas during normal operations. (●) Except for the preoperation installation and alignment, operation of the elution process would be carried out behind shield walls at the system control panel where radiation levels were expected to be about 2.5 millirem per hour. (●) Process water would be diluted. (●) Ventilation control would be provided. (●) Training programs would be conducted for all operators involved to ensure their familiarity with the system, instrumentation, operating principles, and expected radiological conditions.

9.5 Defueling Tools and Systems

9.5.1 Internals Indexing Fixture Water Processing System

- ***Purpose***. To provide reactor coolant water processing capability during head removal until plenum removal. The internals indexing fixture (IIF) processing system was selected based on the limited reactor coolant processing capability in the drained-down condition and the desire to provide adequate water cleanup capacity to minimize radiation dose rates around the IIF.

- ***Evaluation: Occupational Exposure (External Dose)***. [83] The licensee's safety evaluation noted that the IIF processing system was designed to provide increased water processing capability during the period between head removal and plenum removal. This increased capability was required to reduce radioactive contaminants in the reactor coolant system (RCS) and thereby to reduce radiation dose rates for workers on and around the IIF. The RCS level in the IIF would be maintained above the control rod guide tubes to provide adequate shielding of the strong plenum source. Reducing the radioactivity in the water would further reduce dose rates for workers on the IIF.

    - *Source Term*. Without RCS processing, the concentration of radioactive contaminants in the coolant would increase. The concentration of cesium-137 would reach levels of 6 to 8 microcuries per milliliter after several months without processing with the RCS in the drained-down condition. The goal was to have reactor coolant concentrations at about 1 microcurie per milliliter at the time of head removal. Based on the capability of the submerged demineralizer system with the IIF processing system operating at 10 gallons per minute, reactor coolant concentrations of 0.1 microcurie per milliliter and lower could be achieved. This was based on system availability of 70 percent that accounted for bleed tank recirculation and sample time.

Calculated dose rates did not consider any isotope in the reactor coolant except cesium-137. Other isotopes that could contribute significantly to gamma dose rates included cesium-134 and antimony-125. The cesium-134 concentration was normally an order of magnitude less than that of cesium-137. In its current configuration, the submerged demineralizer system did not remove antimony-125 from the coolant with a reliable decontamination factor. However, the dose rate for antimony-125 was less than that of cesium-137 for a given concentration. Before the operation of the IIF processing system, concentrations of antimony-125 were about 0.2 microcurie per milliliter. In addition, if antimony-125 in the reactor coolant became a significant dose contributor to workers on the IIF cover, the IIF processing system could be used to transfer water to the EPICOR II system in a batch processing mode. EPICOR II would remove the antimony-125 with a satisfactory decontamination factor.

- *Dose Rates (with IIF Shielding)*. Calculated radiation dose rates were used here to illustrate the benefits of reducing the radioactive materials in the coolant. Based on data from the underhead characterization program, the anticipated dose rate contribution from the plenum and the reactor coolant in the IIF was calculated. The dose rate increase for workers on the IIF cover was calculated assuming that the IIF cover held 1-inch-thick lead shielding and that the water level in the IIF was 5 feet above the vessel flange. Note that decreasing the water level in the IIF would decrease the shielding of the plenum, thus increasing the dose rate. Increases in general area dose rates were calculated for the time before the IIF processing system was available (with a reactor coolant concentration of 1 microcurie per milliliter) and after IIF processing was operating (with a reactor coolant concentration of 0.1 microcurie per milliliter). The contribution from the plenum was based on a previous licensee's analysis of the dose modeling of reactor vessel underhead radiation sources. These dose rates excluding background on the IIF cover were 5 millirem per hour for 0.1 microcurie per milliliter and 16 millirem per hour for 1.0 microcurie per milliliter.

- *Dose Rates (without IIF Shielding)*. During plenum inspection tasks, workers could remove shielding plates from the IIF cover to provide tool and camera access into the vessel. Dose rates calculated assuming no shielding from the IIF cover were 120 millirem per hour for 0.1 microcurie per milliliter and 610 millirem per hour for 1.0 microcurie per milliliter.

- *Dose Rates (Containment Background)*. The entire IIF processing system was examined for radiological impact to the containment building environment. The long hose lengths that carried reactor coolant through the containment building were evaluated to determine if this new source had the potential to increase general area dose rates. The dose rate calculations showed that the increase in worker exposures due to the hose would be negligible. For cesium-137 concentrations of 0.1 and 1.0 microcurie per milliliter in the reactor coolant, the resulting dose rate 4 feet from the hose (excluding background) was 0.1 and 1.0 millirem per hour, respectively.

- ***Evaluation: Occupational Exposure (Internal Dose)***. [84] The licensee's safety evaluation noted that the dose assessment for operating the submerged demineralizer system, in conjunction with the IIF processing system, was given in Appendix 2 to the technical evaluation report [85] for the submerged demineralizer system. The assessment showed that no increase was expected in the release of airborne radioactivity to the containment building atmosphere or from the containment building to the environment from use of the IIF processing system. In fact, the IIF processing system would help reduce the reactor coolant radioactive materials concentration, which should minimize any airborne radioactivity release from the RCS in the containment building.

- ***Evaluation: Radiation Protection/ALARA***. [86] The licensee's safety evaluation stated that a major goal in the design of the IIF processing system was to ensure that radiation exposure to workers was maintained ALARA. The IIF processing system would reduce radioactive materials in the reactor coolant and thereby reduce radiation dose rates for workers on and around the IIF. ALARA considerations included radiological reviews and design features.

  - *Radiological Reviews*. All work performed in the containment building was reviewed by the radiological controls department and was evaluated to ensure that personnel radiation exposures were minimized according to established procedures.

  - *ALARA Design Features*. Specific design features for the IIF processing system were incorporated to implement the ALARA concept. These features are summarized as follows: (●) The IIF pump was designed to be installed with the IIF immediately after head removal. This eliminated the need for personnel access to the IIF cover before the reactor coolant cleanup. (●) The IIF pump was a commercially available model, which required short leadtime if replacement became necessary. (●) The IIF pump was equipped with lifting eyes and a single hold-down bolt to simplify removal or movement, which permitted operational flexibility. (●) Sufficient hose to the pump was provided to allow relocation of the pump if activities after the removal of the head required such a move. (●) The impact of containment building dose rates due to the hose routing was evaluated to ensure minimal increase in the general area dose rates. The hoses would be routed to maximize distance to work areas and to take advantage of existing structures for shielding as much as possible. (●) Failures of the IIF processing system components would not result in unacceptable radiological conditions for workers in the containment building. (●) Work in the containment building required for the operation of the IIF processing system involved hose disconnections at the fuel transfer canal drain manifold. Realignment of the flowpath to the SDS was expected to be infrequent. (●) All hose connections were equipped with quick disconnect fittings to facilitate connection and disconnection, which would minimize the time required for adjustments.

- ***NRC Review: Radiation Protection/ALARA***. [87] The NRC's safety evaluation stated that the IIF processing system was designed to maintain the radionuclide concentration in the RCS at a maximum of 0.1 microcurie per milliliter. The IIF cover, which included 1 inch of lead shielding, would provide additional radiation protection. The combined effect of these two features was expected to reduce the dose rates above the IIF to near ambient levels. The NRC concluded that the proposed IIF processing system could be operated without posing a significant risk to the occupational workforce or the offsite public. Additionally, the potential environmental impacts resulting from system operation would be minimal.

9.5.2 Defueling Water Cleanup

9.5.2.1 Defueling Water Cleanup System

- ***Purpose***. To remove organic carbon, radioactive ions, and particulate matter from the reactor vessel and fuel transfer canal/spent fuel pool "A" (FTC/SFP-A). The defueling water cleanup system (DWCS) actually comprised two independent subsystems (sometimes called trains): the reactor vessel cleanup system and the FTC/SFP-A cleanup system. The DWCS was totally contained within areas that had controlled ventilation and could be isolated. This limited the environmental impact of the system during normal system operations, shutdown, or postulated accident conditions.

- ***Evaluation: Occupational Exposure***. [88] The licensee's safety evaluations considered onsite doses during the operation of the reactor vessel and the FTC/SFP-A cleanup systems.

  - *Reactor Vessel Cleanup System*. The evaluation considered sources of radiation, dose rates under normal and abnormal operations, and shielding requirements.

    – *Source Term*. There was the potential for a significant increase in the specific activity of the reactor vessel water during defueling, with the disturbance of core debris. Material greater than nominal 0.5 microns would be captured in the system filters. The soluble fission products, particularly cesium-137, would be removed by processing through the associated ion exchange media. The filter canisters were located underwater at a depth greater than 4 feet in the containment building, thus not representing a radiological problem. The licensee did not expect that significant quantities of fuel would accumulate in the post filters. Consequently, the post filters were not expected to be a radiological hazard. However, if the dose rates from these filters began to increase, appropriate measures (e.g., shielding, personnel relocation) would be taken to ensure acceptable dose rates to personnel. The water to be processed was piped through a containment building penetration to the ion exchange media at 20 to 60 gallons per minute (maximum 30 gallons per minute per train), depending on the specific activity of the reactor vessel water. These process lines and the liners for the ion exchange media represented potential radiological hazards.

    Dose rates from soluble materials were based on the specific activity of cesium-137. Other isotopes that could contribute significantly to gamma dose rates were cesium-134 and antimony-125. The cesium-134 concentration was normally an order of magnitude less than that of cesium-137. Antimony-125 was not removed by the DWCS ion exchangers with a reliable decontamination factor. However, the dose rate for antimony-125 was less than that of cesium-137 for a given concentration. If antimony-125 in the DWCS became a significant dose contributor to workers, the reactor coolant could be processed through the EPICOR II system in a batch processing mode.

Batch processing would be used because chemical adjustment of the coolant was required. EPICOR II would remove the antimony-125 with a satisfactory decontamination factor.

- *Dose Rate (Normal Operations)*. To assess and evaluate the radiological hazards, the dose rates from DWCS piping and components during operation were evaluated. Sources in the water were assumed to be fuel particles and dissolved radioactive materials. The design basis concentrations of these sources were 1 part per million suspended solids and a concentration of soluble materials equivalent in dose rate to 0.02 microcurie per milliliter of cesium-137. During operation at the design-basis concentrations, the dose rate from a long 3-inch diameter unshielded hose was 0.2 millirem per hour at a distance of 2 feet.

- *Exposure (Normal Operations)*. Operation of the DWCS reduced the occupational exposure during defueling operations by maintaining specific activities in the FTC/SFP-A and reactor vessel. The DWCS was designed to maintain the maximum cesium-137 concentration in the water to between 0.01 and 0.02 microcurie per milliliter. This would result in a contribution to general area dose rates of 10 to 20 millirem per hour from the water.

  The safety evaluation report (SER) provided tables (refer to SER pages 22 and 23) of the estimates for person-hours and person-rem associated with the installation, operation, maintenance, modification, and removal of the DWCS portions in the containment building and fuel handling building. These estimates were based on person-hour projections and dose rates ranging from 25 to 35 millirem per hour in the containment and 0.3 to 3 millirem per hour in the fuel handling building. The estimated total person-rem attributable to the operation, maintenance, modification, and removal of the DWCS was about 250 person-rem.

  The licensee estimated that about 42 4-foot by 4-foot liners, each loaded with 52 curies of cesium-137 would be required for the reactor vessel cleanup system. The occupational dose to workers during each changeout was estimated to be less than 0.1 person-rem. Therefore, the total accumulated dose for changeout of the estimated 42 4-foot by 4-foot liners was 4.2 person-rem.

  In the original revision of the SER [89]submitted in 1985, the estimated total dose ranged between 65 to 125 person-rem, which included a ± 30-percent range for uncertainties and a 20-percent increase for health physics coverage. This lower overall estimate was based on lower in-containment person-hours than were projected in the original estimate. (Problems encountered with reactor vessel water clarity during initial DWCS operations resulted in more in-containment hours for system modifications.) As of December 31, 1988, installation, operation, and maintenance of the DWCS resulted in about 200 person-rem.

- *Dose Rate (Abnormal Conditions)*. The evaluation considered increases in dose rates during an upset water condition and a pipe/hose break.

  - *Upset Water Condition*. During defueling operations, both the concentrations of soluble and suspended solids in the reactor coolant could increase. To assess increases in dose rates during upset water conditions, a combination of a 20-curie cesium-137 spike and an instantaneous release of about 35 pounds of suspended fine debris to the reactor vessel volume was postulated. A long 3-inch-diameter hose carrying water at the resulting concentrations would result in a dose rate of 9 millirem per hour at a distance of 2 feet from the hose. Process lines, which were downstream of the filters, did not contain the concentrations of suspended solids postulated for the upset water conditions. A 3-inch-diameter hose downstream of the filters would produce a dose rate of 2 millirem per hour at a distance of 2 feet as the result of the remaining soluble radioactive materials in the water.

  - *Hose/Pipe Break*. If a hose or piping broke in the DWCS, water would be released in the containment building or the fuel handling building. This water could contain suspended fuel particles and dissolved radioactive materials. The specific activity of the DWCS water would be maintained as low as possible, so personnel access to the spill area would not always be precluded. However, a DWCS reactor coolant system spill could present personnel access problems due to high beta dose rates. After the removal of the spilled water, the area could require decontamination to reduce loose surface contamination to acceptable levels. The SER concluded that there were no safety concerns associated with the breakage of DWCS hoses or pipes.

- *FTC/SFP-A Cleanup System*. The fuel transfer canal/spent fuel pool cleanup system processed water through a DWCS ion exchanger. The water in the pools would be maintained by this system at 0.01 to 0.02 microcurie per milliliter of equivalent cesium-137, excluding antimony. A flowpath to EPICOR II through the reactor coolant bleed tanks was provided to remove antimony-125 in the event that high antimony-125 levels were encountered. These were significantly lower concentrations than water processed by the submerged demineralizer system. The analysis in this SER for normal dose rates from the reactor vessel cleanup system bounded the dose rates from the FTC/SFP-A cleanup system during normal operation. The accident analysis in the technical evaluation report [90] for the submerged demineralizer system bounded the possible doses from the FTC/SFP-A cleanup system in the event of an accident.

- ***Shielding (Dose Rate)***. [91] The licensee's safety evaluation stated that shielding requirements for the DWCS liners were based on a homogenized 500-curie source in a 4-foot by 4-foot liner, similar in construction to those used for EPICOR II. (Three zeolite ion exchangers were needed to handle the flow from the DWCS. Two were needed for the reactor vessel cleanup system to provide a flow rate of 60 gallons per minute through the ion exchangers. One ion exchanger was used for FTC/SFP-A cleanup.) Since the changeout of liners would be based on radiation level, and the 500-curie loading was conservatively high (the actual loading should be about 100 curies), the calculated shielding requirement was considered acceptable.

The contact dose rate on the side of the liner for a homogenized 500-curie source was about 185 rem per hour. The liners were shielded to limit the shield contact dose rate beside and on top of the liner to a maximum of 5 millirem per hour. The concrete floor reduced the dose rates on lower elevations to less than 5 millirem per hour. Shielded dose rates represented an upper bound and would not pose any undue operational constraints if actually attained.

Shielding of lines upstream of the filters could be applied to reduce dose rates in areas of personnel occupancy. Other DWCS components that were not specifically discussed above would be shielded as necessary.

- ***NRC Review: Occupational Exposure***. [92] The NRC's safety evaluation stated that Revision 6 of the licensee's evaluation projected a total occupational dose commitment of less than 125 person-rem attributed to the DWCS. This included exposure resulting from construction, installation, operation, maintenance, and dismantling of the system. Procedural controls during these activities would ensure that personnel exposure was maintained ALARA. The NRC review of the licensee's estimate concluded that the estimate was based on a reasonable estimate of the person-hours needed for the task and a conservative radiation dose rate, as determined by the review of survey and exposure data from tasks already performed in the same working areas. The dose commitment due to system operations was based on reasonable estimates of the expected activity levels of the reactor coolant system and from experience gained from the submerged demineralizer system and other waste processing system operations. The projected occupational exposure was within the scope of considerations in the PEIS.

9.5.2.2 Cross-Connect to Reactor Vessel Cleanup System (NA)

9.5.2.3 Temporary Reactor Vessel Filtration System

- ***Purpose***. To restore and maintain the visibility in the reactor vessel to acceptable levels to ensure the continuation of the early defueling operations. Operation of the defueling water cleanup system (DWCS) revealed that a differential pressure across its filter canisters would increase rapidly as the result of microorganism growth in the reactor coolant. Consequently, the DWCS was able to process only a relatively small amount of reactor coolant before the maximum design pressure was reached and the filter canister had to be replaced. These developments created the need to design and operate a temporary filter system while a permanent program to control this phenomenon was being developed.

- ***Evaluation: Occupational Exposure***. The licensee's safety evaluations considered radiation doses from the normal operations and off-normal conditions of the temporary reactor vessel filtration system. The evaluations were revised as the system was modified to account for operational improvements. Refer to NUREG/KM Chapter 2 for a summary of revisions.

- *Normal Operations*. [93] The expected maximum radiation level at the external surface of the temporary reactor vessel filtration system filter housing, was 3 roentgens per hour. Highly contaminated diatomaceous earth was transferred from the filter to 55-gallon drums. Shadow shielding was installed to localize the radiation source and minimize the exposure of operating personnel on the defueling platform. Radiation levels from the shielded 55-gallon drum were expected to be as much as 340 milliroentgens on contact. Shadow shielding was used to reduce the dose rate from this source to personnel working at the defueling slot to 1 or 2 milliroentgens per hour.

- *Filter Material Spill*. [94] Within a few months following the startup of the temporary reactor vessel filtration system, a defueling knockout canister was used to dispose of the diatomaceous earth filter residue, eliminating the use of 55-gallon drums. The safety evaluation report (SER) postulated a spill of diatomaceous earth during the transfer from the filter to a knockout canister. In this case, the transfer water, about 6 pounds of diatomaceous earth, along with the filtrate, were spilled onto the surface of the north end defueling work platform. Should such a spill occur, a portion of the platform would be contaminated with up to 2.1 curies of strontium/yttrium-90 and 0.1 curie of cesium-137. If the spill spread to cover a depth of 1/8 inch (3 millimeters), an area of about 500 square feet would be contaminated. Dose rates attributable to this contamination would be in the range of 1.2 rad per hour at 10 centimeters above the floor.

  The SER used a resuspension factor of $1x10^{-4}$ to estimate airborne radioactivity levels in the range of $3.1x10^{-7}$ microcurie per cubic centimeter. Since the majority of the dose consequences were from the spent diatomaceous earth, similar consequences would be obtained at higher levels of reactor coolant contamination. The spill was postulated during filter backwash, which was assumed to occur when the radiation level on the surface or the shield housing was 50 millirem per hour. At these levels, the local area airborne radioactivity monitors would cause alarm within 2 seconds of the spill. A 5-minute stay in this environment would result in 33 maximum permissible concentration-hours for the involved isotope, assuming no protection factor and equal distribution of airborne activity in the canal.

- *Liquid Spill*. [95] A liquid only spill was also considered. A pipe break at the pump discharge could potentially spill liquid from the internals indexing fixture (IIF) onto the 322-foot elevation of the fuel transfer canal floor. This event could be detected using the IIF level monitoring system. This liquid would drain to the sump of the canal floor on the south-east corner of the upper canal where liquid would collect and be pumped to a staging or processing location. With the suction limited to 2 feet below the surface of the water in the IIF, this represented about 4000 gallons of reactor coolant system water. The licensee did not expect that such an event would significantly increase the radiation exposure to workers on the defueling work platform.

- o *Dropped Canister*. Modified versions of the temporary reactor vessel filtration system [96,97] used knockout and filter canisters. The SER [98] for early defueling addressed the effects of a dropped canister on worker doses.

- ***NRC Review: Occupational Exposure***. [99] The NRC's safety evaluation noted that lead shielding would be provided for the filter assembly and the waste drum. In addition, shadow shields would be installed around the entire system. A gamma radiation detector with remote readout would continuously monitor the contact radiation level of the filter housing. This radiation level was currently limited to 3 roentgens per hour, whereas the dose rate to workers outside of the shields on the defueling platform was expected to be no more than 5 milliroentgens per hour. The only times an operator was required to enter the shadow shields were during initial startup of a recharged filter, to monitor filter differential pressure rise, during filter regeneration or replacement, and during waste drum transfer. During initial filter monitoring, the radiation levels were expected to be less than 10 millirem per hour. During filter regeneration or replacement, the shielding on the filter would reduce the radiation levels to less than 100 millirem per hour, and the expected dose to the operator was about 50 millirem. The expected dose to the operator during waste drum transfer was about 50 millirem. Based on these estimates, the NRC concluded that the projected occupational exposure was within the scope of considerations in Supplement 1 to the PEIS.

9.5.2.4 Filter-Aid Feed System and Use of Diatomaceous Earth as Feed Material (NA)

9.5.2.5 Use of Coagulants (NA)

9.5.2.6 Filter Canister Media Modification (NA)

9.5.2.7 Addition of a Biocide to the Reactor Coolant System

- ***Purpose***. To evaluate the effects of adding hydrogen peroxide as a biocide to the reactor coolant system (RCS) to aid the removal of biological contamination. This biological growth reduced water clarity and fouled the water processing system filters.

- ***Evaluation: Occupational Exposure***. [100] The licensee's safety evaluation stated that the RCS fission product activity was not expected to increase activities previously observed in the reactor coolant subsequent to head removal. Before the installation of the shielded work platform, dose rates above the internals indexing fixture were 150 millirem per hour and 15 millirem per hour on the shielded work platform. Furthermore, work on the defueling platform during and subsequent to the addition of hydrogen peroxide would follow the radiological control limitations found in the current defueling operations procedure.

- ***NRC Review: Occupational Exposure***. [101] The NRC's safety evaluation agreed with the licensee's assessment that the most significant effect expected on RCS chemistry was an increase in reactor coolant activity levels. An increase by a factor of 10 could occur but would not have a significant impact on worker safety with the implementation of normal radiological control practices.

9.5.3 Defueling Canisters and Operations

9.5.3.1 Defueling Canisters: Filter, Knockout, and Fuel

- ***Purpose***. To provide loading, handling, and storage of the canisters (filter, knockout, and fuel) for the long-term storage of core debris, ranging from very small fines to partial length fuel assemblies.

- ***Evaluation: Occupational Exposure***. (102) The licensee's safety evaluation noted that canisters were designed to be loaded with core debris from the TMI-2 reactor coolant system. These canisters did not contain internal shielding and must be shielded during all handling and storage operations. The shielding requirements for the various canister operations (e.g., loading, handling, and storage) were discussed in the safety evaluation report (103, 104) for early defueling of the reactor vessel. That safety evaluation report also discussed personnel exposure from the loaded canisters as part of the canister handling sequence.

- ***NRC Review***. (105, 106, 107) The NRC's safety evaluation reports did not specifically address these topics.

9.5.3.2 Replacement of Loaded Fuel Canister Head Gaskets

- ***Purpose***. To replace head gaskets on the loaded fuel canisters in spent fuel pool "A" (SFP-A) because of excessive leakage. The replacement of gaskets involved: (●) moving a loaded canister from the SFP-A storage rack to the dewatering station rack with the canister handling bridge; (●) removing the canister head; and (●) transferring the head to a worktable. At the worktable: (●) the head was rotated for access to the gaskets; (●) the metallic gaskets were removed; (●) new synthetic gaskets were installed; and (●) the head was inspected before reinstallation.

- ***Evaluation: Occupational Exposure***. (108) The licensee's safety evaluation stated that there the potential for fuel fines to collect inside and around the canister head's catalyst bed raised the possibility for personnel radiation exposure. Engineered processes and controls were used to minimize the need for respiratory protective equipment. Provisions to reduce direct radiation exposure dose rates to acceptable levels would also be implemented, as necessary. If flushing the canister head with water was required, a water source containing at least 4350 parts per million boron would be used. The degree of radiological hazard for the gasket replacement was commensurate with other work efforts in contaminated areas. Therefore, contamination controls necessary for gasket replacement operations would not need to be more rigorous than previously implemented control measures. The radiological controls department would monitor dose rates and airborne radioactivity levels during the gasket replacement operations. Based on the radiological conditions observed, the radiological controls department could establish shielding and other protective measures.

- ***Evaluation: Radiation Protection/ALARA***. [109] The licensee's safety evaluation stated that the removal of the canister head presented the potential for contamination spreading to the spent fuel pool water. Consequently, to minimize this potential, a temporary cover was placed on the canister once the head was removed. Any contamination released to the pool would be from the canister's free volume water or from fuel debris within the canister. The relatively small quantities of soluble radioactive materials in the water would not have a significant impact when diluted with the large volume of water in the spent fuel pool. Any fuel particles released to the pool would either settle out in the bottom of the pool or be entrained in the water. Water processing was used as required to maintain acceptable radioactivity concentrations in the pool.

- ***NRC Review: Radiation Protection/ALARA***. [110] The NRC's safety evaluation noted that a temporary cover would be placed over the canister when the head was removed to minimize the spread of contamination from the open canisters to the spent fuel pool water. The licensee's safety evaluation report also stated that water processing would be used as required to maintain acceptable radioactivity concentrations in the pool. The NRC determined that since the proposed activity presented the potential for contamination of water in SFP-A with transuranic material, additional precautions would be necessary after opening the loaded fuel canisters before water processing. More specifically, the NRC stated that the licensee would ensure that any SFP-A water processing was controlled by procedures after the opening of the loaded canisters. These procedures would have appropriate provisions to accurately assess the quantity of transuranic material in the processing stream and to ensure the proper disposal of any waste generated. The NRC concluded that appropriate engineering process controls and normal radiological control practices would be sufficient to preclude any undue radiological hazards.

9.5.3.3 Use of Debris Containers for Removing End Fittings (NA)

9.5.3.4 Fuel Canister Storage Racks

- ***Purpose***. To provide storage for the three different types of canisters (fuel, filter, and knockout) filled with debris material from the reactor vessel. Storage for 263 canisters was available in the racks located in spent fuel pool "A" and in the deep end of the fuel transfer canal.

- ***Evaluation: Occupational Exposure***. [111] The licensee's safety evaluation stated that the occupational exposure attributable to the fuel racks would be the dose accumulated during their installation in and removal from the containment building and fuel handling building.

- *Assumptions*. To calculate the person-rem associated with the installation and removal of the fuel canister storage racks (FCSRs), a general area dose rate of 35 milliroentgens per hour was used for in-containment person-hour estimates. This dose rate was based on the person-millirem per in-containment person-hour. A dose rate of 0.3 milliroentgens per hour was assumed in the person-rem estimates for the installation and removal of the FCSRs in the fuel handling building. This dose rate was based on the fuel handling building being decontaminated before the FCSR installation and the FCSRs being decontaminated before their removal, if required.

- *Dose Rates*. The safety evaluation report included tables (refer to page 14 of the report) that gave estimates of the person-hours and person-rem associated with the installation and removal of FCSRs in the containment building and fuel handling building. The person-hours for removal were estimated to be equal to 50 percent of those required for installation. These estimates were based on person-hour projections for work in the containment building. For example, the work estimate in the containment building for installation of the FCSR was 100 person-hours at a dose rate 35 milliroentgens per hour for a total exposure of 3.5 person-rem. Installation activity in the fuel handling building was 300 person-hours at a dose rate of 0.3 milliroentgens per hour for a total exposure of 3.5 person-rem.

  The total person-rem attributable to the installation and removal of the FCSR was expected to be between 4.5 and 8.5 person-rem. This estimate was based on a total of 5.4 person-rem from the above, increased by 20 percent for radiological control protection coverage, and adjusted by ± 30 percent due to uncertainties.

- ***NRC Review: Occupational Exposure***. (112) The NRC's safety evaluation stated that the only potential radiation exposure resulting from the FCSRs was the plant worker dose attributable to rack installation and subsequent removal. Procedural controls during installation and removal ensured that personnel exposure was maintained ALARA. The licensee projected a maximum exposure of 8.5 person-rem for this task. The NRC review concluded that this estimate had a reasonable basis for the person-hours needed for the task, and the conservative radiation dose rates were determined by a review of existing survey and exposure data from tasks already performed in the same work areas. The NRC concluded that the projected occupational exposure was within the scope of considerations in the PEIS.

9.5.3.5 Canister Handling and Preparation for Shipment

- ***Purpose***. To transfer defueling canisters from spent fuel pool "A" (SFP-A) to the shipping cask for offsite shipment. Defueling canisters were transferred to locations within the containment building and fuel handling building (FHB) using a transfer shield. The transfer of canisters to the shipping cask used a different device called a "fuel transfer cask."

- ***Evaluation: Occupational Exposure***. (113) The licensee's safety evaluation of worker exposure included external and internal exposures. External dose was the leading contributor.

- *External Exposure (Monitoring)*. All individuals entering the FHB were monitored for external exposures in accordance with radiological control procedures. All personnel exposure would be maintained ALARA, and all external radiation exposures would be within the exposure limits established in 10 CFR Part 20. Administrative control points in accordance with the procedures were used to ensure that specified dose limits were not exceeded. Extremity monitoring was performed as needed in accordance with existing procedures. Radiological controls department personnel monitored dose rates in the FHB during canister handling and preparation for shipment activities according to existing procedures.

- *External Exposure (Dose Rates)*. Estimates of radiological conditions during the activities were presented in the safety evaluation for information only. These estimates were based on hypothetical source terms for an average defueling canister that was fully loaded. The source terms used for defueling canisters were described in the licensee's safety evaluation report (SER) [114, 115] for bulk defueling. Radiological controls required during these activities were determined based on actual measured dose rates. During these activities, the evaluation assumed that personnel would be located on the dewatering station platform and along the edge of SFP-A. Note that these estimates did not include background radiation or sources not identified in the SER.

  - *Canister Transfer (from Storage Racks to Dewatering Station)*. The maximum dose rate on the dewatering system platform was expected to last for a very short time, when the canister transfer shield (CTS) collar was fully lowered in the pool water and the canister was drawn fully up in the collar. The average dose rate on the dewatering station platform during canister transfer operations was estimated to range from 50 to 100 millirem per hour depending on the location of the canister being transferred. Maximum dose rates expected at various locations included: (●) 5 millirem per hour at the canister handling bridge trolley; (●) 90 to 240 millirem per hour at the dewatering system platform (6 feet from the transfer shield); and (●) 5 to 16 millirem per hour at the 349-foot level.

  - *Canister Dewatering*. The canister dewatering platform was shielded with 2 inches of lead. Piping was routed below the platform. Reach rods were used to operate valves located under the platform. The platform was designed to maintain dose rates on the platform and on the fuel transfer cask loading station from the canister dewatering piping to about 2.5 millirem per hour.

  - *Canister Transfer (to Fuel Transfer Cask Loading Station)*. The fuel transfer cask loading station platform was shielded with 3 inches of lead. Access to the fuel transfer cask loading station while the canister was being raised or lowered was not expected. Dose rates during the transfer to the fuel transfer cask loading station included: (●) 5 millirem per hour at the canister handling bridge trolley; (●) 40 to 75 millirem per hour at the dewatering system platform; and (●) 16 millirem per hour at the 349-foot level.

  - *Canister Transfer (from Fuel Transfer Cask Loading Station to Shipping Cask)*. The fuel transfer cask was shielded with a total of 4.5 inches of lead and 2 inches of steel. The bottom door was 5-inch-thick lead or shielding equivalent. The maximum dose rates in the FHB would be less than 10 millirem per hour at a distance of 4 feet from the fuel transfer cask.

- *External Exposure (Whole Body)*. The total whole-body exposure anticipated for the activities described in this SER was calculated to be 184 person-rem assuming the shipment of 252 canisters. Additional assumptions included the following:

  – *Activity, Time, and Dose*. The total whole-body dose estimate was based on the following activities that required support from radiological control personnel: (●) shipping cask movement into FHB and preparation for pairing with a fuel transfer cask: 80 person-hours and negligible person-rem per seven canisters (one shipping cask load); (●) canister pre-dewater weight determination and transfer to dewatering station; dewater, post-dewater weight determination, and canister sample retrieval; canister load in fuel transfer cask; and fuel transfer cask transfer to truck bay: 20 person-hours and 0.6 person-rem per canister; (●) lowering the fuel transfer cask; pairing the fuel transfer cask with a shipping cask loading collar; connecting the fuel transfer cask to its control panel; lowering the canister into the shipping cask; disconnecting the control panel; and moving the fuel transfer cask to the storage stand in the truck bay or back to SFP-A: 8 person-hours and 0.10 person-rem per canister for seven canisters; (●) shield plug installation and moving the impact limiter into the shipping cask: 4.2 person-hours and 0.011 person-rem per canister for seven canisters; [i] and (●) preparing the shipping cask for transport offsite: 40 person-hours and 0.14 person-rem for seven canisters.

  – *Radiological Control Support*. The above estimates included person-hours for radiological control support. Based on job loading estimates, the person-rem estimate for radiological control support was 49 person-rem, and the total for all groups other than radiological control was estimated at 135 person-rem.

  – *Uncertainties*. Because of the uncertainty in the person-hour estimate and the expected radiological conditions, the evaluation estimated that the total exposure could vary by up to 30 percent. Considering these uncertainties, a range of 30 to 240 person-rem was selected to be used as the exposure estimate for the performance of the activities within the scope of this SER, including radiological control support.

- *Internal Exposure*. All individuals entering the FHB were monitored for internal radiation exposures according to established procedures. This monitoring was done by either periodic whole-body counting or bioassay, or both. All exposures to airborne radioactivity would be maintained ALARA and within the limits established in 10 CFR Part 20. Airborne radioactivity in work areas was monitored according to established procedures. Air sampling

[i] Editor's Note: The SER stated that the estimated exposure for these final two activities were in units of "person-rem per canister for seven canisters." Since both activities involved closure of the shipping cask, which contained seven canisters, the units are understood as person-rem per shipment preparation.

for particulates was performed using devices such as breathing zone air samplers and grab samples. Tritium grab samples were taken as required by established procedures.

- ***Evaluation: Radiation Protection/ALARA***. [116] The licensee's safety evaluation stated that minimizing occupational exposure was a major goal in the planning and preparation for all recovery activities. The planned actions minimized the time personnel worked in radiation fields, maximized the distance between personnel and radiation sources to the extent practicable, and used shielding where appropriate to meet the ALARA objective. ALARA measures included the following:

  - *Clothing*. Protective clothing was used as necessary to reduce the potential for external contamination and internal exposure of personnel. Exposures while executing individual tasks were maintained ALARA by a detailed radiological review.

  - *Defueling Canister Movement*. Canisters were staged underwater before their removal for shipment off site. When canisters were moved from the fuel canister storage racks to the dewatering station and fuel transfer cask loading station, the canisters were raised directly into the CTS and remained there until they reached their destination.

  - *Shielding*. Shielding was designed for components and structures that could present high dose rates during the activities described in this SER.

    - *Canister Loading Station*. The fuel transfer cask loading station was shielded with 3 inches of lead to ensure reasonably low dose rates on the dewatering station platform and around the edge of SFP-A. A shielded curtain extended down from the south side of the fuel transfer cask loading station into the water to ensure that no streaming would occur during canister transfer. The dewatering station platform provided shielding for any streaming that could occur from the east side of the fuel transfer cask loading station.

    - *Canister Dewatering Station*. The dewatering station platform incorporated 2 inches of lead shielding to shield operators from piping that ran under the dewatering station platform. Shielded handrails could be used to create a low-dose-rate area for operators during canister transfer operations.

    - *Canister Transfer Shield*. The CTS dose rates to trolley operators were limited to less than 5 millirem per hour from the canister being transferred. In addition, dose rates around the edge of the pool were estimated to be less than 16 millirem per hour for all transfer operations within SFP-A.

    - *Fuel Transfer Cask*. The fuel transfer cask dose rates were limited to less than 10 millirem per hour at a distance of 4 feet from the axis of the canister. At the fuel transfer cask loading station, the canister was raised into the fuel transfer cask where the canister remained until it was lowered into the shipping cask. The mini hot cell provided shielding of the canisters while the shield plug and impact limiters were installed on each canister cell in the shipping cask. Shield doors were provided at the bottom of the fuel transfer cask to minimize dose rates below the fuel transfer cask during transfer.

- *Dose Reduction Features*. Other features were incorporated to ensure that operator doses were minimized. Examples included the following: (●) The canister grapple was designed to permit "blind grapple" and to permit canister coupling while allowing for tolerances of the canister racks and the grapple mechanism. This permitted a smooth and efficient operation. (●) Dewatering system piping and valves were located below the shielded platform where practical, and reach rods were provided for valve operation. (●) Sufficient shielding was present on the dewatering system platform and handrails to ensure low dose rates on the platform during all transfer operations. (●) A decontamination ring provided for final washdown of the canister before loading it in the fuel transfer cask. This minimized the spread of contamination during canister transfer and loading of the shipping cask. (●) Canister dewatering couplings were designed to allow easy connection/disconnection. Different size couplings were used for inlet and outlet to ensure easy identification and to prevent misalignment.

• ***Evaluation: Occupational Exposure (Impact on Unit 1)***. [117] The licensee's safety evaluation stated that although the FHB crane and the truck bay were shared by the two plant units, their use by one unit or the other would be determined by operational considerations on a case-by-case basis. Because the Unit 1 and Unit 2 FHBs joined a common area in the truck bay, the activities described in the SER were evaluated for possible radiological impact on Unit 1. The following concerns were considered: (●) liquid release; (●) airborne release; and (●) direct radiation.

- *Liquid Release*. The activities described in the SER did not present a credible potential for radioactive liquid release to Unit 1. Any transfer of liquid, such as by the dewatering system, was controlled and maintained within the Unit 2 FHB.

- *Airborne Release*. Since the activities described in this SER were not expected to generate significant quantities of airborne radioactivity, no increase in airborne radioactivity in Unit 1 was expected.

- *Direct Radiation*. During loading of the defueling canisters into the shipping cask, the canister sources could increase the gamma dose rates in the Unit 1 FHB. To minimize the impact on Unit 1 operations and personnel exposures, operations in Unit 2 would be carried out so that the expected dose rate in Unit 1 from Unit 2 activities would not exceed 2.5 millirem per hour. The truck bay was normally considered part of Unit 1; however, no Unit 1 work would be performed in the truck bay area when Unit 2 fuel shipment activities were being performed without approved procedures. Consequently, for the purpose of this dose assessment, the Unit 1/Unit 2 boundary was considered to be at the interface between accessible areas of the Unit 1 FHB and the truck bay. Specific points of interest were the 347.5-foot elevation of the Unit 1 FHB, the open stairway on the Unit 1 side of the truck bay, and the environmental barrier. The environmental barrier was an unshielded structure in the north side (Unit 1 side) of the truck bay that separated the Unit 1 and Unit 2 atmospheres below the 347.5-foot elevation.

- *Source Term*. The canister source term used to calculate the dose rates at the Unit 1/Unit 2 boundaries was more conservative than that used to predict average dose rates to workers in Unit 2. Previous analyses were done to develop best estimates of anticipated dose rates. Unit 1 dose rates were analyzed to determine the maximum credible dose rate. Therefore, the source term employed to predict Unit 1 dose rates included conservative parameters not used to provide best estimate average dose rates. These parameters used a maximum amount of cobalt-60, which was based on material assay of cobalt-59 present in-core structural materials, and a hot channel factor of 1.9 to account for areas of the core where higher specific activity of radioactive materials could occur. Other conservative parameters used in previous analyses, such as maximum loaded canister and no shielding credit for the canister or its internal structures, were maintained in the maximum credible dose rate model.

- *Dose Rate (SFP-A Activities)*. Activities carried out within the confines of SFP-A would not affect dose rates in Unit 1 because of the distance to Unit 1 and the shielding provided by SFP-A concrete walls.

- *Dose Rate (Canister Transfers)*. Canisters were transferred from the fuel transfer cask loading platform in the spent fuel pool to the shipping cask using the fuel transfer cask. The was then moved along the west side of the FHB to the truck bay, where the transfer cask was lowered onto the shipping cask loading collar. The maximum dose rate at locations in Unit 1 from a single canister loaded in the fuel transfer cask was provided in the SER (refer to page 34 of Revision 4 of the SER) for various activities. Dose rates in Unit 1 ranged from 0.3 to 1.7 millirem per hour.

- *Dose Rate (Shipping Cask)*. The shipping cask could contain up to seven canisters and was designed to meet U.S. Department of Transportation regulations for shipment on public highways. These requirements included a limit of 10 millirem per hour at 6.6 feet from the cask. The cask was designed to ensure that this limit would not be exceeded. The safety analysis report [118] for the 125-B fuel canister shipping cask showed that the highest calculated dose rate from a fully loaded shipping cask was 6.3 millirem per hour at a distance of 6.6 feet. The total dose rate from a fully loaded shipping cask or from the fuel transfer cask with a single canister and a shipping cask with up to six canisters was expected to be 2.5 millirem per hour or less at all accessible areas in Unit 1.

- *Conclusion*. The licensee concluded that activities described in this SER could be performed without an unacceptable increase in personnel exposure in Unit 1 as access to the FHB was gained through the FHB truck bay door. The aircraft missile shield for this door was controlled by Unit 1. Access to the FHB was frequently required during normal plant operations; therefore, Unit 2 use of this door for fuel shipping activities would not affect normal operation of Unit 1. This SER demonstrated that activities described in this safety evaluation would not have an unacceptable impact on the safe operation of Unit 1.

- ***NRC Review: Occupational Exposure***. [119] The NRC's safety evaluation stated that all systems and components used in the licensee's proposed canister handling and preparation for shipment program were designed with appropriate engineered features to minimize the radiation exposure to plant personnel. The equipment would be operated by personnel trained in normal radiation protection practices and would be controlled by approved procedures that incorporated normal radiological controls. The licensee performed a radiological review of the proposed activities and projected a total dose commitment for the program of 184 person-rem. The NRC's review of the licensee's estimate concluded that the estimate was based on expected person-hours needed for the proposed tasks and the maximum radiation levels expected at various locations. The projected occupational exposure was within the scope of consideration in the PEIS.

The licensee's analysis also demonstrated that normal activities associated with the program would not result in radiation levels in excess of 2.5 millirem per hour in any exposed areas of Unit 1. Therefore, the program would have no adverse impact on the operation of Unit 1.

9.5.3.6 Canister Dewatering System (NA)

9.5.3.7 Use of Nonborated Water in Canister Loading Decontamination System (NA)

9.5.4 Testing of Core Region Defueling Techniques (NA)

9.5.5 Fines/Debris Vacuum System

- ***Purpose***. To modify the fines/debris vacuum system using a knockout canister and a filter canister in series. Modifications included: (●) use of a vacuum nozzle to allow larger debris particles to be vacuumed into the knockout canisters; (●) use of mechanical probes and water jets on the end of the vacuum nozzle to loosen the packed rubble; (●) use of a larger vacuum tool to allow debris removal from the lower head; and (●) temporary use of the vacuum system without a filter canister. The safety evaluation report [120, 121] for early defueling had previously approved the initial use of the fines/debris vacuum system.

- ***Evaluation: Occupational Exposure***. [122] The licensee's safety evaluation stated that bypassing the filter canister during operation of the fines/debris vacuum system was intended to be a short-term solution until issues relating to the rapid increase of pressure buildup across the filter canisters were satisfactorily resolved. The bypass mode of operation could increase the radioactive isotopic concentration in the reactor vessel water but should not appreciably increase radiation dose levels to the operators because of the shielding provided by the defueling work platform and the operation of the defueling water cleanup system. Radiation exposure rates inside the containment building were continuously monitored during defueling activities. Precautions, such as shielding or personnel relocation, would be used to minimize worker exposure. In addition, the release of additional particulates into the reactor vessel water was not expected to increase airborne particulates within the containment building because of the scrubbing action of the water and the operation of the off-gas system under the defueling work platform, as appropriate. Short-term testing of vacuum operations in the bypass mode confirmed that visibility, although impaired, could be recovered quickly, and no increase in radiation levels was expected on the defueling platform. Therefore, bypassing the filter canister during fines/debris vacuum system operation should not significantly impact the radiation exposure of personnel nor increase the quantity of airborne particulates in the containment building.

- ***NRC Review: Occupational Exposure***. [123] The NRC's safety evaluation noted that operations of the vacuum system without a filter on the knockout canister discharge were likely to cause high turbidity and an increased radionuclide concentration in the reactor coolant system water. Increased turbidity would require periodic shutdown of defueling operations until visibility improved. This action would present no adverse safety impact, provided that prudent judgment was exercised and that existing procedural controls were followed. Increased activity levels could cause slight increases in radiation levels on the defueling platform; however, procedures approved by the NRC were in place to control the radiological working conditions and would ensure that no adverse impact on the health and safety of the workers.

9.5.6 Hydraulic Shredder

- ***Purpose***. To use a hydraulically powered shredder to reduce the size of fuel pins and other core debris and to facilitate the loading of fuel canisters or debris buckets.

- ***Evaluation: Occupational Exposure***. [124] The licensee's safety evaluation stated that the shredder operated close to the rubble bed, and a significant portion of the generated particulate matter was assumed to settle to the rubble bed. Therefore, shredder operations should not substantially increase the dose contribution from particulates in the reactor vessel water. The operators' distance to the shredder and the amount of shielding afforded by the water and the defueling work platform would minimize the dose contribution from shredder operation. Airborne releases of radioactivity to the containment building were not expected to increase from shredder operation because of the scrubbing action of the reactor vessel water.

The operation of the shredder had the potential to increase the dissolved radionuclide concentration in the reactor vessel water and could increase the radiation levels experienced by the operators. If the shredder operation produced a higher concentration of radionuclides than had been experienced previously, water processing could be undertaken to lower the activity in the water. The radiological controls department monitored dose rates on the platform to ensure that dose rates to the operators were acceptable. The radiation control department would continue to monitor shredder operations and provide guidance for reactor coolant system processing and continued shredder operation to ensure that doses to operators were within an acceptable range.

- ***NRC Review***. [125] The NRC's safety evaluation report did not specifically address these topics.

9.5.7 Plasma Arc Torch (NA)

9.5.7.1 Use of Plasma Arc Torch to Cut Upper End Fittings (NA)

9.5.7.2 Use of Plasma Arc Torch to Cut Lower Core Support Assembly (NA)

9.5.7.3 Use of Plasma Arc Torch to Cut Baffle Plates and Core Support Shield (NA)

9.5.7.4 Use of Air as Secondary Gas for Plasma Arc Torch (NA)

9.5.8 Use of Core Bore Machine for Dismantling Lower Core Support Assembly (NA)

9.5.9 Sediment Transfer and Processing Operations

- ***Purpose***. To collect sediment from tanks and sumps in the auxiliary and fuel handling buildings, and also from the containment building basement and sump, in order to transfer the sediment to the spent resin storage tanks (SRSTs) and treat or process the sediment (for disposal).

- ***Evaluation: Occupational Exposure***. [126] The licensee's safety evaluation concluded that the assembly, installation, and operations of the sediment transfer and processing system resulted in personnel exposure of about 44 person-rem (as of the submittal of this safety evaluation report (SER)). The projected expenditure for continued operation of the sediment transfer and processing system was estimated to be about 58 person-rem.

- ***Evaluation: Radiation Protection/ALARA***. [127] The licensee's safety evaluation considered ALARA practices, work area accessibility, radiation monitoring, shielding, and access controls.

  - *ALARA Considerations*. All personnel performing work within the bounds of this evaluation would use every means available to maintain their exposures to radiation ALARA. Measures included: (●) Radiological control personnel would monitor work areas, as required, and provide dose rate information to aid individuals in performance of their tasks if radiological work practices were of concern. (●) Extensive planning of tasks to be conducted in radiation areas and training of personnel would be used to reduce the time necessary to complete tasks. (●) Mockups for training and proof-of-principle testing would be used. (●) Higher radiation areas would be identified to workers and shielded where practical. (●) Work would be structured to avoid those areas to the extent possible.

  - *Work Area Accessibility*. Cubicles (rooms) in the auxiliary and fuel handling building (AFHB) that could require worker accessibility for the removal of sediment from sumps and tanks were identified. Each cubicle would be decontaminated to remove contamination hot spots to minimize the need for respirators and to reduce general area dose rates in accordance with the licensee's criteria for the decontamination of the AFHB. Respirators could still be necessary for tank sediment removal operations because of the potential for airborne radioactivity caused by spraying and mixing operations.

The licensee would stipulate any respirator requirements. The potential sediment source tank temporary manway covers were designed to seal and secure each tank to minimize the possibility of airborne leakage. In addition, the manways were fitted with a vented plexiglass box enclosure to further minimize the potential for airborne release. Existing air monitoring systems continuously monitored auxiliary building airborne activity.

- *Radiation Monitoring*. Portable radiation monitors were used to detect radiation levels in general work areas. The monitors were located near potential sources of radiation and in areas where a buildup of radioactive material could result in excessively high radiation levels. As determined by the licensee, airborne monitoring would be required of the spent resin transfer pump cubicle and auxiliary building basement hallway during system activities.

- *Shielding and Access Control*. The primary objective of shielding design and access control was to protect plant personnel from radiation sources in the auxiliary building, including the sediment transfer header, transfer pumps, SRSTs, and the concentrated slurry transfer hose.

  - *Shielding Design*. Corridors were shielded to maintain dose rates at less than or equal to 2.5 millirem per hour during normal sediment transfer operations. Shielding design in the auxiliary building was based on the shielding of process equipment, resin tanks, and the transfer of spent resins and solidified waste. The transfer and processing system design incorporated provisions to prevent the formation of stagnant sediment slurry zones, which included interconnected flowpaths and use of flush water. Therefore, localized piping hot spots were minimized.

  - *Cubicle Walls*. The concrete walls of the SRST cubicles were originally designed to allow a maximum of 2.5 millirem per hour in adjacent areas while processing spent resin. Design radioactivity levels were based on 1-percent leakage of fission products from the fuel rods. Since the total activity in the sediment to be processed was less than in the design case, radiation levels in adjacent areas due to sediment contained in the tanks was not expected to be greater than 2.5 milliroentgens per hour. Calculations made using a computer code confirmed the dose rates to be much less than 0.01 milliroentgens per hour.

  - *Tanks*. The bottoms of the SRSTs could be shielded inside their respective cubicles to allow for maintenance of the mixers. Shielding could consist of lead blankets or lead sheets and cover the lower nozzle and discharge piping. The amount of shielding required could be minor since the tank would be empty and flushed. If necessary, the shielding would be used to reduce the dose rate before performance of any maintenance inside the cubicle.

– *Transfer Manifold*. The sediment transfer manifold and hoses were shielded to maintain general area radiation levels at or below 2.5 milliroentgens per hour during normal transfer operations. The hose from AFHB sediment sources to the SRST should result in about 54 milliroentgens per hour (unshielded) at a 1-foot distance with 5 percent by weight sediment in the line. The installed shielding (lead bricks) reduced that to less than 2.5 millirem per hour.

– *Hoses*. Hoses from the SRST to the solidification feed station could result in up to 2 roentgens per hour at 1 foot. The transfer pump skid was shielded to maintain workstation radiation levels at or below 2.5 millirem per hour. Process flow valves that required operation were located behind shielding and provided with reach rods or remote operators. The hoses from the chemical addition skid were not shielded since no radioactivity existed in these lines. The discharge transfer hose from the containment building basement sediment removal skid was routed on the 305-foot elevation (entry level) to a designated containment penetration for transfer to the SRST. The hose from the containment building basement to the SRST could result in about 200 millirem per hour (unshielded) at a 1-foot distance with 5-percent sediment in the line. The transfer hose was shielded to reduce hot spots and to minimize any contribution to general area dose. The licensee determined the shielding design and desired dose reduction.

- ***NRC Review***. [128] The NRC's SER did not specifically address these topics.

9.5.10 Pressurizer Spray Line Defueling System

- ***Purpose***. To flush fuel fines and core debris from the pressurizer spray line to the pressurizer vessel and the reactor coolant system cold-leg loop 2A. The source of flush water for the pressurizer spray line defueling system (PSLDS) was the defueling water cleanup system. Defueling consisted of flushing the pressurizer spray line in a series of steps to adequately remove fuel fines and debris in each different flowpath from the spray line tie-in.

- ***Evaluation: Occupational Exposure***. [129] The licensee's safety evaluation stated that defueling the pressurizer spray line could increase the specific activity in the reactor vessel water. This could occur during operation of the PSLDS through disturbance of the core debris by the introduction of flush water into the cold-leg loop 2A. Fuel material greater than 0.5 micron and other soluble fission products should be removed by the operation of the defueling water cleanup system filter train "A."

Operation of the defueling water cleanup system train "A" would reduce the occupational exposure during the defueling of the pressurizer spray line by maintaining low specific activities in the reactor vessel. The safety evaluation estimated the person-hours and person-rem associated with the installation, operation, and removal of the PSLDS (refer to page 10 of the safety evaluation report for details). The total person-rem attributable to the installation, operation, and removal of the PSLDS, as a whole, was expected to be between 13.6 and

25.2 person-rem. The estimate was based on a baseline of 16.14 person-rem, increased by 20 percent for radiation control coverage, and adjusted ± 30 percent due to uncertainties. Appropriate plant procedures would address personnel protection against airborne and other potential contamination generated by the installation, use, or removal of the PSLDS.

- ***NRC Review: Occupational Exposures***. [130] The NRC's safety evaluation concurred with the licensee' s assessment that proposed activities would not pose a risk to the health and safety of the public or the occupational workforce, nor did the activities exceed the scope of activities and associated environmental impacts considered in the PEIS.

9.5.11 Decontamination Using Ultrahigh Pressure Water Flush

- ***Purpose***. To use ultrahigh pressure (UHP) water flush at 20,000 to 55,000 pounds per square inch (psi) to remove surface coatings and surface contamination inside the containment building.

- ***Evaluation: Occupational Exposure (External Dose)***. [131] The licensee's safety evaluation stated that all individuals entering the containment building were monitored for external exposures in accordance with the radiological control procedures to ensure that personnel exposures were maintained within 10 CFR Part 20 dose equivalent limits. Administrative dose limits in accordance with these procedures were used to ensure that 10 CFR Part 20 dose limits were not exceeded. Extremity monitoring would be performed as needed in accordance with existing procedures.

Person-rem estimates for the decontamination activities within the scope of this safety evaluation were included in the safety evaluation report (SER) [132] for the containment building decontamination and dose reduction activities. The estimates presented in the report were based on job-hour estimates for these tasks. The job-hours to be spent in the D-rings and basement areas were separated from the general building activities because of the radiation levels. The dose rates for activities in the containment building were based on historical data collected in similar areas for similar tasks. For work in the D-rings, dose rate estimates were based on thermoluminescent dosimeter data, along with assumptions regarding decontamination and dose reduction effectiveness.

- ***Evaluation: Occupational Exposure (Internal Dose)***. [133] The licensee's safety evaluation stated that the use of UHP water flush would create the potential for local increases in airborne radioactivity. This potential increase was difficult to quantify because of a lack of operational experience at TMI-2 with water jets with pressures exceeding 10,000 psi and the limited information on the character of the surface contamination to be encountered in areas such as the "B" D-ring. Several precautions would be taken to ensure that worker internal exposures were maintained within regulatory limits. These precautions included the following:
(●) Radiological control personnel closely monitored the initial UHP water flush operators.
(●) Workers were required to wear respiratory protection with protection factors of 1000 for

particulates. (●) Breathing zone air samples for all workers involved with initial UHP water flush operations were required; initial breathing zone sample results were used to assess potential internal exposures in future operations. (●) In all cases, it would be an operational parameter to limit average worker exposures to airborne radioactivity to 1 maximum permissible concentration-hour per hour. (●) Engineering controls, such as local ventilation and respiratory protective devices, were used as required by radiological control personnel to limit exposures to this level.

- ***Evaluation: Radiation Protection/ALARA***. [134] The licensee's safety evaluation stated that minimizing occupational exposure was a major goal in the planning and preparation for all activities in the containment. Protective clothing and respirators would be used to reduce the potential for external contamination and internal exposure of personnel. The chosen techniques and sequence of operations were developed to achieve the greatest decontamination at minimum job-hour and person-rem expenditure in the containment. Other ALARA considerations included the following:

  - *Reviews*. Exposures while executing individual decontamination tasks were maintained ALARA through a detailed radiological review by the radiological controls department and very substantial mockup training for the work crew. This training approximated the actual work situation as closely as possible for each task, using appropriate equipment, protective clothing, and respiratory protection.

  - *Planning and Training*. Planning and training were proven methods of ensuring that personnel were properly prepared to conduct the assigned task expeditiously. These methods included the following: (●) Extensive planning of tasks to be conducted in a radiation field and training of personnel were used to reduce the time needed to complete a task. (●) Training aids were used extensively to familiarize personnel with the work area. (●) Higher radiation areas were identified to personnel, and the work was structured to avoid these areas to the extent practical. (●) Practice sessions were held as necessary to ensure that personnel understood their assignments before entering the containment.

  - *Evaluation and Feedback*. Potential improvements in operational technique were fed back into future work packages and mockup trainings in a manner consistent with the development of work activities. If the observed techniques definitively demonstrated major operational problems or the ineffectiveness of a particular decontamination technique, the decontamination activities would be altered to accommodate this feedback. The SER noted that the evaluation of the adequacy of a particular decontamination technique must consider and weigh several operational factors such as person-rem and job-hour expenditure, personnel safety, operational complexities, and training requirements.

  - *Decision Analysis*. Decisions concerning decontamination and dose reduction tasks and techniques were made with consideration of personnel exposure. Decision analysis was needed to evaluate different options to accomplish the desired task. Different levels of radiation protection for a given task would also be considered. The decision analysis was not intended to force the option that entailed the lowest worker exposure but instead to ensure that personnel exposures were considered along with other variables. Procedures were developed for this decisionmaking process to make the ALARA philosophy part of the work task from its inception and engineering to implementation.

- *Contamination Control*. The degree of contamination control employed in the containment building varied with locations and the number of job-hours to be spent in the area. The defueling area and associated transit areas on the 347-foot elevation (operating level) were the most controlled, whereas access to higher dose rate areas on the 305-foot elevation (entry level) were less stringently controlled. Additionally, the spread of contamination (cross contamination) would be minimized by a combination of planning and administrative and engineered controls, as described in the licensee's SER [(135)] for the containment building decontamination and dose reduction activities.

- ***NRC Review: Radiation Protection/ALARA***. [(136)] The licensee's safety evaluation stated that a small amount of airborne radioactivity, in the form of particulates and tritium, could be introduced into the containment building atmosphere during UHP water flush. During initial operations of the system, respiratory protection devices with appropriate protection factors would be worn. Normal radiological control practices would be sufficient to ensure that worker exposures remained ALARA. The NRC concluded that the proposed operation was within the scope of the decontamination activities addressed in the PEIS.

9.6 Evaluations for Defueling Operations

9.6.1 Preliminary Defueling

- ***Purpose***. To allow movement of debris within the reactor vessel, to allow installation of defueling equipment, and to identify core debris samples for later removal and analysis by INEL. The movement of debris within the vessel was prepared for early defueling and required some rearrangement of debris in the reactor vessel before the actual removal of fuel. These activities included the loading of small pieces of core debris into debris baskets but did not include actual loading of fuel debris into fuel canisters.

- ***Evaluation***. [(137)] Editor's Note: The licensee's safety evaluation report did not specifically address these topics.

- ***NRC Review: Radiation Protection/ALARA***. [(138)] The NRC's safety evaluation concluded that the licensee had implemented an acceptable program to maintain worker exposures ALARA during the proposed activities. During preliminary defueling activities, the licensee would take the same measures to maintain worker radiation exposures ALARA as described in the licensee's safety evaluation report [(139)] for early defueling.

- *Measures*. The NRC's safety evaluation noted that the following measures would be taken to maintain worker exposures ALARA: (●) For the activities performed in the containment building, the licensee's radiological controls department would determine monitoring requirements and evaluate the use of respirators for the proposed activities. (●) Worker training would be conducted on the defueling test assembly to simulate the activities to be performed; this would improve efficiency and minimize worker exposures. (●) During preliminary defueling, both a radiological control technician and an NRC-licensed fuel handling senior reactor operator would be stationed on the defueling platform to provide radiation monitoring and to assess existing radiation protection measures. (●) Typical dose rates on the defueling work platform were expected to range from 10 to 20 milliroentgens per hour. A dose rate of 30 milliroentgens per hour on the defueling work platform had been established as a point for evaluation of radiological conditions before the continuation of activities. Activities would be halted if platform dose rates reached 300 milliroentgens per hour, except as specified in the procedures. (●) Radiation shielding and other radiation protection measures in place for preliminary defueling were designed to protect workers during the full scope of early defueling activities.

  Since the fuel would not be loaded into canisters or removed from the reactor vessel during preliminary defueling, dose rates would not increase significantly. Therefore, the protective measures taken would provide a high degree of assurance that worker exposures would be maintained ALARA.

- *Conclusion*. The NRC reviewed the radiation monitoring program that would be in place and concluded that this program would provide adequate data and alarm functions. The licensee had established procedures to take appropriate actions should radiation levels be significantly higher than expected. The licensee also had a program to maintain occupational exposure ALARA. An integral part of this ALARA program were the management reviews to be performed during these preliminary defueling activities to ensure that ALARA objectives were met and feedback on ongoing activities to be reviewed for improvement.

9.6.2 Early Defueling

- ***Purpose***. To remove end fittings, structural materials, and related loose debris that would not involve removal of significant amounts of fuel material, to remove intact segments of fuel rods and other pieces of core debris, and to remove loose fuel fines (particles) by vacuuming operations.

- ***Evaluation: Occupational Exposure***. The licensee's safety evaluation included a review of radiological controls for external and internal exposures and exposure estimates.

- *External Exposure (Controls)*. Editor's Note: The licensee's safety evaluation for external exposure controls during early defueling was similar to the subsequent evaluation for bulk defueling. Refer to NUREG/KM Section 6.4 (below) on bulk defueling in this chapter for an updated evaluation.

- *Internal Exposure (Controls)*. Editor's Note: The licensee's safety evaluation for internal exposure controls during early defueling was similar to the subsequent evaluation for bulk defueling. Refer to NUREG/KM Section 6.4 (below) on bulk defueling in this chapter for an updated evaluation.

- *Exposure Estimates (Early Defueling)*. [140] Because of the nature and duration of the early defueling activities, the development of detailed person-rem estimates for early defueling was impractical at that time. As the early defueling plans became better established and the person-hour estimates were well defined, a refined person-rem estimate would be developed. Until then, the comparative collective personnel radiation exposures to workers during different defueling options, using nonspecific person-hour assumptions, were evaluated.

  - *Target Dose Rates*. Doses were evaluated on the basis of target dose rates for work locations in the containment building and estimated work hours. Based on recent dose rate reductions in the containment building, proposed dose reduction options, and recommended dose rate targets, the work locations were assigned the following mean dose rates (in units of millirem per person-hour): (●) 305-foot elevation, 110; (●) 347-foot elevation, 55; (●) canister handling bridge in the containment building, 30; (●) canister handling bridge in the fuel handling building, 2; and (●) defueling platform, 15. The dose rate targets were considered to be reasonably achievable. By the start of early defueling, the expected mean millirem per person-hour at specific areas of the 305-foot and 347-foot elevations were 85 and 45 millirem per hour, respectively. These dose rate decreases would be achieved by floor scabbling, decontamination, and source shielding. However, person-rem estimates were based on the following dose rate targets:

    - *Canister Handling Bridge*. The dose rate target for the canister handling bridge in the containment building was based on decontamination of the bridge, the effect of dose reduction on the 347-foot elevation, decontamination of the fuel transfer canal liner, shielding of the operator area on the bridge, and use of the canister transfer shield.

    - *Defueling Work Platform*. The dose rate target could be achieved by shielding the platform, as described in the safety evaluation report (SER), maintaining water activity at the defueling water cleanup system design specification, and decontaminating the fuel transfer canal liner.

    - *Fuel Handling Building*. The dose rate target was based on existing area dose rates, which were not expected to change significantly. An evaluation showed that canisters would be well shielded by the pool water and would have minimal impact on dose rates, other than at locations just above the surface of the spent fuel pool.

  - *Results*. The comparative evaluation indicated that the defueling option selected would take about 31,336 person-hours and 700 person-rem. These estimates were considered preliminary and were used as an indication of the magnitude of the expected doses (not an exact estimate). As more training and experience with the actual defueling activities were gained, more refined person-rem estimates would be established. The comparative evaluation indicated that the selected defueling option was the most person-rem efficient option of the manual techniques reviewed.

The SER presented the estimated person-hours and associated person-rem for the proposed defueling option (refer to Table 5.3-1 of the SER). These estimates included both early and bulk defueling activities and included activities such as installation, operation, maintenance, decontamination, and removal of the defueling equipment. The person-hours were divided by locations corresponding to the target dose rate locations, including the 305-foot and 347-foot elevations, canister handling bridge in the containment building, fuel handling building, and defueling work platform. Detailed exposure estimates would be developed on a task-by-task basis as part of the ALARA review of work in the containment building to ensure that each activity was performed to reduce collective worker exposure.

- ***Evaluation: Radiation Protection/ALARA***. (141) Editor's Note: The licensee's safety evaluation for radiation protection/ALARA during early defueling was similar to the subsequent evaluation for bulk defueling. Refer to NUREG/KM Section 6.4 on bulk defueling in this chapter for an updated evaluation.

- ***Evaluation: Shielding***. (142) Editor's Note: The licensee's safety evaluation for shielding during early defueling was similar to the subsequent evaluation for bulk defueling. Refer to NUREG/KM Section 6.4 on bulk defueling in this chapter for an updated evaluation.

- ***NRC Review: Occupational Exposure***. (143) In Supplement 1 to the PEIS, the NRC estimated that under the cleanup plan, reactor disassembly and defueling could result in 2600 to 15,000 person-rem. Although not separately listed, defueling activities alone would account for over half of the estimated occupational exposure. Since 1983, the licensee had made substantial progress through the dose reduction program in reducing the radiation levels in the containment building, especially in areas where defueling workers would spend most of their time. For example, the present dose rate at the defueling platform was less than 10 millirem per hour. The licensee had a continuing program to further reduce the ambient dose rate in the containment building, and the NRC expected that while defueling was in progress, the background radiation levels would continue to decrease. Considering this improvement, along with the ALARA program the licensee would implement during defueling, the NRC estimated that the occupational dose resulting from defueling operations was likely to be close to or fall below the lower range estimated in Supplement 1 of the PEIS.

- ***NRC Review: Radiation Protection/ALARA***. (144) The NRC's safety evaluation considered ALARA measures and radiation monitoring during defueling.

  - *Dose Reduction*. Along with the program to reduce ambient dose rate levels, the licensee implemented a program to maintain dose rates ALARA during defueling. This ALARA program would be achieved through design features, operator training, and operating procedures.

- *Shielding*. Except during the transfer of loaded canisters from the reactor vessel to the fuel transfer canal, the fuel and debris would be shielded by submergence underwater. Dose to defueling workers would mainly result from cesium-137 activity in the water. This activity, at about 0.05 microcurie per milliliter, would be kept low by processing through the defueling water cleanup system and submerged demineralizer system. The shield plates in the platform and the closure heads of loaded canisters would provide additional shielding for workers at the defueling work platform. Other design features provided shielding during the transfer of loaded canisters to the fuel transfer canal. Examples of these were the shield boot under the work platform, the canister transfer shield with an extendable shield collar, and the shielded canister handling bridges in the reactor and fuel buildings.

- *Training*. The licensee built a full-scale mockup outside of the containment building, called the "defueling test assembly," to train every defueling worker. For each defueling tool, a duplicate was available at the test assembly, and workers practiced the use of all defueling tools at the test assembly. Through this training, which simulated actual situations in the reactor vessel, the operators would be able to perform the defueling manipulations more efficiently. This would result in reduced radiation exposure.

- *Procedures*. Operating procedures for defueling considered promotion of the ALARA concept. For example, procedures precluded the raising of fuel debris outside of the 4 feet of water coverage zone unless radiological control personnel appropriately monitored the situation to determine that such action was in keeping with the ALARA objective. Procedures also required the flushing of debris from defueling tools as they were withdrawn from the vessel to prevent the spread of contamination. More importantly, the operating procedures were developed, in part, based on the experience with defueling tool operations at the defueling test assembly. This feedback promoted efficiency and shortened overall stay times in the radiation area.

- *Conclusion*. Based on the review of the above ALARA considerations, the NRC concluded that the licensee had an acceptable program to maintain the collective defueling occupational dose ALARA and that the occupational dose incurred during early defueling should be within or below the range discussed in Supplement 1 of the PEIS. Average dose rates at the defueling work platform were expected to be relatively low (about 15 to 25 millirem per hour).

- *Radiation Monitoring*. The NRC reviewed the radiation monitoring system that would be in place during defueling. The agency determined that this system would provide adequate data and appropriate alarms if radiation levels were significantly higher than those expected. The NRC calculated radiation levels at worker-occupied areas during unplanned events and accident circumstances (e.g., rising of the fuel debris above the normal 4 feet of water coverage, canister drop over the work platform). The NRC determined that the radiation monitoring system and the continuous monitoring by radiation control personnel should enable the workers to properly respond to those situations. In addition, the estimated radiation levels were such that workers would be able to exit the containment building without endangering their health and safety.

9.6.3 Storage of Upper End Fittings in an Array of 55-Gallon Drums

- ***Purpose***. To use shielded 55-gallon drums to store end fittings and other structural material removed from the reactor vessel.

- ***Evaluation: Occupational Exposure***. [145] The licensee's safety evaluation showed that each end fitting storage container would produce a maximum radiation level of 2 roentgens per hour at 1 foot from the shielded container. The storage area would be roped off to prevent inadvertent access, and shielding could be used to lower dose rates. Based on the maximum expected radiation levels, the licensee estimated that the exposure received in transferring the end fittings would be between 10 and 20 person-rem for the entire operation.

- ***NRC Review***. [146, 147] The NRC's safety evaluation report did not specifically address these topics.

9.6.4 Defueling (Also Known as "Bulk" Defueling)

- ***Purpose***. To develop defueling tools and to conduct activities necessary to remove the remaining fuel and structural debris located in the original core volume. An additional activity was to address vacuuming of debris from the lower head of the vessel, a primary activity associated with lower head defueling.

- ***Evaluation: Occupational Exposure***. [148, 149] The licensee's safety evaluation considered radiological controls for limiting external and internal doses, evaluated airborne radioactivity sources based on the plenum inspection activities, and estimated the job-hours and person-rem for the remainder of the defueling activities.

  - *Exposure Controls*. The safety evaluation report (SER) discussed the radiological controls for limiting external and internal doses.

    - *External Dose (Control)*. All individuals entering the containment building were monitored for external radiation exposures according to established radiological control procedures. All external radiation exposures were maintained within the dose equivalent limits established in 10 CFR Part 20. All personnel exposures were maintained ALARA. Administrative dose limits were applied according to established procedures to ensure that 10 CFR Part 20 limits were not exceeded. Extremity monitoring would be performed, as needed, according to existing radiological control procedures. Radiation exposure rates inside the containment building were monitored during defueling operations. The radiological controls department would determine the requirements for radiation monitoring for personnel protection during defueling.

- *Internal Exposure (Control)*. All exposures to airborne radioactivity were maintained ALARA within the limits established by 10 CFR Part 20. Internal exposures were controlled through monitoring and respiratory protection.

  - *Monitoring*. All individuals entering the containment building were monitored for internal radiation exposures according to established procedures. This monitoring was done by routine breathing zone air sampling and periodic whole-body counting. Airborne radioactivity in work areas was monitored according to established procedures. Air sampling for particulates was performed using devices such as lapel samples and grab samples. Tritium grab samples were taken as required by established procedures.

  - *Respiratory Protection*. Respiratory protection devices would be used to minimize the uptake and deposition of airborne radioactivity in the body. The use of respiratory protection devices could, by reducing uptakes of radioactive materials, result in overall dose savings (internal and external); however, if they impeded work, total dose could increase by causing an elevated external dose. The radiological controls department would determine, through the prework radiological review process, if the use of respiratory devices was needed to meet the ALARA standard for a particular task. This review would include an examination of the current radiological conditions in the work area, an assessment of the potential for the task or other concurrent tasks to perturb the radiological conditions, and a check of the results of previous airborne activity measurements in the work area for similar tasks.

- *Internal Exposure (Source Term)*. Airborne radioactivity source estimates from plenum inspection activities were used for planning purposes. An estimate of airborne radioactivity to be encountered by individuals performing defueling activities was derived from the continuous air monitors. These monitors were positioned on the internals indexing fixture (IIF) during plenum inspection activities to monitor radioactive particulates and tritium grab samples taken in the containment building before, during, and after reactor vessel head removal. The evaluation assumed that concentrations would remain in or about the ranges of airborne radioactivity concentrations encountered through March 31, 1986, for the remainder of the defueling operations.

  - *Beta-Gamma Activities*. Breathing zone samples from workers on the IIF platform during plenum inspection and fuel assembly end fitting separation activities showed a gross beta-gamma activity equivalent to an airborne concentration of 8.4×10^{-10} microcurie per milliliter for all beta- and gamma-emitting nuclides.

  - *Alpha Activity*. Alpha activity was below the minimum detectable for the breathing zone samples.

– *Tritium/Particulates*. The planned activities were not expected to increase the tritium or particulate levels inside the containment building.

  ▪ *Tritium Source*. The additional release of tritium to the containment building atmosphere because of evaporation of the reactor coolant was not expected to significantly increase the tritium level in the containment building atmosphere because of the low tritium concentration in the reactor coolant (about 0.03 microcurie per milliliter).

  ▪ *Particulate Sources*. Equipment and defueling canisters being removed from the reactor vessel would be wiped or sprayed down, as necessary, to remove particulates and enhance radiological control. Therefore, the quantity of particulates resulting from these activities that had the potential to become airborne was not expected to significantly increase the particulate concentration in the containment building atmosphere. While the canisters were located in the deep end of the fuel transfer canal (FTC), the opening of the canister relief valves, in spent fuel pool "A" or the direct venting of canisters or debris containers to the pools could cause particulates to be released to the surrounding water. However, these particulates would be entrained in the water, and their potential to become airborne would be minimal.

o *Exposure Estimates (Bulk Defueling)*. The SER provided the job-hours and corresponding person-rem expended during defueling through March 31, 1986. Totals were 15,055 job-hours and 213 person-rem. Preparations and installations contributed 3930 job-hours and 100 person-rem, defueling operations contributed 9300 job-hours and 86 person-rem, and maintenance activities contributed 1825 job-hours and 27 person-rem.

  The SER also estimated job-hours and person-rem for the remainder of the defueling activities (refer to Table 5.3-2 of the SER). Totals were 58,706 job-hours and 1199 person-rem. Preparations and installations contributed 26,740 job-hours and 768 person-rem, defueling operations contributed 25,000 job-hours and 244 person-rem, maintenance activities contributed 1900 job-hours and 22 person-rem, and decontamination and removal of the defueling equipment contributed 5066 job-hours and 165 person-rem. As more training occurred and experience with actual defueling activities was gained, these estimates would be reviewed and updated when necessary.

  The estimated person-rem exposures at the different locations were evaluated using the average dose rates during defueling activities through March 31, 1986. These dose rate estimates were: (●) 305-foot elevation, 75 millirem per hour; (●) 347-foot elevation, 50 millirem per hour; (●) canister handling bridge in the containment building, 20 millirem per hour; (●) canister handling bridge in the fuel handling building, 2 millirem per hour; and (●) defueling platform, 10 millirem per hour.

• ***Evaluation: Radiation Protection/ALARA***. [150, 151] The licensee's safety evaluation stated that principles of ALARA were considered during the design of the defueling tools and the planning of defueling activities. Also, ALARA was considered when studying alternatives for defueling operations and equipment. Specific design changes were made to the tools and equipment to enhance performance of certain operations. Operational sequences were reviewed and changed to allow performance of work in lower radiation areas, when possible. Minimizing occupational exposure was a major goal in the planning and preparation of all activities in the containment. Protective clothing and respirators would be used as required to reduce the potential for external contamination and internal exposure of personnel. Other measures included the following:

- *Planning*. Extensive planning of tasks to be conducted in a radiation field and training of personnel would reduce the time needed to complete a task. The higher radiation areas would be identified to personnel, and the work would be structured to avoid these areas to the extent practical. Practice sessions would be held, as necessary, to ensure that personnel understood their assignments before entering the containment building. Planning and training were proven methods of ensuring that personnel were properly prepared to conduct the assigned tasks expeditiously.

- *Training and Mockups*. Exposures while executing individual tasks were maintained ALARA through a detailed pretask radiological review by radiological engineering and mockup training. The need for mockup training would be determined on a case-by-case basis. A detailed mockup, called the "defueling test assembly," simulated the configuration and orientation of the rotatable work platform, vacuum system, T-slot, working slot, handrails, single canister support bracket, canister positioning system, and debris bed. The long-handled tools would be representative of the actual tool lengths to be used. Extensive training of workers on the defueling test assembly and other mockups would familiarize the workers with the tasks to be performed. This training would result in increased worker efficiency, less time in the containment building, and less personnel exposure.

- *Tool Design*. Tooling was designed with the intent of keeping radiation exposures ALARA. Because of the large component sizes, the equipment had to be assembled inside the containment building. The components were designed for rapid assembly. The defueling tools were assembled and tested outside of the containment building, then disassembled during the mockup training. Where practicable, shielding would be provided. For ease of decontamination, all tools were fabricated of stainless steel, aluminum, or both, with smooth inside and outside surfaces and no blind holes. Flushing and draining holes were provided, where required, with flushing capability from the top of the tooling. Spray rings were located under the defueling work platform to provide a washdown capability for canister removal areas and the long-handled tool slot.

- *Communications and Control*. Simplifying the communication and control required during defueling by having a centralized operating and control station would save time in the containment building. This centralized station allowed all operations to be conducted with constant monitoring and coordination. To minimize radiation exposure to personnel and provide control functions near the defueling operations, the control station was located on the auxiliary work platform in the south end of the canal.

- ***Evaluation: Shielding***. [152, 153] The licensee's safety evaluation included a series of calculations to evaluate the dose rates to personnel from various components that could constitute radiation sources during defueling. Also considered were various shielding configurations to minimize the radiological impact of these sources. The analyses were performed to ensure that defueling systems were adequately designed to minimize personnel occupational exposures based on theoretical or design-basis source terms. The shielding configurations and dose rates were provided as an estimate of conditions that could exist during defueling. Operations during defueling would be governed by actual measured radiation dose rates. Adequate precautions, such as shielding or personnel relocation, would be taken as necessary to ensure worker safety and to minimize collective personnel exposures.

  - *Shielding Source Terms*. The primary sources during defueling operations included loaded defueling canisters in the reactor vessel, the radioactivity in the reactor coolant, and the particulates in the vacuum system components. Source terms were developed for these items for use throughout the shielding review program.

    - *Defueling Canisters*. Source terms were developed for the three different types of defueling canisters: fuel, knockout, and filter canisters. The source term in general was calculated assuming the following: (●) Total core inventory of fission products, activation products, and actinides distributed throughout the 93.1 metric tons of uranium dioxide, which made up the total initial fuel load. (●) Core inventory at shutdown was predicted by the ORIGEN-2 computer code. (●) Predicted core inventory of cobalt-60 was adjusted to reflect cobalt-59 impurity levels in structural materials. (●) Predicted core inventory of cesium-137 was reduced by the number of curies removed by water processing, and the inventory of cesium-134 was adjusted to reflect actual cesium-134 to cesium-137 ratios. (●) No retention of noble gases was assumed. (●) Core inventory was decayed to October 1, 1985. (●) Core debris was assumed to be composed of uranium dioxide, Zircaloy, and stainless steel in the same proportions as originally present in the core region. (●) Each canister was assumed to contain its maximum permitted weight of fuel debris. (●) Fuel debris weight was limited by the maximum allowable weight of a loaded, fully dewatered canister. (●) Fuel debris was assumed to contain the average core fission product specific activity and also assumed to be distributed homogeneously throughout the usable canister volume. (●) Free volume in the canister was assumed to be water for the canisters in the reactor vessel and air (i.e., canister dewatered) for the canisters being transferred.

    - *Reactor Coolant*. During defueling, the reactor coolant in the IIF and reactor vessel contained dissolved radioactive materials. The water level was assumed to be at the 327-foot elevation. Various concentrations of radioactivity were evaluated to estimate dose rates for the start of defueling and after water processing had reduced cesium concentrations to an equilibrium level. Radioactive particulates suspended in the water were not considered.

    - *Fines/Debris Vacuum System*. The fines/debris vacuum system was used during defueling to remove fines and debris from the core region by vacuuming. The source terms for these components were derived from the system design bases. The water upstream of the knockout canister was assumed to contain 8000 parts per million (ppm) fuel debris, based on the expected loading rate of the knockout canister. Components downstream of the knockout canister were assumed to contain water with a suspension of 1400 ppm solid fuel debris.

- *Shielding Design*. A structured program of independent verification was instituted during the design stage. This provided additional confidence in the shielding analyses that were performed for the early defueling systems. Shielding requirements for major defueling components (e.g., defueling work platform, stationary platform, canister shield collars, and canister shield plugs) were calculated independently by the licensee's hardware vendor. Calculated dose rates were compared during shielding design, and any significant differences were investigated. A simplified standard calculation test case was performed to compare computer code results. Results of the independent verification program indicated good agreement between the two analytical programs. Many separate calculations were performed for various sources, shielding components, and operational scenarios. From these calculations came the final design recommendations for defueling system shielding components. Dose rates were calculated for normal defueling operations. Note that all dose rates were from the sources described and did not include background radiation.

- *Shielding Components*. For purposes of these analyses, normal defueling operations were assumed to include manual defueling with the entire tool slot in the defueling work platform unshielded. All areas on the work platforms were assumed to be occupied during normal operations. Access under the auxiliary work platforms on the north and south ends of the FTC was not considered part of normal operations. The individual shielding components described below were evaluated for their impact on dose rates in work areas.

  - *Defueling Work Platform*. The shielded defueling work platform was designed to limit dose rates to operators to less than 1 millirem per hour from sources in the reactor vessel during normal defueling operations. These sources included five fully loaded fuel canisters in the canister positioning system, two fully loaded filter canisters, vacuum system components, and reactor coolant. The shielding requirement for the platform was determined to be 6-inch-thick steel.

  - *Vertical Support Structure Shielding*. The north end of the support structure had vertical shielding attached that extended from the canal floor to the platform elevation. This vertical shield essentially spanned the width of the refueling canal and was made of 2-inch-thick steel.

  - *Service Platform*. The service platform was designed to limit dose rates to operators to less than 1 millirem per hour from sources in the vessel during normal defueling operations. The platform was constructed with 3-inch-thick steel plate.

  - *Auxiliary Work Platforms*. The auxiliary work platform was installed on the south end of the canal (located next to the defueling work platform at the opposite end of the deep end of the FTC). The platform was constructed of 1-inch-thick steel shielding or the equivalent. The dose rates to operators would be limited to about 2 millirem per hour from sources in the vessel during normal defueling operations. The auxiliary work platform on the north end of the canal (located between the defueling work platform and the deep end of the FTC) was not a shielded structure. However, the 2-inch-thick steel vertical support structure shield effectively reduced dose rates on the north end auxiliary work platform from sources in the reactor vessel to less than 1 millirem per hour during normal operations.

- *Unshielded Sources*. During normal operations, workers would be positioned along the open tool slot in the defueling work platform. The calculated dose rates to operators from the reactor coolant assumed predicted concentrations of key radionuclides that were based on anticipated water processing activities. A dose rate of 11 millirem per hour at 3 feet above the work platform at the open tool slot was calculated, assuming the following reactor coolant concentrations: 0.03 microcurie per milliliter of cesium-137, 0.1 microcurie per milliliter of antimony-125, and 0.007 microcurie per milliliter of cobalt-60. These were the expected concentrations after processing reactor coolant through the submerged demineralizer system on a 5-day cycle, for about a month. The maximum dose rate from the vacuum system components along the open slot was calculated to be about 1.3 millirem per hour. The vacuum system components considered were the pump and piping.

- *Canister Shielding*. To minimize the dose rates to operators from canisters in the canister positioning system within the reactor vessel, a 4-inch-thick steel shield plug could be placed on fuel canisters after they were loaded and placed in the topmost canister position. In addition, a 1-inch-thick steel collar could be incorporated into the canister sleeve design in the canister positioning system. The design for the filter canister support structure incorporated vertical shielding as well. This steel shield collar extended about 2 feet down from the top of the canister source region. These precautions would limit the maximum contribution from all canisters in the vessel to about 3 millirem per hour at the open tool slot.

- *Canal Deep End Sources*. Dose rates at the work platform due to sources in the deep end of the FTC were evaluated. These sources included the plenum at its storage location, loaded fuel and filter canisters in the storage racks, and a postulated deep-end water concentration equivalent in dose rate to 0.02 microcurie per milliliter cesium-137. The total dose rate from these sources to operators on the work platform would be less than 2 millirem per hour. This estimated dose rate did not take credit for the shielding effect of the 0.375-inch-thick steel dam.

o *Shielding Canister Transfer Operations*. Defueling systems were designed such that workers could remain on the defueling work platform during canister transfers from the canister positioning system. Therefore, special shielding components were designed to maintain reasonably low dose rates on the work platform during transfer operations.

– *Canister Transfer Shield and Shield Collar*. The canister transfer shield (CTS) was designed to be supported from the canister handling bridge. The CTS consisted of a fixed mast cylindrical shield of 2.5-inch-thick lead and a 9-foot-long sliding collar of 1.5-inch-thick lead. The collar would be used to ensure that the entire canister was shielded during all transfer operations. The shielding thickness was limited by the structural loading of the existing refueling bridge and by space limitations. Dose rate estimates were as follows:

  - *Loaded CTS Mask Over Vessel*. When the CTS was positioned over the defueling work platform, a total of 4-inch-thick lead shielding extended about 9 feet above the work platform. Dose rates due to direct radiation to operators on the defueling work platform as the canister was being transferred were calculated to be about 47 millirem per hour at a location 3 feet from the CTS. The dose rates dropped to 42 millirem per hour at a distance of 7 feet from the CTS and to 23 millirem per hour at a distance of 14 feet from the CTS.

  - *CTS Unshielded Bottom End*. The CTS did not have a shielded bottom; therefore, there would be an unshielded beam of radiation directed downward from the bottom of the CTS during canister transfers. The 4.5-inch gap between the bottom of the fixed mast and the defueling work platform provided a pathway for scattered radiation to reach personnel locations. This beam had the potential to increase dose rates on the platform because of backscattered radiation from surfaces in the path of this beam. The worst scattering surface was from the 6-inch-thick steel of the platform. The maximum dose rate estimated due to scatter at 3 feet above the platform was 13 millirem per hour at a distance of about 2 feet from the CTS. At 7 feet from the CTS, the scattered dose rate was 9 millirem per hour at 3 feet above the platform.

  - *Loaded CTS Mask over Deep End*. During the lowering of a canister into the deep end of the FTC, a part of the canister was shielded only by the 1.5-inch-thick lead CTS collar for a short time. Dose rates on the north end auxiliary work platform (located between the defueling work platform and the deep end of the FTC) were calculated to be as high as 560 millirem per hour, close to the CTS during this short time period. Routine access to the north end auxiliary work platform during the lowering of canisters into the deep end of the canal would not be permitted. The maximum dose rates on the service platform during any transfer operation were calculated as 120 millirem per hour at the north edge and 39 millirem per hour at the reactor vessel centerline. The maximum dose rate on the south end auxiliary work platform (located on the other side of the defueling work platform) during transfer to the deep end was 15 millirem per hour. The underside of the CTS was not shielded, so no access to the canal floor areas under the platforms would be permitted during canister transfers.

  - *CTS Shielded Top End*. The CTS also incorporated a 3-inch-thick lead shield plug above the top of the canister. Calculated dose rates to the operators on the canister handling bridge would be at most 5 millirem per hour from the canister being transferred. This dose rate did not take credit for any structural material in the bridge. The contribution to the dose rate to workers on the 347-foot elevation (operating level) around the canal would be at most 24 millirem per hour during any transfer operation. This dose rate did not take credit for the extra lead shielding provided by the collar.

– *Shield Boots*. A vertical shield extending down from underneath the defueling work platform into the reactor coolant was used during transfers to shield the canister between the water and the CTS. This boot structure was positioned at the primary canister transfer port (hole) on the defueling work platform for the canister positioning system (CPS) and also surrounded the two filter canister ports. These structures were constructed of 5-inch-thick steel and extended 2 feet into the reactor coolant. This shielding limited whole-body dose rates along the open slot to about 25 to 50 millirem per hour. Because of interferences with CPS or vacuum system piping, some sections of the boot were less than 5-inch-thick steel, and some sections did not extend the full 2 feet into the water. These cases were evaluated individually to ensure that dose rates would be maintained in the ranges calculated for the complete boot. These dose rates were calculated for canister transfer from the CPS. Transfers of filter canisters would result in similar dose rates.

– *Single Canister Support Bracket*. The single canister support bracket (SCSB) consisted of a single canister suspended from the defueling work platform in the center of the slot. The SCSB could be located anywhere along the slot within the diameter of the core former. During normal operations, there would be a dose rate increase along the open slot of less than 1 millirem per hour since there was greater than 6 feet of water shielding above the canister top. Shielding calculations assumed that, during transfers from the SCSB, the working slot would be closed with shielded panels during each transfer from the SCSB. These shielded panels were made of 6-inch-thick steel and designed so that there was no radiation streaming between the panels and the CTS. Assuming that the top of the canister was at the platform elevation, dose rates on the rotatable platform would be about 12 to 25 millirem per hour. Assuming that the canister transfer was from the alternate canister transfer position for the CPS, dose rates at the south end auxiliary work platform would be 100 millirem per hour or less. Dose rates on this auxiliary work platform would be lower if the SCSB were positioned in the north half of the reactor vessel during transfer. Since the SCSB was a limited use item to be utilized primarily until the CPS was installed, this operational constraint was considered to provide adequate protection of personnel.

o *Shielding Dose Rate Estimates*. The analytical approach taken in the shielding program was to provide a reasonably accurate assessment of the worst case radiological impact expected during defueling. The canister source terms were based on the maximum weight of fuel debris permitted to be loaded into a canister. For normal operations, the maximum number of canisters was assumed to be present. No shielding credit was taken for the massive structural components of the support structure or the work platforms.

– *Source Terms (Coolant Concentrations)*. The reactor coolant source terms reflected the calculated concentrations of soluble radioactive materials, based on expected water processing scenarios before and after the start of defueling. If necessary, defueling with operators working directly over the open slot would be temporarily interrupted when the reactor coolant concentrations (e.g., resulting from a crud burst) caused dose rates to increase above acceptable levels. A range of water concentrations at the start of defueling was evaluated to provide predictions of dose rate variability due to different water processing scenarios. These scenarios included: (1) no processing before the start of defueling, (2) one additional processing for cesium, antimony, and cobalt removal, and (3) the same as Scenario 2, with one additional processing for cesium removal.

  These scenarios were used to predict concentrations at the start of defueling (before routine processing of the reactor coolant was initiated). Scenario 1 was the worst case, and Scenario 2 was considered the most likely. In addition, an equilibrium water concentration was calculated that assumed Scenario 2 before the start of defueling. This equilibrium concentration was calculated after the start of defueling and assumed that the reactor coolant was processed through the submerged demineralizer system on a 5-day cycle, for 1 month.

– *Results*. The dose rate results at 3 feet above the defueling work platform along the open tool slot for these cases were as follows: (●) Scenario 1 was 46 millirem per hour; (●) Scenario 2 was 24 millirem per hour; (●) Scenario 3 was 15 millirem per hour; and (●) Scenario 2 with equilibrium water concentration was 11 millirem per hour.

  Opening the T-slot on the defueling work platform would increase the dose rates for operators working at that location. At 3 feet above the platform, and assuming the equilibrium water concentrations, the dose rate increase from opening the T-slot was 4 millirem per hour, for a total of 15 millirem per hour from the reactor coolant.

– *Dose Rate Goals*. Early in the defueling planning, the radiological controls department developed dose rate goals for the defueling effort. These dose rate goals were given as average millirem per hour for specific work locations and included background radiation and transit doses. The dose rate goals were 30 millirem per hour on the containment building canister handling bridge and 15 millirem per hour on the defueling work platform.

– *Dose Rate Design Parameters*. To ensure that defueling systems were adequately designed to achieve the dose rate goals, the dose rate was limited to the following dose rate design parameters: (●) 12 millirem per hour at the open tool slot; (●) 2 millirem per hour on the work platforms; (●) 10 millirem per hour on the canister handling bridge during canister transfer; and (●) 15 millirem per hour at 7 feet from the CTS during canister transfer.

- *Dose Rate Evaluations*. Dose rate estimates were compared with design parameters. The results below were based on estimates for early defueling (Revision 4 of the SER).
  - *Open Tool Slot*. The dose rate at the open slot due to sources in the vessel would be dominated by the contribution from radioactivity in the reactor coolant. Only a small fraction of the total dose rate was calculated to come from canister or vacuum system sources. The maximum dose rate at the open tool slot, assuming the equilibrium reactor coolant concentrations, was 13 to 15 millirem per hour at 3 feet above the platform. This only slightly exceeded the dose rate design parameter of 12 millirem per hour.
  - *Work Platforms During Normal Operations*. The dose rates on the work platforms were calculated to be about 2 millirem per hour or less during normal operations, which met the dose rate design parameter.
  - *Canister Handling Bridge During Canister Transfer*. The maximum dose rate to operators on the canister handling bridge trolley during a canister transfer was 5 millirem per hour from the canister in the CTS. This met the dose rate design parameter of 10 millirem per hour.
  - *Near CTS During Canister Transfer*. The maximum dose rate at 7 feet from the CTS during canister transfer was calculated as 51 millirem per hour, including direct and scattered radiation. This exceeded the design-basis dose rate of 15 millirem per hour. However, to meet the average personnel dose rate goals, operators could be temporarily relocated on the work platform area during certain transfer operations.
- *Dose Rate Reevaluations (Bulk Defueling)*. A significant effort was made throughout the design process to ensure that defueling systems would provide adequate radiation protection for operators and would result in the ALARA collective dose for defueling operations. Revision 10 of the SER noted that the overall exposure to workers was less than that predicted by the above calculations for early defueling in Revision 4 of the SER.

- ***NRC Review: Occupational Exposure***. [154] The NRC's safety evaluation stated that prior defueling experience had demonstrated the effectiveness of the licensee's ALARA program in maintaining occupational exposure. The licensee's SER [155, 156] for bulk defueling (Revision 10) reported that 213 person-rem of radiation exposure was incurred from defueling activities through March 31, 1985. The licensee currently estimated that an additional 1200 person-rem would be incurred through the completion of defueling operations. Although measured dose rates were generally lower than predicted, the licensee's revised exposure estimate of about 1400 person-rem for defueling activities was higher than earlier estimates due to the increased estimate of job-hours required for the completion of defueling. This increase was based on a more accurate assessment of the remaining activities and the time required for performing those activities. As discussed in Supplement 1 to the PEIS, the NRC estimated that reactor disassembly and defueling activities would result in a total exposure of 2600 to 15,000 person-rem, over half of which would come from defueling activities alone. Therefore, the licensee's revised estimate, based on the most recent defueling experience, was within the lower end of the range of the NRC's estimate.

- ***NRC Review: Radiation Protection/ALARA***. [157] In the NRC's previous SER [158] for early defueling, the agency approved the licensee's program for maintaining ALARA radiation exposures to workers. This comprehensive program would continue to be implemented throughout defueling activities. This program limited exposure through decontamination of work areas, design of defueling equipment, training of defueling workers, and development of appropriate operating procedures. Because of earlier decontamination activities in the containment building, measured dose rates in work areas were currently at or below the licensee's target levels for early defueling.

  - *Dose Reduction*. The licensee's dose reduction measures included the following: (●) Bulk defueling tools were designed to be compatible with existing tools (e.g., hydraulic systems, long-handled tool extensions) that permitted defueling activities at a safe distance from the core debris. (●) Radiation shielding for defueling workers was provided by the water in the reactor vessel and by the lead shield plates on the defueling work platform. (●) Design of the defueling canisters and handling equipment provided additional shielding when the canisters were transferred in air between the vessel and spent fuel pool "A." (●) The level of activity in the reactor coolant system was controlled to limit its contribution to area dose rates. (●) A full-scale mockup of the reactor vessel and defueling work platform, called a "defueling test assembly," was used to train defueling workers in the use of new tools and procedures, thereby reducing the time spent in radiation areas and limiting worker exposure. (●) As defueling continued, existing procedures would be revised or new procedures developed, as needed, to maintain exposures ALARA.

  - *Conclusion*. Based on the NRC's earlier conclusions in its safety evaluation of early defueling and on the effectiveness of the licensee's ALARA program as evidenced by defueling experience to date, the NRC concluded that the licensee had established an acceptable program for maintaining worker exposures ALARA during bulk defueling activities.

9.6.5 Use of Core Bore Machine for Bulk Defueling (NA)

9.6.6 Lower Core Support Assembly Defueling

- ***Purpose***. To dismantle and defuel the lower core support assembly (LCSA) and to partially defuel the lower reactor vessel head. Structural material removed from the LCSA included the (●) lower grid rib assembly; (●) lower grid forging; (●) distribution plate; and (●) in-core guide support plate. The lowest elliptical flow distributor and the gusseted in-core guide tubes would not be removed under this proposed activity. The use of the core bore machine, plasma arc cutting equipment, cavitating water jet, robot manipulators, automatic cutting equipment system, and other previously approved tools and equipment was included in this activity and associated safety evaluations.

- ***Evaluation: Occupations Exposure***. [159] Based on a comparison of activities associated with bulk reactor vessel defueling to those associated with LCSA defueling, the licensee's safety evaluation concluded that the radiological considerations associated with LCSA defueling were bounded by the license's safety evaluation report (SER) [160] for bulk defueling (refer to Section 5 of that report). However, to prevent exposure to operating personnel, special precautions would have to be taken during the transport of radioactive and contaminated pieces of the LCSA from the reactor vessel to the storage location. The plan presented in the SER for bulk defueling (refer to Section 2.1 of that report) was to flush all LCSA pieces before removal from the reactor vessel in order to remove the visible fuel debris and reduce radioactive contamination on each piece.

The evaluation assumed that pieces nearest the core would be the more radioactive because of cobolt-60 activation, making the lower grid rib assembly the most radioactive piece removed from the reactor vessel. The estimated radiation level of an 8-foot by 8-foot piece of the lower grid rib assembly would be 280 rem per hour at 1 foot. At a distance of 30 feet, the radiation level would be 2.1 rem per hour. This plate would have to be rigged, moved, and unrigged, if necessary, remotely. The other sections of the LCSA would pose less of a radiation hazard. Control of personnel exposure during the handling of these pieces would follow approved ALARA practices. An update of the job-hours and person-rem expended (as of October 1987) for all defueling activities was as follows: (●) preparations/installation was 3930 hours and 100 person-rem; (●) operations was 34,899 hours and 343 person-rem; and (●) maintenance/support was 20,570 hours and 311 person-rem. The overall estimated occupational exposure to complete the reactor vessel defueling would remain at about 1400 person-rem.

- ***NRC Review***. [161] The NRC's SER did not specifically address these topics.

9.6.7 Completion of Lower Core Support Assembly and Lower Head Defueling

- ***Purpose***. To remove the elliptical flow distributor and gusseted in-core guide tubes of the lower core support assembly (LCSA) and to defuel the reactor vessel lower head (RVLH).

- ***Evaluation: Occupational Exposure***. [162, 163] Based on a comparison of activities addressed by the licensee's safety evaluation report (SER) [164] for bulk defueling to those associated with LCSA/RVLH defueling, the licensee concluded that radiological considerations for LCSA/RVLH defueling were bounded by those for bulk defueling. However, special precautions would be taken to prevent exposure of operating personnel during transport of radioactive and contaminated pieces of the LCSA from the reactor vessel to their storage location within the containment building. Although these pieces of the LCSA would be inspected to ensure that there was no visible fuel debris, all pieces were radioactive due to cobolt-60 activation and surface contamination by soluble fission products.

- *Dose Rates*. The sections of the LCSA to be removed under the scope of this SER were less radioactive than the lower grid rib assembly. The measured radiation level of a 5-foot by 5-foot section of the lower grid rib assembly removed from the LCSA was 80 rem per hour within 1 foot of the surface. At a distance of 30 feet, the radiation level was less than 1 rem per hour following removal. This plate was rigged, moved, and unrigged remotely. Since the sections of the LCSA to be removed from the reactor vessel, within the scope of this SER, represented less of a radiation hazard, the adequacy of personnel exposure control practices was demonstrated by the radiation levels for removal of a section of the lower grid rib assembly.

- *Exposure Estimate (Revision 0)*. The SER for bulk defueling estimated an occupational exposure to complete bulk defueling of about 1400 person-rem. At the time this SER was submitted, this estimate had not been exceeded. However, the licensee expected that activities described in the SER for LCSA/RVLH defueling and the SER [165] for LCSA defueling (excluding the in-core guide tubes and elliptical flow distributor) would cause this estimate to be exceeded. Thus, the licensee would provide a subsequent update on the expected occupational exposure to complete reactor vessel defueling and the job-hours and person-rem expended to date for defueling activities.

- *Exposure Estimate (Revision 1 Update)*. An update of the actual job-hours and person-rem expended as of May 31, 1988, for all defueling activities was as follows: (●) preparation and installation was 5120 hours and 120 person-rem; (●) operations was 43,534 hours and 423 person-rem; (●) defueling support was 28,793 hours and 440 person-rem; and (●) maintenance was 970 hours and 45 person-rem. (No activity associated with final decontamination and removal of defueling equipment was performed. Decontamination maintenance in the containment building was not considered part of this activity.)

  The overall estimated occupational exposure to complete reactor vessel defueling was about 1580 person-rem.

- ***NRC Review***. [166] The NRC's SER did not specifically address these topics.

9.6.8 Upper Core Support Assembly Defueling

- ***Purpose***. To cut and move the baffle plates and to defuel the upper core support assembly (UCSA). This evaluation addressed the following activities: (●) cutting the baffle plates for later removal; (●) removing bolts from the baffle plates; (●) removing the baffle plates; and (●) removing core debris from the baffle plates and core former plates.

- ***Evaluation: Occupational Exposure***. [167] Based on a comparison of activities addressed by the licensee's safety evaluation report [168] for bulk defueling to those associated with UCSA defueling, the licensee concluded that radiological considerations associated with UCSA defueling were bounded by those for bulk defueling.

- *Precautions*. Special precautions would be taken to prevent operating personnel exposure during transport of radioactive and contaminated pieces of the UCSA from the reactor vessel to their storage location within the reactor vessel. Although these pieces of the UCSA would be inspected to ensure that there was no visible fuel debris, all pieces were radioactive due to cobolt-60 activation and surface contamination by soluble fission products. If the baffle plates were transferred to the modified core flood tanks by use of the transfer shield, the plates would be visually inspected to ensure that no fuel was present.

- *Dose Rates*. The measured radiation levels from the baffle plates reached 3000 roentgens per hour at a standoff distance of about 2 inches. If the baffle plates would be removed from the reactor vessel, the 3 inches of steel on the transfer shield would limit the dose rate at a standoff distance of 50 feet to about 1 roentgen per hour. Also, additional temporary shielding would be used as necessary to limit personnel exposures to considerably lower levels. The adequacy of personnel exposure control practices was demonstrated by handling core debris in the core region and during LCSA plate handling without shielding the plates.

- *Exposure Estimates*. The licensee's previous safety evaluation report [169] for the completion of lower core support assembly and lower head defueling estimated the occupational exposure to complete reactor vessel defueling at about 1580 person-rem. This estimate included the evolution of moving the baffle plates to modified core flood tanks (about 30 person-rem).

  The person-hours and person-rem expended previously through May 31, 1988, were as follows: (●) preparation and installation—5120 person-hours, 120 person-rem; (●) operations—43,534 person-hours, 423 person-rem; (●) defueling support—28,793 person-hours, 440 person-rem; and (●) maintenance—970 person-hours, 45 person-rem).

- ***NRC Review***. [170] The NRC's safety evaluation report did not specifically address these topics.

9.7 Evaluations for Waste Management

9.7.1 EPICOR II

- ***Purpose***. To decontaminate accident-generated, intermediate-level radioactive wastewater being held in tanks in the auxiliary building. Later, the system was used to polish effluents from the submerged demineralizer system (SDS) during the cleanup of highly radioactive water from the containment building sump, reactor coolant system, and reactor coolant drain tanks.

Following the decommissioning of the SDS, EPICOR II was used to clean residual wastewater from decontaminating the structures and systems.

- ***Evaluation***. Editor's Note: The licensee's safety evaluation was not located.

- ***NRC Review***. Editor's Note: The NRC documented its formal review in an environmental assessment [(171)] issued as NUREG-0591, "Environmental Assessment on the Use of EPICOR-II at Three Mile Island, Unit 2." An updated environmental assessment that applied to EPICOR II and all other cleanup activities was documented in the PEIS [(172)]issued March 1981. Refer to these reports for details of the following safety topics: (●) occupational exposure; (●) radiation protection/ALARA/shielding; and (●) radiological release.

Also, refer to the NRC's safety evaluation of the submerged demineralizer system (below) for thorough and updated radiological safety evaluations that are similar to those for EPICOR II operations.

9.7.2 Submerged Demineralizer System

9.7.2.1 Submerged Demineralizer System Operations

- ***Purpose***. To decontaminate the containment building sump water and reactor coolant system (RCS) water using the submerged demineralizer system (SDS), followed by effluent polishing with the EPICOR II system.

- ***Evaluation: Occupational Exposure***. [(173)] The licensee's safety evaluation was documented in a technical evaluation report (TER). [(j)] The SDS was designed to maintain radiation exposures to operating personnel ALARA. Several features were incorporated into the SDS design to implement the ALARA concept. Further, several activities were implemented before and shortly after the SDS startup to minimize occupational exposures.

  - *ALARA Design Features*. To implement the ALARA concept, the following features were incorporated into the SDS design: (●) Shielding was designed to limit whole-body dose rates in operating areas to less than 1 millirem per hour. The filters and ion exchange vessels were located about 16 feet underwater for shielding. Components and piping carrying high-activity water that were not underwater in the spent fuel pool were shielded to maintain external dose rates at acceptable levels. Controls and instrumentation were located in low-radiation areas. Components containing high-activity water were designed for venting exhaust gases to the SDS off-gas system. This system would minimize the potential for excessive airborne radioactivity releases in the work areas and to the environment. Additional design and operational ALARA features were also provided, as discussed in this TER.

---

[j] Editor's Note: The licensee's safety evaluation of a system design and its operations was typically documented in what the licensee called a "technical evaluation report."

The occupational exposure for the EPICOR II system was addressed in the NRC's environmental assessment [(174)] for the use of the EPICOR II system (NUREG-0591). The occupational radiation exposure for the EPICOR II system would be lower for processing the effluent from the SDS than the exposure for processing by EPICOR II directly.

- *Exposure Planning*. Several activities would be conducted before and shortly after the SDS startup to ensure that occupational exposures were minimized. These activities included the following: (●) review of operating, maintenance, and surveillance procedures to ensure that precautions and prerequisites were adequate; (●) review of the installed system to identify potential problems during operation and the implementation of corrective actions; (●) operational evaluations during preoperational testing and system training to update exposure estimates; and (●) determination of radiation dose rates during normal operations and maintenance evolutions.

  As these reviews were completed, operating and surveillance frequencies could be established, and the total occupational exposures could be updated for the various activities during SDS operation. This exercise would permit review of those activities estimated to yield the highest person-rem expenditure. Preexamination to ensure that every reasonable effort was made to minimize personnel exposure could include: (●) reduction of the frequency of operation; (●) temporary or additional shielding; (●) tool modifications; (●) procedure modifications; (●) personnel training to reduce worktime; and (●) component modifications.

- *Exposure Updates (1982 Addendum)*. The licensee submitted the first addendum [(175)] to its TER for the SDS. The addendum added Appendix 1 that discussed the reactor coolant processing plan. During SDS operation that processed gallons of containment building sump water and reactor coolant bleed tank water through December 31, 1981, exposure accumulation was 7.37 person-rem. This exposure considered operation, maintenance, outages, health physics, chemistry, and management. This was about one-third of the total radiation exposure estimated for all operations by the SDS as reported in the NRC's original safety evaluation report [(176)] for the SDS (21 person-rem). Since the RCS wastewater had significantly lower radioactivity levels, the TER update concluded that processing of the RCS water would fall within this estimate, when added to the end of the containment building sump water process.

• ***Evaluation: Radiation Protection/ALARA***. [(177)] The licensee safety evaluation stated that the objectives for SDS operations were to ensure that operations that supported the ongoing demineralization program were conducted in a radiologically safe manner, and further, that operations that might entail radiation exposure would strive to maintain radiation exposure at ALARA levels.

During the operational period of the system, the effective control of radiation exposure would be based on the following considerations: (●) sound engineering design of the facilities and equipment; (●) use of proper radiation protection practices, including work task planning for the proper use of the appropriate equipment by qualified personnel; and (●) strict adherence to the radiological control procedures developed for TMI-2.

- *Design Considerations*. The SDS was specifically designed to maintain exposure to operating personnel ALARA. To implement this concept, the components carrying water with high levels of activity were provided with additional shielding or submerged in the spent fuel pool. Shielding was designed to limit whole-body exposure rates in operating areas to about 1 milliroentgen per hour. In addition, components carrying high-level process fluids would exhaust to the SDS off-gas system. This method of off-gas treatment would minimize the potential for airborne releases in the work areas.

- *Operational Considerations*. The system design incorporated the following operational ALARA considerations: (●) Personnel servicing a specific component on the SDS would have reduced exposure because of the shielding between the individual components that constitute substantial radiation sources to the receptor. (●) Exposure of personnel who operate valves on the SDS would be reduced by using reach rods through lead and steel shield boxes. (●) Controls for the SDS would be located in low-radiation zones. (●) Airborne radioactive material concentration would be minimized by routing the off-gas effluent from the SDS to the building ventilation system for further treatment. (●) Sampling stations for the feed stream and filters that contained high levels of radioactive materials would be exhausted through the SDS ventilation system. (●) All samplings were performed in shielded glove boxes to minimize the possibility of inadvertent leakage and spread of contamination during routine operations.

- *Facility Design Features*. In terms of protection of the public, the SDS was designed to take maximum advantage of station features already in place and operational. In addition, design features provided by the system were intended to reduce releases of radioactive material to the environment. The following features provided individual protection from radiological hazards during normal operations, such as external exposure, and unanticipated occurrences, such as spills: (●) The SDS primary demineralization unit was housed under about 16 feet of shielding water in the spent fuel pool. (●) The entire process and all equipment were housed in the fuel handling building, which was a seismic Category I structure with air handling and ventilation systems and designed to mitigate the consequences of a radiological accident. (●) The SDS was designed to allow zero discharge of liquid effluents. (●) Off-gas system effluents would be filtered and monitored before input to existing ventilation exhaust systems. (●) SDS components and associated couplings were operated in underwater containment devices; each containment was connected to a pump manifold. A continuous flow of about 10 gallons per minute was maintained through each containment, and the combined flow was processed through a separate ion exchange column and then discharged back to the spent fuel pool. (●) Loaded vessels would be placed in a shielded cask underwater. (●) To the extent possible, all welded, stainless-steel construction was specified to minimize the potential for leakage. (●) Lead or equivalent shielding was provided for pipes, valves, and vessels (except those located underwater), where necessary for personnel protection. (●) Design of a sequenced multibed process (three beds in series) precluded breakthrough and contamination of the outlet stream.

(●) The entire process stream was designed with appropriate pressure indicators. (●) Inlet, outlet, and vent connections were made with remote-operated, quick-release valve couplings.

- *Shielding*. The minimum shielding thickness required for radiological protection was designed to reduce levels in occupied areas to less than 1 milliroentgen per hour. Operating panels and instrumentation racks were located away from potential sources of radiation, or adequate shielding was provided to meet radiological exposure design limits. All movements of the vessels out of the fuel pool would be performed using a shielded transfer cask.

- *Ventilation*. The SDS ventilation and off-gas system was designed to minimize airborne radiological releases to the environment. These design features included: (●) a manual level-controlled off-gas separator tank with mist eliminator to receive vent connections from the ion exchange and filter vessels, sample glove boxes, piping manifolds, and the dewatering station; (●) a roughing filter with differential pressure indication; (●) two high-efficiency particulate air (HEPA) filters with differential pressure indication; (●) a centrifugal off-gas blower with flow indication; (●) sample ports for monitoring the system and dispersed oil particulate smoke test ports for HEPA testing; and (●) effluent of the SDS off-gas system routed to the existing building ventilation system exhaust, which was filtered again through the fuel handling building exhaust HEPA filters before discharge from the plant.

- *Area Radiation Monitoring Instrumentation*. General area radiation monitors were provided to alert personnel of increasing radiation levels during normal operations or maintenance activities.

- ***NRC Review: Occupational Exposure (Normal Operations)***. [178, 179] The NRC's safety evaluation was published in the PEIS and NUREG-0796, "Operation of the Submerged Demineralizer System at Three Mile Island Nuclear Station, Unit No. 2." The NRC considered the licensee's estimates for occupational doses that were provided in a subsequent letter [180] to the NRC. The NRC developed its own estimated cumulative occupational doses associated with process operations, vessel handling, packaging, storage, and the decommissioning of the SDS.

- *Original Evaluation (PEIS)*. The NRC estimated in the PEIS [181] the occupational dose associated with the processing of the containment building sump and RCS water in an SDS-type system. (During the writing and public reviews of the PEIS, the SDS was being designed.) Further, the NRC estimated the occupational dose associated with vessel handling, packaging, and storage. As discussed in Sections 7 and 8 of the PEIS, the total for these estimates ranged from about 11 to 15 person-rem. In the preparation of the PEIS, the NRC did not quantify the cumulative occupational dose associated with decommissioning of the SDS; however, based on the design and construction of the SDS (i.e., the use of small-bore stainless-steel piping, flushing provisions, ease of system access, and other features), the NRC expected the corresponding occupational dose to be insignificant (i.e., a small fraction of the dose anticipated for the entire cleanup of TMI-2).

- *Updated Evaluation (NUREG-0796)*. The evaluation done by the NRC for the PEIS was updated in the agency's safety evaluation report (SER) (182) for the as-designed SDS. Based on the now completed construction, layout, and design of the SDS and on more detailed information regarding staffing requirements, process operations, sampling, vessel handling, packaging and disposition, the NRC reevaluated the occupational dose associated with process operations, vessel handling, and packaging. The NRC estimated that these SDS operations would result in cumulative occupational doses totaling 21 person-rem. Table 27 of the NRC's SER presented a more detailed breakdown of this estimate. In addition, the NRC estimated that the cumulative occupational dose associated with the decommissioning of the SDS would total about 35 person-rem. This quantitative estimate of the occupational dose that would result from decommissioning yielded a value that was a small fraction of the cumulative dose estimated in the PEIS for the entire cleanup (2000 to 8000 person-rem). [k] In addition, the NRC estimate of occupational dose that resulted from decommissioning of the SDS was a small fraction of the dose estimated in the PEIS for decommissioning the entire plant (1800 person-rem). Table 27 of the NRC's SER also presented details of the NRC's dose estimate for decommissioning.

- *Conclusion*. Based on this most recent estimate of the cumulative occupational radiation doses associated with SDS activities, the NRC concluded that the likely occupational doses resulting from use of the SDS fell within the scope of those previously considered in the PEIS.

• ***NRC Review: Radiation Protection/ALARA***. (183) The NRC's safety evaluation noted that as part of the licensing process for TMI-2, the original radiation protection program was described in the preaccident TMI-2 final safety analysis report (refer to Section 12.0). After the accident in March 1979, several changes were made to the radiation protection program. These changes, which were incorporated to reflect the unique postaccident radiological environment at TMI-2, were described in the licensee's radiological controls program management plan (184) and in the quarterly status reports on the implementation of this program. The licensee's radiation protection plan (185) supplied additional information.

The licensee's TER and system description (186) for the SDS included information on SDS layout and equipment design, operating procedures and techniques, and practices proposed to protect personnel against radiation. Personnel would be protected by shielding to reduce levels of radiation, ventilation arranged to control the flow of potentially contaminated air, and radiation monitoring systems to measure levels of radiation in potentially occupied areas and to measure

[k] Editor's Note: The original PEIS issued early in 1981 defined the cleanup period as the time of the decision point on whether the plant would be refurbished or decommissioned. The first Quick Look video inspection inside the reactor core cavity in the summer of 1982 revealed a degree of damage beyond what the licensee planners had expected. The first official report of core melt by INEL was in 1986, over 7 years following the accident. However, the decision point was reached shortly after the Quick Look inspection.

airborne radioactivity throughout the plant. A health physics program would be provided for plant personnel during SDS operation, maintenance, radwaste handling, and inservice inspection.

The NRC reviewed and evaluated the licensee's description and analysis of the radiation protection program. The agency concluded that the radiation protection program would provide reasonable assurance based on the following: (●) Doses to personnel would be less than those established by 10 CFR Part 20. (●) Design features and program features were consistent with the 10 CFR Part 20 ALARA criteria. (●) The program was acceptable for the planned operation of the SDS. Details of the NRC's review included shielding, radiation monitoring (area and airborne), ventilation, radiation protection features and equipment, personnel dosimetry, procedures, and personnel qualifications.

- *Shielding*. The design objectives for SDS shielding were to ensure that radiation exposure rates to operating personnel would be within the required limits of 10 CFR Part 20 and that these exposure rates would also be maintained at an average level of less than 1 milliroentgen per hour.

  The licensee included many features in the design and layout of SDS equipment and facility shielding to maintain radiation exposures ALARA. These features included: (●) the location of major sources of radiation underwater; (●) the use of labyrinths to eliminate direct shine to accessible areas; (●) the use of shielded valve stations; and (●) provisions for adequate flushing of system piping. These features contributed significantly to minimizing radiation exposures to the personnel who would operate the system.

  Principal radiation sources in the SDS and associated systems included the feed tanks, piping from the containment building to the feed tanks, and the ion exchange vessels. The feed tanks were shielded with 3 feet of concrete. The piping from the containment building was shielded with a minimum of 2.5 inches of lead. Valve containment boxes were shielded with a minimum of 20.5 inches of concrete. The ion exchange vessels would be shielded by greater than 12 feet of water during processing and greater than 8 feet of water during vessel changeout.

  The NRC found that the licensee had designed the facility to keep radiation exposures within the applicable limits of 10 CFR Part 20. The design and arrangement features were included to reduce unnecessary exposure during operations. Based on its review, the NRC concluded that the shielding and arrangement of the facility were acceptable.

- *Area Radiation Monitoring*. The licensee described the area radiation monitoring system in the system description report [187] for the SDS and in a letter [188] to the NRC on the SDS radiation protection program. The monitoring system comprised area radiation monitors, continuous air monitors, and portable tritium samplers.

  – *Area Monitors*. Area radiation monitors would be placed at spent fuel pool "A" (SFP-A) and at the north end of the spent fuel pool "B" (SFP-B) between the cask washing pit and the surge pit. If radiation levels exceeded the alarm setpoint on either of these monitors, the radiation monitor alarm would be actuated. A trip signal from the alarm at SFP-B would also close the feed isolation valve.

– *Continuous Air Monitors*. Three continuous air monitors were installed in the vicinity of the SDS: (●) north of SFP-B; (●) at SFP-A; and (●) at the valve containment box at the southwest end of SFP-A. All continuous air monitors alarmed locally.

– *Tritium Samplers*. Portable tritium samplers were installed in the vicinity of SFP-B. Grab samples would be taken and analyzed, and the results made available to the licensee's radiation protection personnel on a timely basis to permit appropriate operational analysis.

The NRC concluded that there was reasonable assurance that radiation levels within the plant would be adequately monitored and that the area radiation monitoring system was acceptable. This conclusion was based on the locations chosen, detector sensitivities, alarm settings that were set to alarm at radiation levels that were low enough to ensure worker dose minimization, and the calibration program.

- *Ventilation*. During operation of the SDS, the plant's ventilation system would be operated to maintain a suitable environment for personnel and equipment in the fuel handling building. The SDS also had a separate off-gas system that interfaced with the plant's ventilation system.

  – *Objectives*. The objectives of the off-gas and plant ventilation systems were to protect operating personnel from possible airborne radioactivity and to provide assurance that the maximum expected airborne radioactivity concentrations would be maintained within the limits of 10 CFR Part 20 and ALARA.

  – *Tritium Concentrations*. The NRC previously determined in its evaluation of tritium releases from SFP-B during operation of the SDS that the fuel handling building ventilation system was capable of maintaining concentrations of airborne tritium in the building below the limits in 10 CFR Part 20, even with concentrations of tritiated water in the fuel pool as high as 1.0 microcurie per milliliter. With tritiated water at a concentration 0.15 microcurie per milliliter in the spent fuel pool, grab samples indicated airborne tritium concentrations at about $1x10^{-8}$ microcurie per milliliter. This value was well below the 10 CFR Part 20 limits of $5x10^{-6}$ microcurie per milliliter.

  – *Design Features*. To meet the objectives, the adopted design features included: (●) ventilation airflow from areas of least radioactive contamination to areas of progressively greater radioactive contamination; (●) ventilation exhaust to ventilation filters; (●) maintenance of slightly negative pressures in selected areas; and (●) selection of airflow rates to maintain airborne concentrations of radioactive material below the limits of 10 CFR Part 20. Additionally, the ventilation system contained a quick access filter housing design feature to permit changing of the filter quickly and with minimal worker exposure. This design also minimized the probability of the spread of radioactive contamination.

– *Conclusion*. The NRC concluded that the off-gas ventilation system, as presented in the system description for SDS, and the fuel handling building ventilation system, as described in Sections 9.4 and 12.2 of the NRC-approved preaccident TMI-2 final safety analysis report, was acceptable because the system met the radiation protection design objectives, and the ventilation systems would maintain doses from airborne radioactive materials below the limits of 10 CFR Part 20 and ALARA.

- *Radiation Protection Features*. The radiation protection features at TMI-2 included: (●) a health physics counting room; (●) several radiochemistry counting areas; (●) personnel decontamination and emergency treatment areas; (●) access control points; (●) calibration facility; (●) respirator testing facility; (●) respirator maintenance facility; (●) contaminated clothing laundry; (●) whole-body counting facility; and (●) radiological control offices. Operation of the SDS would use these systems and facilities as appropriate.

  Additional radiation protection features to support the SDS included: (●) three shielded glove boxes with filtered ventilation; (●) use of shielded and ventilated valve manifold boxes; and (●) use of shielded, all-welded piping runs. The glove boxes and valve manifold boxes were drained and included flush connections to aid in minimizing radiation fields by enabling decontamination without opening the manifold. An off-gas system maintained a negative pressure in the glove boxes, valve manifolds, feed standpipe, off-gas separator tank, beta monitoring manifold, and feed tanks. The off-gas system would continuously sweep those components. If the off-gas system was not operating, neither filling of the tanks nor processing of water by the SDS would be permitted.

  The NRC concluded that these radiation protection features were appropriate and found that they were acceptable for limiting occupational exposure during SDS operation.

- *Radiation Protection Equipment*. Radiation protection equipment included: (●) portable survey instruments; (●) personnel monitoring equipment; (●) fixed and portable area and airborne radioactivity monitors; (●) laboratory equipment; (●) air samplers; (●) respiratory protective equipment; (●) protective clothing; and (●) contamination control equipment. SDS workers would use this equipment during SDS operation. Personnel working in the SDS areas were required to use a combination of portable friskers, hand and foot monitors, and portable monitors before leaving the licensee's protected area. Since the SDS was installed in the licensee's restricted area, the above contamination control equipment was part of the existing plant equipment. Continuous air monitors, tritium samplers, and area radiation monitors were located on the 347-foot elevation of the fuel handling building, near the SDS.

  The NRC concluded that the number and types of equipment used were adequate and provided reasonable assurance that the licensee would be able to maintain occupational radiation exposures ALARA.

- *Personnel Dosimetry*. Radiation dosimetry used during SDS operations included thermoluminescent dosimeter (TLD) badges, self-reading dosimeters, and bioassays.

  – *TLD Badges*. All permanent and temporary plant personnel would be assigned beta-gamma TLD badges when working in the SDS areas and any other restricted (radiologically controlled) areas. These badges would be processed monthly or more frequently if a significant exposure was expected or if required by the licensee's administrative control procedures.

  – *Self-Reading Dosimeters*. All personnel were also required to wear self-reading dosimeters when working in radiation work permit areas. The readings from these dosimeters were used to keep a cumulative and easily obtainable total of an individual's dose before TLD badge processing.

  – *Bioassays*. As a minimum, whole-body counting was performed annually on personnel who entered radiation work permit required areas or who wore respirators. Whole-body counting and other bioassays were also performed when required by the radiological control procedure manual or when deemed necessary by radiological control personnel. All radiation exposure information would be processed and recorded in accordance with 10 CFR Part 20.

  – *Conclusion*. The NRC concluded that the licensee's personnel dosimetry program was acceptable for SDS operation.

- *Plant Procedures and Practices*. Maintenance, repair, and surveillance activities in addition to the methods that were used by the licensee were reviewed by the licensee's ALARA group to ensure that all plant radiation protection procedures, practices, and criteria were considered and also to ensure that occupational radiation exposures would be ALARA and in accordance with Regulatory Guide 8.8, [189] "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Plants Will Be as Low as Is Reasonably Achievable."

  – *Procedures*. Procedures were in effect for the following functions: (●) ensure that exposure limits were not exceeded by plant or visitor personnel on site; (●) administer and control conditions of radiation work permits; (●) establish radiological survey frequencies; (●) post radiation areas to control access to various categories of radiologically controlled areas; (●) control all radioactive material entering or leaving the plant site; (●) train plant and contractor personnel in radiation protection policies and procedures; and (●) meet the recommendations of Regulatory Guide 1.33, [190] "Quality Assurance Program Requirements (Operation)."

  – *Operations and Personnel Qualifications*. The radiological control organization, the qualifications of the health physics personnel, the objectives of the radiological control program, and the ways the objectives would be implemented were in accordance with Regulatory Guides 8.10, [191] "Operating Philosophy for Maintaining Occupational

Radiation Exposures as Low as Is Reasonably Achievable," and 1.8, [192] "Qualification and Training of Personnel for Nuclear Power Plants," where determined by the NRC to be acceptable.

  – *Training*. Radiation protection personnel who were assigned to support SDS operations would receive training on the SDS. Topics included system description, system design bases, and SDS radiological control responsibilities. The training included a classroom phase, system walkthrough, and a written examination. At a minimum, a radiological control technician would be assigned to the SDS when filling the feed tanks or processing water.

- *Conclusion*. Based on the information presented in the SDS description, the licensee's radiation protection plan, the licensee's radiological control program management plan, and the licensee's responses to the NRC's questions, the NRC's safety evaluation of the operation of the SDS concluded that the licensee's radiation protection program was capable of maintaining occupational radiation exposures within the applicable limits of 10 CFR Part 20; 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities"; and ALARA.

9.7.2.2 Submerged Demineralizer System Liner Recombiner and Vacuum Outgassing System

- ***Purpose***. To eliminate the potential of a combustible hydrogen and oxygen mixture existing in the submerged demineralizer system (SDS) liners and to facilitate the ultimate shipment and burial of the SDS liners. The liner recombiner and vacuum outgassing system (LRVOS) was designed to remove moisture by evaporation from the zeolite beds of SDS spent liners. This operation dried the beds but did not remove the water in the zeolite.

- ***Evaluation: Radiation Protection/ALARA***. [193, 194] The licensee's safety evaluation concluded that activities associated with the operation of the LRVOS did not involve an unacceptable radiological safety concern. The basis for this was covered in the SDS technical evaluation report (TER).

- ***NRC Review: Radiation Protection/ALARA***. [195] The NRC's safety evaluation concluded that the risk to the health and safety of the public and the occupational workforce was minimal. Further, the use of the LRVOS did not change the boundary of accidents previously considered in the NRC's safety evaluation report [196] for the SDS. During normal system operation, the LRVOS would interconnect and communicate with the spent SDS liner ventline and the SDS off-gas filtration system. All potential radionuclide transport pathways in the LRVOS would communicate through a knockout drum [l] and two 0.2-micron filters. In reviewing the designs of the LRVOS, SDS high-efficiency particulate air (HEPA) filtration system, and SDS monitoring system, the NRC concluded that adequate radiological controls (e.g., shielding, particulate

[l] The vacuum system contained a suction water separator, called the knockout drum, to keep any slugs of water from getting to the pump.

filters, and radiation monitors) would be implemented. The lead shielding on the knockout drum and upstream filter would provide added margin to maintain ambient dose fields at less than 5 millirem per hour. The catalyst loading tool design, which provided remote handling capabilities with the SDS liner under 24 feet of water, also ensured that operator doses would be low.

System dose rates and liquid samples from the water separator would provide early indications of significant entrapment and transport of radionuclides (cesium-137 and strontium-90). In the unlikely event of this occurrence, the vacuum drying operation would be terminated. Small quantities of tritiated water would be transferred through the LRVOS into the SDS off-gas filtration system. However, this tritium would be negligible compared to the airborne tritium from normal water evaporation in the spent fuel pool "B" (50 curies per year) and well within the bounds considered in the PEIS. Based on these findings, the NRC concluded that the operation of the LRVOS should not measurably increase the occupation doses for plant operators and that there were adequate radiation controls.

9.8 Endnotes

Reference citations are exact filenames of documents on the Digital Versatile Discs (DVDs) to NUREG/KM-0001, Supplement 1, (197) "Three Mile Island Accident of 1979 Knowledge Management Digest: Recovery and Cleanup," issued June 2016, DVD document filenames start with a full date (YYY-MM-DD) or end with a partial date (YYYY-DD). Citations for references that are not on a DVD begin with the author organization or name(s), with the document title in quotation marks. Contents of the DVDs are available at the Idaho National Laboratory Web site (https://tmi2kml.inl.gov).

[1] USNRC, "Three Mile Island Accident of 1979 Knowledge Management Digest: Recovery and Cleanup, Overview," NUREG/KM-0001, Revision 1, June 2016 [Available at nrc.gov]
[2] NUREG-0640, TMI-2, Radiation Protection Program Report of the Special Panel (1979-12)
[3] 1981 NRC Annual Report (NUREG-0920) [Available on NUREG/KM-0001 DVD-1]
[4] NUREG-0683, SUPP. 1, PEIS, Occupational Radiation Dose (1984-10)
[5] *U.S. Code of Federal Regulations*, "Standards for Protection Against Radiation," Part 20, Chapter I, Title 10, "Energy"
[6] USNRC, "Three Mile Island Accident of 1979 Knowledge Management Digest: Recovery and Cleanup," NUREG/KM-0001, Supplement 1, June 2016 [Available at nrc.gov]
[7] (1980-01-24) GPU, TMI-2 Radiation Protection Plan, Rev. 1
[8] (1979-11-29) NRC Review, Radiation Protection Plan (re 11-21-1979)
[9] (1980-07-25) GPU, TMI-II Radiation Protection Plan, Rev. 2 (Transmittal Letter Only)
[10] (1981-03-05) NRC Review, TMI-2 Radiation Protection Plan, Rev. 2 (re 07-25-1980)
[11] (1982-12-06) GPU Nuclear Corporation Radiation Protection Plan, Rev. 0 (Replaces TMI-II Rad Plan)
[12] (1983-01-11) NRC Review, GPUNC Radiation Protection Plan, Rev. 0
[13] (1985-12-27) GPU, Corporate Radiation Protection Plan, Rev. 1
[14] (1986-04-09) NRC Review, Corporate Radiation Protection Plan (re 12-27-1985)
[15] (1988-05-25) Operating License Amendment 30

[16] (1993-01-04) GPU, Proposed TMI-2 Possession Only License, Radiation Protection Plan, Rev. 4
[17] (1986-06-23) GPU, Handling of Radioactive Core Debris
[18] (1990-03-01) Final Report, Three Mile Island Unit 2 Safety Advisory Board
[19] (1982-06-15) GPU Safety Evaluation, Axial Power Shaping Rod Insertion Test
[20] (1982-05-17) Order Amendment
[21] (1982-07-06) GPU Safety Evaluation, Insertion Camera Through Reactor Vessel Leadscrew Opening, Rev. 2
[22] (1982-07-13) NRC Review, Control Rod Drive Mechanism Quick Look Camera Inspection (re 07-06-1982) (2)
[23] (1983-05-19) GPU Safety Analysis, Radiation Characterization Under Reactor Vessel Head (No Polar Crane), Rev. 0
[24] (1983-05-19) GPU Safety Analysis, Radiation Characterization Under Reactor Vessel Head (No Polar Crane), Rev. 0
[25] (1983-05-19) GPU Safety Analysis, Radiation Characterization Under Reactor Vessel Head (No Polar Crane), Rev. 0
[26] (1983-07-13) NRC Review, Reactor Vessel Underhead Characterization (re Various Letters)
[27] (1983-07-20) GPU Safety Evaluation, Underhead Characterization, Core Sampling Addendum
[28] USNRC, “Three Mile Island Accident of 1979 Knowledge Management Digest: Recovery and Cleanup,” NUREG/KM-0001, Supplement 1, June 2016 [Available at nrc.gov]
[29] (1983-07-20) GPU Safety Evaluation, Underhead Characterization, Core Sampling Addendum
[30] (1983-05-19) GPU Safety Analysis, Radiation Characterization Under Reactor Vessel Head (No Polar Crane), Rev. 0
[31] (1983-07-20) GPU Safety Evaluation, Underhead Characterization, Core Sampling Addendum
[32] (1983-05-25) GPU Safety Analysis, Radiation Characterization Under the Reactor Vessel Head, Rev. 0
[33] (1983-08-19) NRC Safety Evaluation, Addendum to the Underhead Characterization Study (re 07-30-1983)
[34] (1986-06-11) GPU Safety Evaluation, Core Stratification Sample Acquisition, Rev. 4
[35] (1986-06-11) GPU Safety Evaluation, Core Stratification Sample Acquisition, Rev. 4
[36] International Commission on Radiological Protection, Publication 26, “Recommendations of the ICRP,” 1977, Ottawa, Ontario, Canada
[37] (1986-06-11) GPU Safety Evaluation, Core Stratification Sample Acquisition, Rev. 4
[38] (1986-06-11) GPU Safety Evaluation, Core Stratification Sample Acquisition, Rev. 4
[39] (1986-05-05) NRC Safety Evaluation, Core Stratification Sample Acquisition SER, Rev. 1 (re 08-30-1985, 12-31-1985)
[40] (1989-08-18) GPU Safety Evaluation, Remove Metallurgical Samples from Reactor Vessel SER
[41] (1989-10-20) GPU Safety Evaluation, Remove Metallurgical Samples from Vessel, Rev. 1 (re 08-18-1989) (effective pages)
[42] (1986-05-15) GPU Safety Evaluation, Defueling SER, Rev. 10
[43] (1989-11-28) NRC Safety Evaluation, Reactor Vessel Lower Head Metallurgical Sampling (re 08-18-1989)
[44] (1986-03-28) GPU Safety Evaluation, Reactor Building Decontamination and Dose Reduction Activities for 1986, Rev. 0

[45] (1986-03-28) GPU Safety Evaluation, Reactor Building Decontamination and Dose Reduction Activities for 1986, Rev. 0
[46] USNRC, "Acceptable Programs for Respiratory Protection," Regulatory Guide 8.15, October 1976 [Current and past regulatory guides are available at nrc.gov]
[47] (1986-03-28) GPU Safety Evaluation, Reactor Building Decontamination and Dose Reduction Activities for 1986, Rev. 0
[48] (1986-03-28) GPU Safety Evaluation, Reactor Building Decontamination and Dose Reduction Activities for 1986, Rev. 0
[49] (1984-02-03) NRC Safety Evaluation, Containment Decontamination and Dose Reduction (re 09-29, 01-23-1983)
[50] (1984-02-03) NRC Safety Evaluation, Containment Decontamination and Dose Reduction (re 09-29, 01-23-1983)
[51] (1983-02-18) GPU Safety Evaluation, Polar Crane Load Test
[52] (1983-02-18) GPU Safety Evaluation, Polar Crane Load Test
[53] (1983-11-18) NRC Safety Evaluation, Reactor Building Polar Crane Load Test
[54] (1983-09-29) GPU, Plans for First Pass Stud Detensioning for Head Removal
[55] (1984-01-30) GPU Response to NRC, First Pass Stud Detensioning SER (re 09-29, 12-06-1983)
[56] (1984-02-17) NRC Safety Evaluation, First Pass Vessel Stud Detensioning (re 09-29, 12-06-1983, 01-30-1984)
[57] (1984-03-09) GPU Safety Evaluation, Head Removal, Rev. 5
[58] (1984-03-09) GPU Safety Evaluation, Head Removal, Rev. 5
[59] (1984-03-09) GPU Safety Evaluation, Head Removal, Rev. 5
[60] (1984-07-17) NRC Safety Evaluation, Reactor Pressure Vessel Head Lift (re -3-09-1984)
[61] (1984-07-17) NRC Safety Evaluation, Reactor Pressure Vessel Head Lift (re -3-09-1984)
[62] (1985-04-19) GPU Safety Evaluation, Heavy Load Handling Over Reactor Vessel SER
[63] (1985-05-02) NRC Safety Evaluation, Heavy Load Handling Over the TMI-2 Reactor Vessel (re 04-19-1985)
[64] (1984-06-18) GPU Safety Evaluation, Preparatory Activities for Plenum Assembly Removal
[65] (1984-06-18) GPU Safety Evaluation, Preparatory Activities for Plenum Assembly Removal
[66] (1984-06-18) GPU Safety Evaluation, Preparatory Activities for Plenum Assembly Removal
[67] (1984-07-31) NRC Review, Preparatory Activities for Plenum Assembly Removal SER (re 06-18-1984)
[68] (1984-09-14) NRC Safety Evaluation, Plenum Assembly Removal (re various letters)
[69] (1982-07-13) NRC Review, Control Rod Drive Mechanism Quick Look Camera Inspection (re 07-06-1982)
[70] (1983-07-13) NRC Review, Reactor Vessel Underhead Characterization (re Various Letters)
[71] (1984-09-14) NRC Safety Evaluation, Plenum Assembly Removal (re various letters)
[72] (1984-09-14) NRC Safety Evaluation, Plenum Assembly Removal (re various letters)
[73] (1985-05-01) GPU Safety Evaluation, Plenum Removal and Transfer, Rev. 3
[74] (1985-05-01) GPU Safety Evaluation, Plenum Removal and Transfer, Rev. 3
[75] (1985-05-01) GPU Safety Evaluation, Plenum Removal and Transfer, Rev. 3
[76] (1985-05-07) NRC Safety Evaluation, Lifting, Transfer, and Storage of Plenum Assembly (re various letters)
[77] (1984-09-14) NRC Safety Evaluation, Plenum Assembly Removal (re various letters)
[78] (1983-02-25) GPU Safety Evaluation, Makeup and Purification Demineralizer Resin Sampling
[79] (1984-07-19) GPU Safety Evaluation, Makeup and Purification Cesium Elution SER
[80] (1984-07-19) GPU Safety Evaluation, Makeup and Purification Cesium Elution SER

[81] (1984-09-19) NRC Safety Evaluation, Makeup and Purification Demineralizer Cesium Elution (re 07-19, 08-29-1984)
[82] (1984-09-19) NRC Safety Evaluation, Makeup and Purification Demineralizer Cesium Elution (re 07-19, 08-29-1984)
[83] (1984-05-31) GPU Safety Evaluation, Internals Indexing Fixture Processing System, Rev. 1
[84] (1984-05-31) GPU Safety Evaluation, Internals Indexing Fixture Processing System, Rev. 1
[85] (1982-11-03) GPU Technical Evaluation, RCS Processing with RCS in a Partial Drained Condition
[86] (1984-05-31) GPU Safety Evaluation, Internals Indexing Fixture Processing System, Rev. 1
[87] (1984-07-24) NRC Safety Evaluation, Internals Index Fixture Water Processing System (re 05-31, 07-16-1984)
[88] (1989-05-12) GPU Technical Evaluation, Defueling Water Cleanup System, Rev. 12
[89] (1985-01-14) GPU Technical Evaluation, Defueling Water Cleanup System (with System Descriptions)
[90] (1987-09-28) GPU Technical Evaluation, SDS, Rev. 5
[91] (1989-05-12) GPU Technical Evaluation, Defueling Water Cleanup System, Rev. 12
[92] (1985-08-06) NRC Safety Evaluation, Defueling Water Cleanup System (re various letters)
[93] (1986-02-06) GPU Safety Evaluation, TMI-2 Temporary Reactor Vessel Water Filtration System
[94] (1986-04-14) GPU Safety Evaluation, Temporary Reactor Vessel Filtration System, Rev. 2
[95] (1986-02-06) GPU Safety Evaluation, TMI-2 Temporary Reactor Vessel Water Filtration System
[96] (1986-04-14) GPU Safety Evaluation, Temporary Reactor Vessel Filtration System, Rev. 2
[97] (1986-11-17) GPU Safety Evaluation, TMI-2 Temporary Reactor Vessel Filtration System, Rev. 3
[98] (1985-10-10) GPU Safety Evaluation, Early Defueling of the TMI-2 Reactor Vessel, Rev. 4
[99] (1986-02-07) NRC Safety Evaluation, Temporary Reactor Vessel Water Filtration System
[100] (1986-05-05) GPU Safety Evaluation, Addition of a Biocide to the TMI-2 Reactor Coolant System, Rev. 0
[101] (1986-05-06) NRC Review, Addition of a Biocide to the TMI-2 Reactor Coolant System (re 05-05-1986)
[102] (1987-09-22) GPU Technical Evaluation, Defueling Canister TER, Rev. 4
[103] (1985-05-20) GPU Safety Evaluation, Early Defueling of the TMI-2 Reactor Vessel, Rev. 1
[104] (1985-10-10) GPU Safety Evaluation, Early Defueling of the TMI-2 Reactor Vessel, Rev. 4
[105] (1985-11-05) NRC Safety Evaluation, Defueling Canisters (re various letters)
[106] (1986-08-12) NRC Review, Defueling Canister TER, Rev. 2 (re various letters)
[107] (1987-12-04) NRC Review, Defueling Canister TER (re various letters)
[108] (1986-04-11) GPU Safety Evaluation, Replacement of Loaded Fuel Canister Head Gaskets
[109] (1986-04-11) GPU Safety Evaluation, Replacement of Loaded Fuel Canister Head Gaskets
[110] (1986-04-29) NRC Review, Replacement of Loaded Fuel Canister Head Gaskets (re 04-11-1986)
[111] (1986-05-29) GPU Technical Evaluation, Fuel Canister Storage Racks, Rev. 1
[112] (1985-05-03) NRC Review, Fuel Canister Storage Racks TER (re 02-27-1985)
[113] (1987-04-02) GPU, Safety Evaluation, Canister Handling and Preparation for Shipment SER, Rev. 4 (Effective Pages Only)
[114] (1986-05-15) GPU Safety Evaluation, Defueling SER, Rev. 10
[115] (1986-05-20) GPU Safety Evaluation, Defueling SER, Rev. 10 (Correction)
[116] (1987-04-02) GPU, Safety Evaluation, Canister Handling and Preparation for Shipment SER, Rev. 4 (Effective Pages Only)

[117] (1987-04-02) GPU, Safety Evaluation, Canister Handling and Preparation for Shipment SER, Rev. 4 (Effective Pages Only)
[118] Pacific Nuclear Systems, Inc., "Safety Analysts Report for the NuPac 125-B Fuel Shipping Cask," Rev. 0, June 1985 [ADAMS Accession No. ML20132E700]
[119] (1986-06-20) NRC Safety Evaluation, Canister Handling and Preparation for Shipment Program (re 02-17, 06-11-1986)
[120] (1985-10-10) GPU Safety Evaluation, Early Defueling of the TMI-2 Reactor Vessel, Rev. 4
[121] (1985-11-12) NRC Safety Evaluation, Early Defueling of the TMI-2 Reactor Vessel (10-10-1985)
[122] (1986-01-28) GPU, Fines-Debris Vacuum System Modifications (re various letters)
[123] (1986-02-21) NRC Review, Modification of Fines and Debris Vacuum System (re various letters)
[124] (1986-03-26) GPU, Plan to Use Hydraulic Shredder to Reduce Core Debris
[125] (1986-04-16) NRC Review, Use of Hydraulic Shredder
[126] (1989-01-10) GPU Safety Evaluation, Sediment Transfer and Processing Operations, Rev. 4
[127] (1989-01-10) GPU Safety Evaluation, Sediment Transfer and Processing Operations, Rev. 4
[128] (1986-09-25) NRC Review, Sediment Transfer and Processing Operations (re 03-18-1985)
[129] (1987-01-15) GPU Safety Analysis, Pressurizer Spray Line Defueling System, Rev. 2
[130] (1986-12-22) NRC Review, Proposal for Defueling the Pressurizer Spray Line (re 08-06-1985, 11-26-1986)
[131] (1986-03-14) GPU Safety Evaluation, Decontamination Using Ultrahigh Pressure Water Flush, Rev. 0
[132] (1986-03-28) GPU Safety Evaluation, Reactor Building Decontamination and Dose Reduction Activities for 1986, Rev. 0
[133] (1986-03-14) GPU Safety Evaluation, Decontamination Using Ultrahigh Pressure Water Flush, Rev. 0
[134] (1986-03-14) GPU Safety Evaluation, Decontamination Using Ultrahigh Pressure Water Flush, Rev. 0
[135] (1986-03-28) GPU Safety Evaluation, Reactor Building Decontamination and Dose Reduction Activities for 1986, Rev. 0
[136] (1986-07-16) NRC Safety Evaluation, Decontamination using Ultrahigh Pressure Water Flush (re 03-14-1986)
[137] (1985-10-24) GPU, Commencement of Preliminary Defueling Operations
[138] (1985-10-29) NRC Safety Evaluation, Commencement of TMI-2 Preliminary Defueling Operations (re 10-24-1985)
[139] (1985-10-10) GPU Safety Evaluation, Early Defueling of the TMI-2 Reactor Vessel, Rev. 4
[140] (1985-10-10) GPU Safety Evaluation, Early Defueling of the TMI-2 Reactor Vessel, Rev. 4
[141] (1985-10-10) GPU Safety Evaluation, Early Defueling of the TMI-2 Reactor Vessel, Rev. 4
[142] (1985-10-10) GPU Safety Evaluation, Early Defueling of the TMI-2 Reactor Vessel, Rev. 4
[143] (1985-11-12) NRC Safety Evaluation, Early Defueling of the TMI-2 Reactor Vessel (10-10-1985)
[144] (1985-11-12) NRC Safety Evaluation, Early Defueling of the TMI-2 Reactor Vessel (10-10-1985)
[145] (1986-08-16) GPU, Storage of Upper End Fittings in Shielded 55 Gal Drums
[146] (1986-09-10) NRC Review, Plans to Store Upper End Fittings (re various letters)
[147] (1986-09-25) NRC Safety Evaluation, Storage of Upper End Fittings in an Array of 55 Gallon Drums (re various letters)
[148] (1986-05-15) GPU Safety Evaluation, Defueling SER, Rev. 10
[149] (1986-05-20) GPU Safety Evaluation, Defueling SER, Rev. 10 (Correction)

[150] (1986-05-15) GPU Safety Evaluation, Defueling SER, Rev. 10
[151] (1986-05-20) GPU Safety Evaluation, Defueling SER, Rev. 10 (Correction)
[152] (1986-05-15) GPU Safety Evaluation, Defueling SER, Rev. 10
[153] (1986-05-20) GPU Safety Evaluation, Defueling SER, Rev. 10 (Correction)
[154] (1986-07-24) NRC Review, Core Region Bulk Defueling Operations Limited to the Core Region (re 05-15, 07-23-1986)
[155] (1986-05-15) GPU Safety Evaluation, Defueling SER, Rev. 10
[156] (1986-05-20) GPU Safety Evaluation, Defueling SER, Rev. 10 (Correction)
[157] (1986-07-24) NRC Review, Core Region Bulk Defueling Operations Limited to the Core Region (re 05-15, 07-23-1986)
[158] (1985-11-12) NRC Safety Evaluation, Early Defueling of the TMI-2 Reactor Vessel (10-10-1985)
[159] (1988-01-18) GPU Safety Evaluation, Lower Core Support Assembly Defueling, Rev. 2
[160] (1986-05-15) GPU Safety Evaluation, Defueling SER, Rev. 10
[161] (1988-04-01) NRC Safety Evaluation, Lower Core Support Assembly Defueling (re 01-18-1988)
[162] (1988-06-06) GPU Safety Evaluation, Completion of Lower Core Support and Lower Head Defueling, Rev. 0
[163] (1988-06-27) GPU Safety Evaluation, Completion of Lower Core Support and Lower Head Defueling, Rev. 1
[164] (1986-05-15) GPU Safety Evaluation, Defueling SER, Rev. 10
[165] (1988-01-18) GPU Safety Evaluation, Lower Core Support Assembly Defueling, Rev. 2
[166] (1988-12-01) NRC Safety Evaluation, Lower Core Support and Lower Head Defueling (re 06-06-1988)
[167] (1988-09-15) GPU Safety Evaluation, Completion of Upper Support Core Assembly Defueling, Rev. 0
[168] (1986-05-15) GPU Safety Evaluation, Defueling SER, Rev. 10
[169] (1988-06-27) GPU Safety Evaluation, Completion of Lower Core Support and Lower Head Defueling, Rev. 1
[170] (1989-04-04) NRC Safety Evaluation, Completion of Upper Core Support Assembly Defueling (re 9-15-1988)
[171] NUREG-0591, Environmental Assessment, Use of EPICOR-II at TMI-2 (1979-08)
[172] NUREG-0683, Vol. 1, PEIS-Decontamination and Disposal of Radioactive Wastes Resulting from TMI-2 (1981-03)
[173] (1981-03-11) GPU Technical Evaluation, SDS, Revised
[174] NUREG-0591, Environmental Assessment, Use of EPICOR-II at TMI-2 (1979-08)
[175] (1982-03-23) GPU Technical Evaluation, Addendum to SDS, RCS Processing Plan
[176] NUREG-0796, Operation of the Submerged Demineralizer System at TMI-2 (1981-06)
[177] (1981-03-11) GPU Technical Evaluation, SDS, Revised
[178] NUREG-0683, Vol. 1, PEIS-Decontamination and Disposal of Radioactive Wastes Resulting from TMI-2 (1981-03)
[179] NUREG-0796, Operation of the Submerged Demineralizer System at TMI-2 (1981-06)
[180] (1981-05-15) GPU Response to NRC (Discussions), Man-Rem Exposures for Specific SDS Operations
[181] NUREG-0683, Vol. 1, PEIS-Decontamination and Disposal of Radioactive Wastes Resulting from TMI-2 (1981-03)
[182] NUREG-0796, Operation of the Submerged Demineralizer System at TMI-2 (1981-06)
[183] NUREG-0796, Operation of the Submerged Demineralizer System at TMI-2 (1981-06)

[184] Arnold, R.C., Metropolitan Edison Company, letter to B.H. Grier, USNRC, "Radiation Safety Program Report," TLL-054, February 8, 1980

[185] (1980-01-24) GPU, TMI-2 Radiation Protection Plan, Rev. 1

[186] (1981-05-11) GPU System Description, SDS, Rev. 2

[187] (1981-05-11) GPU System Description, SDS, Rev. 2

[188] (1981-05-11) GPU Response to NRC (Discussions), SDS Radiation Protection Plan

[189] USNRC, "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Plants Will Be as Low as Is Reasonably Achievable," Regulatory Guide 8.8, Rev. 3, June 1978 [Current and past regulatory guides are available at nrc.gov]

[190] USNRC, "Quality Assurance Program Requirements (Operation)," Regulatory Guide 1.33, Rev. 2, February 1978 [Current and past regulatory guides are available at nrc.gov]

[191] USNRC, "Operating Philosophy for Maintaining Occupational Radiation Exposures as Low as Is Reasonably Achievable," Regulatory Guide 8.10, Rev. 1-R, May 1977 [Current and past regulatory guides are available at nrc.gov]

[192] USNRC, "Qualification and Training of Personnel for Nuclear Power Plants," Regulatory Guide 1.8, Rev. 1-R, May 1977 [Current and past regulatory guides are available at nrc.gov]

[193] (1982-10-13) GPU Safety Evaluation, SDS Liner Recombiner and Vacuum Outgassing System

[194] (1982-11-11) GPU Response to NRC (Discussions), SDS Liner Recombiner and Vacuum Outgassing System

[195] (1982-11-23) NRC Review, SDS Liner Recombiner and Vacuum Outgassing System

[196] NUREG-0796, Operation of the Submerged Demineralizer System at TMI-2 (1981-06)

[197] USNRC, "Three Mile Island Accident of 1979 Knowledge Management Digest: Recovery and Cleanup," NUREG/KM-0001, Supplement 1, June 2016 [Available at nrc.gov]

10 RADIOLOGICAL RELEASE SAFETY EVALUATIONS

TABLE OF CONTENTS

Note: “NA” (not applicable) means the licensee and the NRC determined that the safety topic was not important for the activity.

10.1 Introduction

10.1.1 Background

The broad safety topic of radiological releases covered essentially the entire fuel cycle, ending with storage or disposal. This chapter focuses on the safety evaluations of radiological releases and resulting offsite doses during postaccident TMI-2 cleanup. These evaluations included normal operations and accident conditions.

- ***Recovery and Early Cleanup***. The issue of radiological releases at TMI-2 was of interest to the NRC, licensee, State and local governments, and the public. The interest was most intense during the first year after the accident. Early sources of radioactive materials included: (●) large quantities of radioactive krypton-85 gas (about 44,000 curies) inside the containment building [(1)] ; (●) highly contaminated water in the containment building's basement (about 630,000 gallons of water that contained about one-half of the core inventory of the radionuclide cesium-137) [(2)] ; (●) inadvertent (unplanned) releases of small quantities of radioactive gases that occurred during routine maintenance activities; (●) about 66 to 80 curies of krypton gas that leaked out of the containment building every month [(a)] ; (●) wastewater storage near capacity and contaminated with accident-generated water [(3)] ; and (●) operational leakage from support systems located in the auxiliary and fuel handling building, which added to the mix about 800 to 1000 gallons a day of mostly uncontaminated water. [(4)]

The NRC responded in several ways during the first few years with the issuances of policy statements, interim release criteria, and environmental impact statements.

- *1979 Policy Statement*. Almost 2 months following the accident, the NRC Commissioners issued a policy statement that prohibited the treatment or discharge of contaminated water, except for certain routine operational releases, until completion of an environmental assessment. [(5)] This policy was in response to a civil action lawsuit by the City of Lancaster intended to prevent the contamination of the city's water supply, which was taken from the Susquehanna River downstream from TMI-2. [(b)] However, the NRC's interpretation of the policy's prohibitions hampered the licensee's ability to obtain timely information and data to plan for later cleanup steps and handle unforeseen consequences. [(6)] In response to the recommendation of the NRC internal special task force on the TMI-2 cleanup, the Commissioners approved the interim criteria for radiological releases from such activities in April 1980. [(7)]

---

[a] The partial vacuum on the secondary side of the steam generator caused a pressure difference between the containment building and turbine building, which enhanced leakage through the packing of various steam valves. These gases were subsequently discharged from the secondary system through the auxiliary building ventilation system to the environment.

[b] The City of Lancaster's case was settled on February 28, 1980. The NRC Commissioners reiterated the Commission's intent to prepare a programmatic environmental impact statement and agreed that no accident-generated wastewater would be discharged into the Susquehanna River until completion of that statement or such other environmental review as was contemplated by the Commission's November 21, 1979, policy statement, or until December 31, 1981, whichever was earlier.

- *1980 Environmental Assessment (Containment Building Purge)*. In May 1980, the NRC issued the "Final Environmental Assessment for Decontamination of the Three Mile Island Unit 2 Reactor Building Atmosphere" (NUREG-0662). (8) The NRC Commissioners immediately issued an order (9) that authorized the licensee to clean the containment building's atmosphere by means of a controlled purge in June 1980.

- *1980 Policy Statement and Interim Release Criteria*. In November 1980, the NRC Commissioners issued a policy statement (10) for the preparation of a "programmatic" environmental impact statement on the decontamination and disposal of radioactive waste resulting from the accident. The program-level environmental review would cover all planned and anticipated cleanup activities, including cleanup alternatives.

- *1981 Programmatic Environmental Impact Statement*. In January 1981, the NRC issued the programmatic environmental impact statement (PEIS) (11) on the decontamination and disposal of radioactive wastes at TMI-2. The NRC Commissioners endorsed the PEIS in their policy statement (12) in April 1981. The policy allowed the NRC staff to act on each major cleanup activity without the Commissioners' approval if the activity and associated impacts fell within the scope of those assessed in the PEIS. The PEIS generally covered every aspect of the cleanup campaign. The PEIS became the guiding document for all licensee and NRC safety evaluations involving cleanup activities.

- ***Cleanup (Including Defueling)***. Radiological releases were a safety consideration for most safety evaluations of proposed cleanup activities. Evaluations included offsite dose analyses from releases during routine or planned operations and from postulated accidents. The main concern was releases to the atmosphere. Liquid releases to the environment were less of a concern because of the NRC restrictions of such releases and because the liquid waste systems and storage tanks were contained in buildings. Further, an earlier study (13) on the migration of radioactivity from a leak of accident-generated water from the containment building basement concluded that predominant radionuclides would take many years to migrate through the bedrock to the nearby river.

Primary source terms at TMI-2 were noble gases (krypton-85), tritium (H-3), and particulates (cesium-134, cesium-137, strontium-90, and others to a lesser extent). Sources of radioactivity evaluated in the PEIS and safety evaluations included: (●) highly contaminated, accident-generated water in the containment building sump and storage tanks in the auxiliary and fuel handling building; (●) small quantities of krypton-85 gas from gas pockets in the fuel debris bed and enclosed components; (●) tritium in water vapor from the natural evaporation of accident-generated water and processed water used as shielding for spent fuel pools and fuel transfer canal; and (●) airborne radioactive particulates from the plateout on surfaces inside the containment building and auxiliary and fuel handling building.

The safety evaluations also included assessments of the mitigation of radiological releases. These mitigating measures generally included: (●) ventilation using high-efficiency particulate air (HEPA) filters; (●) defueling work platform off-gas system; (●) processing of contaminated water in the reactor vessel and spent fuel pools; (●) leak prevention; (●) decontamination of surfaces to remove sources of airborne contamination; (●) prevention of accidents such as load drops; (●) monitored releases to the environment; and (●) monitoring, isolation, and containment.

During the cleanup campaign, no planned or unplanned releases from TMI-2 exceeded the limits of the plant's environmental technical specifications or of 10 CFR Part 20, "Standards for Protection against Radiation." This chapter summarizes the licensee's and the NRC's safety evaluations associated with radiological releases and resulting offsite exposures.

10.1.2 Chapter Contents

This chapter presents radiological release evaluations of the postaccident TMI-2 and defueling operations. It describes the many studies performed to give the reader an understanding of the thinking of the analysts at the time, the expectations and the reality, the uncertainties in the data, and the measurement and mitigation methods. It also presents a high-level timeline of cleanup activities.

The evaluations presented in this chapter ensured that all activities that could result in planned and inadvertent releases of radioactive materials to the environment under normal operations and accident conditions were addressed and consequences evaluated; controls were maintained in accordance with the requirements of the plant's license, technical specifications, procedures, and applicable regulatory requirements; and adequate contingencies were developed for normal and accident conditions.

Radiological release was a concern in most cleanup activities and system operations. This chapter excludes: (●) planned containment building purge of krypton-85 in 1980 (refer to NUREG-0662, "Final Environmental Assessment for Decontamination of the Three Mile Island Unit 2 Reactor Building Atmosphere" [(14)]); (●) radiological effluents from early data-gathering and maintenance operations (refer to NUREG-0681, "Environmental Assessment of Radiological Effluents from Data Gathering and Maintenance Operation on Three Mile Island Unit 2" [(15)]); (●) planned disposal of decontaminated accident-generated water through evaporation (refer to Supplement 2 of the PEIS [(16)]); and (●) effluents during post-defueling monitored storage of TMI-2 (refer to Supplement 3 of the PEIS [(17)]).

Additional evaluations of radiological release from heavy load drops can be found in NUREG/KM Chapter 7 on lead drop evaluations.

Section 2 summarizes a key study that was used to support safety evaluations. The following sections present the safety evaluation for each applicable cleanup activity or system. Section 8 lists endnotes showing references cited throughout this chapter.

10.2 Key Reports

10.2.1 Programmatic Environmental Impact Statement
(NUREG-0683, USNRC, March 1981)

- ***Background***. The NRC's PEIS was an important document for the licensee and the NRC in their safety evaluations of cleanup activities. The report was cited in all safety evaluation reports and NRC approval letters. In 1981, the NRC Commissioners endorsed the PEIS in their policy statement that concluded that the PEIS satisfied the NRC's obligations under the National Environmental Policy Act (NEPA). The policy statement also stated that, as the licensee proposed specific major decontamination activities, the NRC staff would determine whether these proposals and their predicted impacts fell within the scope of those already assessed in the PEIS. With the exception of the disposition of processed accident-generated water (the Commissioners wanted to decide this issue later), the staff was allowed to act on each major cleanup activity without the Commissioners' approval if the activity and associated impacts fell within the scope of those assessed in the PEIS. [18]

The PEIS became a guiding document for all licensee and NRC safety evaluations to ensure that each major cleanup activity would be bounded within the scope of those assessed in the PEIS and supplements. [19] As required in a NEPA assessment, the PEIS provided assessments of radiological releases to the environment for all projected cleanup activities during normal operations and postulated accidents. In addition, these assessments included associated occupational and offsite (public) dose estimates.

- ***Scope*** *(Editor's Note)*. The PEIS described sources of release of radioactivity, source terms, measures for the prevention of releases, and onsite and offsite dose estimates from normal operations and postulated accidents. PEIS evaluations of cleanup activities included in this chapter of the NUREG/KM include: (●) maintenance of the reactor in safe condition; (●) containment building and equipment decontamination; (●) reactor coolant system (RCS) inspections; (●) removal of the reactor vessel head and internals; (●) core examination and defueling; (●) decontamination of primary system components; and (●) liquid waste treatment.

Activities that were covered in the PEIS but not in this chapter include: (●) auxiliary and fuel handling building and equipment decontamination (refer to Section 5.1 of the PEIS); (●) disposal of processed accident-generated water (refer to Supplement 2 of the PEIS); (●) waste packaging and handling of process solid wastes (refer to Section 8.1 of the PEIS); (●) waste packaging and handling of chemical decontamination solutions (refer to Section 8.2 of the PEIS); (●) waste packaging and handling of solid materials (refer to Section 8.3 of the PEIS); and (●) waste packaging and handling of fuel assemblies and core debris (refer to Section 8.4 of the PEIS).

The following summary includes only certain parts of the detailed assessments from the PEIS. To avoid reproducing the entire PEIS, only a few cleanup activities are included to provide a general perspective of the assessments that were performed for radiological releases. Further, the complete description of the activity, including details of the work, equipment, and facilities, is not included in the summary. However, the evaluations of radiological releases and offsite exposures for those selected activities are provided almost in their entirety. The interested reader should refer to the PEIS for the unabridged version of the safety considerations for each activity. This summary does not include Supplement 1 of the PEIS, [20] which updated the assessments of occupational doses.

The methods used in the PEIS for estimating the radionuclide concentrations and associated offsite dose could have been simplified or may be outdated. In addition, the cleanup plans assessed during the writing of the PEIS may not reflect the actual activity that was conducted years later. Nevertheless, the NRC's safety reviews throughout the decade determined that all proposed activities were bounded by those presented in the PEIS [c] and its supplements.

- ***Analysis Methods***. The calculational models and parameters used in estimating doses and interpreting model results were provided in Appendix W to Volume 2 of the PEIS. The appendix provided calculational methods used to estimate doses from atmospheric and liquid releases due to planned cleanup activities and postulated accidents. Key assumptions used to estimate doses from atmospheric releases are summarized below. Refer to the PEIS for a complete description of the methods that were used to calculate doses form releases to the atmosphere and river.

  - *Assumptions*. The calculational methods used to estimate doses from atmospheric releases due to routine decontamination or accidental atmospheric releases were those described in Regulatory Guide 1.109, [d, 21] "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I." The NRC found that this regulatory guide was also appropriate for calculating short-term releases with certain minor adjustments. These adjustments and assumptions included the following:

    – *Release and Consumption Time Periods*. The period of time that an individual consumed contaminated food was 1 year. Releases occurred uniformly over a period of a year.

    – *Dispersion Parameters*. The hourly atmospheric dispersion parameter values for the location resulting in the highest doses were used. The atmospheric dispersion parameters were selected to ensure that calculated doses overestimated potential actual ones. For example, the dispersion value used for the nearest residence and garden (food) calculations was not selected on the basis of the actual residence and garden

[c] Editor's Note: The PEIS was supplemented three times, first to update occupational exposure estimates (1984), then to update the environmental impacts of the disposition of processed accident-generated water (1987), and then again to place the facility into monitored storage for an unspecified period of time (1989). The latter two supplements were prepared and issued before the submission of the licensee's proposals. The supplement for occupational exposure estimates was in the anticipation of increase in the accumulated dose of future activities; however, by the end of the cleanup, the original exposure estimates were found to be adequate.

[d] 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," Appendix I, "Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion 'As Low as is Reasonably Achievable' for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents."

nearest to TMI, but the value was selected on the basis of an actual residence and garden where the relative annual meteorological deposition rate was highest.

- *Consumables*. For population dose estimates, all milk, meat, and vegetables consumed by this population were assumed to be produced uniformly within the 50-mile radius. Consumption rates were those of Table E-4 of Regulatory Guide 1.109.

- *Release Assessments Methods*. The long-term (continuous) release assessments used methods described in Regulatory Guide 1.111, [22] "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors," and the short-term release assessments used methods described in Regulatory Guide 1.145, [23] "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants."

- *Uncertainties*. For the maximum concentration of radioactivity in a plume at short downwind distances (within 10 kilometers (km)), an uncertainty of ± 20 percent was applied to account for a ground-level release, generally uniform terrain, and steady winds.

  The ratio of the predicted concentration to observed concentration for specific locations within 10 km ranged from 0.1 to 10.0 with the Gaussian plume model.

  In complex wooded terrain conditions, such as around TMI, the Gaussian model for short-term releases overestimated the concentrations under poor diffusion conditions of stable atmosphere and low windspeeds when compared to actual measurements. The amount of overestimation could range from 50 to 500 times the value observed. Annual average release evaluations were shown generally to be within a factor of 2 to 4 for regional distances (out to about 100 km), depending on the terrain conditions.

- *Deposition Rate*. The method that was used to determine the accident relative deposition rate assumed, at the time of the release, 100 percent of the effluent went to the receptor location during the entire release. The deposition analysis used a method described in Regulatory Guide 1.111.

- *Decay Adjustment*. No decay adjustments were provided for the accident atmospheric dispersion factors due to the brief duration of a short-term release.

- *Population Dose Calculations*. No population dose calculations were made for accidental atmospheric releases.

o *Results*. Results for the maximum exposed individual were presented in the PEIS for three locations (nearest garden, nearest milk goat, and nearest cow and garden), all of which were about 1 mile from the release point; three or four exposure pathways (inhalation, ground shine, vegetable and/or milk use); and doses to the total body and organs (gastrointestinal tract, bone, liver, kidney, thyroid, lung, and skin). Doses were calculated for four age groups (adults, teenagers, children, and infants). The summaries below report only the highest total location dose (i.e., the total of all dose results of total body and organs from all exposure pathways at a specific location). The PEIS presented the results in tables.

- ***Maintenance of Reactor in Safe Condition***. The maintenance of the TMI-2 reactor in a safe condition included: (●) achievement of a thermally stable primary system in which the decay heat from fission products was continually being removed; (●) maintenance of subcriticality of the reactor core; and (●) confinement of the radioactivity within the containment building.

  - *Normal Releases (Gaseous)*. Early in the recovery, steam generator "A," which operated in the steaming mode for 13 months, removed decay heat. In the steaming mode, the turbine side of the steam generator was maintained in a partial vacuum by the condenser air ejectors. Consequently, the pressure difference between the containment building atmosphere and the turbine side of the steam generator enhanced leakage of krypton-85 from the containment building through the packing of various steam valves to the secondary system. Before venting of the containment building in June 1980, the early PEIS reported that about 66 to 80 curies of krypton-85 leaked out of the containment building every month. The kyrpton-85 gas and other gases (nonradioactive) were subsequently discharged from the secondary system through the auxiliary building ventilation system to the environment. Following the massive purge in 1980, the containment building was periodically purged of kyrpton-85 to lower occupational exposures of workers inside the containment. Small amounts of kyrpton-85 were released from other sources within the plant (e.g., the auxiliary building ventilation system). The release rates were near or at the limits of radiation equipment detection and were less than 1 curie per month.

  - *Normal Releases (Liquid Wastes)*. Similar to operations in other commercial nuclear power plants, the auxiliary and fuel handling building floor drain systems collected nonradioactive water that became contaminated by radioactive material present in the building. As a result of cleanup efforts, the volume and radioactivity content of routine operational liquid wastes at TMI were similar to those of a normally operating power reactor. Wastewater sources included river water leakage from pump seals; air conditioning condensation; closed (nonradioactive) cooling water system leakage; demineralized water (nonradioactive) flush water; personnel showers; secondary plant sampling; and chemistry laboratory drains. About 250 gallons per day were generated (refer to Table 4.1 of the PEIS). Before discharge to the river, [e] the slightly radioactive wastewater would be treated by an ion exchange system and sampled to ensure compliance with all regulatory criteria. The radioactivity content of the treated water would be less than that at a normally operating nuclear power plant (refer to Table 4.2 of the PEIS).

  - *Releases (Accident Conditions)*. The categories of core-related accidents that could result in the release of additional radioactive fission products from the damaged fuel in the reactor

[e] Editor's Note: In the endorsement of the PEIS in their April 1981 policy statement, the NRC Commissioners stated that any future proposal for disposition of processed accident-generated water would require the Commissioners' approval.

core included: (●) criticality due to boron dilution; (●) core cooling failures; and (●) leakage of containment building sump water.

– *Criticality*. Subcriticality would be maintained by ensuring that boron concentration in the RCS met the recovery technical specification limits and that measures would be in place to prevent inadvertent boron dilution. Most of the fission products produced in a postulated criticality accident would be extremely short lived. The containment building was specifically designed to contain fission product inventories that were orders of magnitude greater than the current core inventory at containment building pressures far greater than those that would exist during defueling.

– *Core Cooling Failures*. Loss of core cooling could occur from two causes: loss of electrical power or reactor vessel leakage. To improve the reliability of electrical power supplied to nonsafety grade decay heat removal equipment, an additional offsite transmission line and two redundant diesel generators were installed at TMI-2. An accident scenario involving a leak in the reactor vessel that could lead to uncovering and overheating of the core could be mitigated in time because of the very low decay heat rate. However, if overheating was assumed to occur and all the remaining krypton-85 and cesium were released to the containment building, the total activity from this hypothetical release would be about the same as the activity inside the containment building before the postaccident purge. [f] Given the long time available to take corrective actions (many hours to several days), the containment building, if occupied at the time, could be evacuated and isolated before any significant release from the core had occurred.

– *Leakage of Containment Building Sump Water*. The largest amount of contaminated water on the site at the time the PEIS was written was the 700,000 gallons of highly contaminated water in the basement of the containment building. This water contained an estimated 500,000 curies of radionuclides. The evaluation postulated that if this water should leak through the steel-lined concrete base of the containment building, the water would ultimately reach the nearby Susquehanna River. However, the NRC stated in the PEIS that such an accidental release was considered highly unlikely. Appendix V, "Assessment of Groundwater Liquid Pathway from Leakage of Containment Water at TMI-2," to Volume 2 of the PEIS discussed this pathway in detail.

– *Detection and Mitigation*. In the unlikely event that water from the containment building basement should leak through these barriers, several methods for early detection and control of this leakage were available. Indicators included: (●) sump water level; (●) radioactivity in the foundation tendon gallery; (●) radioactivity in the site monitoring wells; and (●) results from the environmental monitoring program. Leakage from the containment building basement could be controlled by any of the following measures: (●) The remaining water in the sump could be pumped into various available storage

[f] Editor's Note: The PEIS assumed that most of the fuel assemblies in the damaged reactor core were intact. The videos from the first Quick Look examination in 1982 revealed the severity of the damage.

tanks. (●) Contaminated water could be pumped out of the surrounding monitoring wells. (●) A grout curtain could be installed between the containment building and the Susquehanna River.

- *Assumptions*. For the purposes of analyzing the transport of strontium-90 and cesium-137 from a containment building basement leak, the analysis conservatively assumed that the water was released into the ground over an area the size of the entire base of the containment building. Such a postulated release could drain only that portion of the contaminated water that was above the water table. To simplify the calculation, 470,000 gallons were assumed to enter the ground water instantaneously. For tritium, the analysis depended on a somewhat different set of phenomena, and the entire 470,000 gallons were conservatively assumed to be released instantaneously from a point at the center of the building.

- *Results*. After the initial holdup in the ground, the wave front of the strontium-90 and cesium-137 would reach the river in 23 years and 284 years, respectively, and the wave would continue to enter the river over periods of about 8.5 and 140 years, respectively. By the time the strontium-90 reached the river, radioactive decay would have reduced the total available activity to 2660 curies. When the cesium-137 reached the river, its total activity would have decayed to 416 curies. The maximum annual average concentration during a given year would be close to the peak concentrations for these two nuclides. The calculated river concentrations were orders of magnitude below maximum permissible concentrations limits in 10 CFR Part 20.

- *Offsite Doses*. The PEIS provided dose estimates for maintaining the reactor in a safe condition from gaseous and liquid releases.

  - *Gaseous Release Doses*. The dose estimates from gaseous releases were based on krypton-85 source terms (refer to Section 4.4.1 of the PEIS). Releases of krypton-85 were assumed to be made at the level of 67 curies per month, which was equivalent to an annual level of about 800 curies per year. The estimate of total body dose to the maximally exposed individual offsite due to release of this amount of krypton-85 was expected to be less than 0.001 millirem. The 50-mile total body population dose for this release was estimated to be 0.03 person-rem. These individual and population doses were based on the external exposure pathway (from krypton-85). Lung doses were expected to be higher than total body doses by about a factor of 3, and skin doses were expected to be higher by a factor of about 80 over total body doses.

  - *Liquid Release Doses*. The dose estimates from the release of treated routine operational liquid wastes were based on cesium-134, cesium-137, strontium-90, and tritium (refer to Section 4.4.1.2 of the PEIS). The dose estimates to the maximally exposed individual were 0.0015 millirem per year total body dose and 0.002 millirem per year bone and liver doses. The downstream total body population dose was estimated to be less than 0.02 person-rem per year.

- ***Containment Decontamination***. Effluents and releases that were associated with decontamination of the containment building and equipment included gases (primarily krypton-85), solids, liquids, tritium, and particulates. Releases from postulated accidents were also considered.

  - *Normal Releases (Gaseous)*. The largest release to the environment from the containment building due to cleanup activities was 44,000 curies of krypton-85 that was purged (vented) to the environment between June 28 and July 11, 1980. These releases were controlled to limit the cumulative maximum individual offsite doses to the public resulting from the purge to less than the annual dose design objectives (15 millirem to the skin, 5 millirem total body) of Appendix I to 10 CFR Part 50.

    Subsequent periodic releases containing much smaller amounts of krypton-85 generated by outgassing, which was primarily from the contaminated water in the containment building basement. These planned releases minimized worker exposure in the containment building during decontamination operations. The PEIS assumed that based on actual data, 15 curies per month was a conservative upper bound for the krypton-85 releases from the containment building during decontamination operations

  - *Normal Releases (Solids)*. Solids generated by decontamination activities in the containment building came from: (●) sludge debris and contaminated equipment removed from the building; (●) contaminated filters from vacuuming operations; (●) solid residue removed from contaminated liquids by liquid processing operations; and (●) contaminated laundry and materials (e.g., wipes) used for decontamination operations. The effluents and releases from these materials were associated with waste disposal operations, which were discussed in Section 8 of the PEIS.

  - *Normal Releases (Liquids)*. Decontamination activities within the containment building initially would not result in any direct releases of liquid to the environment. The NRC assumed that the most likely volume of liquids that would be generated by all building and equipment surface decontamination operations was 150,000 gallons of water and 40,000 gallons of detergent solution. The NRC estimated that at least 90 percent of the plateout in the containment building would be removed in the water, and most of the remainder would be removed in the detergent solution. Section 7 of the PEIS discussed the processing and disposal of these liquids. The operations were expected to last 18 to 54 months.

  - *Normal Release (Tritium)*. If the water in the containment building basement were not removed before building decontamination operations or isolated by a vapor barrier from the rest of the building, most of the tritium released from the building would be tritiated water vapor from evaporation of the water in the basement. The maximum possible release of 2500 curies of tritium would occur if all of the water in the basement were removed by evaporation. If saturation conditions (100 percent humidity) could be maintained with the building ventilation system in operation (at 50,000 cubic feet per minute), the entire 700,000 gallons of water would evaporate in about 1 year. (This would correspond to an evaporation rate of about 0.0001 pounds of water per square foot per minute.) The concentration of tritium in the air under these conditions would be $2x10^{-5}$ picocurie per milliliter, which was larger than the guidelines for restricted areas in 10 CFR Part 20 by a factor of 4.

These limiting conditions were unrealistic and unattainable; therefore, the NRC considered an evaporation rate of about 20 percent of this limit to be a reasonable bounding case if the building ventilation system were in operation and no special precautions were taken to prevent evaporation of the water. If this situation continued for a period of 18 months, about 750 curies of tritium in the form of tritiated water vapor would be released to the containment building atmosphere. Under these conditions the average concentration of tritium in the containment building atmosphere would be $4x10^{-6}$ picocurie per milliliter.

The NRC estimated that the amount of tritiated water that would evaporate and be released from semiremote operations would be about 2 curies. This represented a lower bound to the release of tritium than would occur if the water in the basement were removed before decontamination or effectively isolated. Under these conditions, the average concentration of tritium in the containment building atmosphere would be negligible.

- *Normal Release (Particulates)*. The NRC estimated total airborne radioactivity discharged to the environment as a consequence of decontamination operations in the containment building to be about $1x10^{-4}$ curie (excluding tritium). The ventilation flow rate was assumed to be 50,000 cubic feet per minute. The air cleaning system contained two stages of high-efficiency particulate air (HEPA) filtration. A penetration factor of 0.001 was assumed in calculating the releases. The penetration factor was the fraction of material entering a filter that passes through the filters. Thus, for a given filter, the penetration factor equals 1 minus the filter efficiency when the filter efficiency is expressed as a fraction. This factor was based on NRC Regulatory Guide 1.140, [24] "Design, Testing, and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Absorption Units of Light-Water-Cooled Nuclear Power Plants," which specified a very conservative penetration factor of 0.01 (corresponding to 99-percent efficiency) for the entire exhaust system and then only if the HEPA filters test to an efficiency of 99.95 percent or greater. The regulatory guide gave no additional credit for HEPA stages in series.

  Table 5.10 of the PEIS lists the release concentrations of principal radionuclides (i.e., cesium-134, cesium-137, and strontium-90). These estimates were regarded as upper bounds. If a more realistic estimate, based on the use of a penetration factor of 0.001 for each of the two filters in series (i.e., a total penetration factor of 10^{-6}) were used, all numbers in Table 5.10 would be reduced by a factor of 1000.

- *Normal Release (Particulate Plateout)*. Both sets of normal release estimates (crediting one or both HEPA filters) were based on the amount of airborne contamination dislodged from plateout on upper floor level surfaces and sludge on the basement floor.

- *Surface Plateout*. The NRC estimated the amount of plateout on readily accessible surfaces on the 305-foot (entry level) and higher elevations of the containment building to be 2 curies. An uncertain additional amount was present in less accessible locations such as in floor drains and in the basement (counting only the contaminated film that would remain after the sludge was removed) and inside the D-rings, [g] and the NRC expected that this would probably be much less than 100 curies. This plateout would be dissolved or suspended in the decontamination liquids and removed with them. For computational purposes, the NRC conservatively assumed that about 0.1 percent of the surface plateout on the 305-foot and higher elevations could become airborne during decontamination operations (primarily during water jet operations). The fraction of contamination from less accessible locations on the 305-foot elevation and above that became airborne would be less; the NRC made the conservative assumption that an equal contribution to airborne contamination might occur from these sources. Conditions in the basement were largely unknown. In view of the relatively small volume of the basement (less than 5 percent of the total volume of the containment building) and area (about 20 percent of the total area), the NRC assumed that the contribution to airborne contamination from the basement area would not exceed the contribution from the 305-foot and higher elevations.

- *Basement Sludge*. The sludge, which was estimated to have a wet volume of about 100 cubic feet and a radioactive content of about 160 curies, would be removed primarily by resuspension in the sump water before the sludge was pumped out. The contribution of this operation to airborne releases would be negligible compared to other contributions. It could be necessary to dislodge some fraction, up to about 5 percent of the total, by use of a water jet. The NRC again conservatively assumed that up to 0.1 percent of this residue could become airborne during this operation.

o *Releases (Accident Conditions)*. The types of accidents that could occur during containment building decontamination included: (●) a liquid spill, either before or after processing the liquid; (●) dropping of a solid waste package; (●) a fire in a barrel of contaminated trash; and (●) failure of HEPA filters in the ventilation system. Section 7 of the PEIS discussed effluents and releases from accidents associated with processing the liquids from the containment building decontamination. Section 8 of the PEIS discussed effluents and releases from accidents in the handling of solid wastes.

  A failure of one of the HEPA filters in the ventilation system would increase the airborne particulate release described in Table 5.10 of the PEIS until the ventilation system was secured (15 minutes). Based on the analysis in Section 7 of the PEIS, Table 5.11 of the PEIS shows the estimated airborne release amounts and concentrations from a HEPA filter failure during decontamination of the containment building surfaces.

[g] D-rings were shield enclosures around the steam generator compartments; they were so named because of their shape.

- *Offsite Doses (Normal Operations)*. Offsite doses were estimated for the first large purge of krypton gas in the containment building and subsequent purges of much smaller quantities of krypton gas.

  – *1980 Purge*. The point of maximum exposure during the June 28 to July 11, 1980, purge was about 0.4 mile from the site in an east-southeast direction. If a person had remained at this location throughout the purge, that person would have received a beta skin dose of 4.5 millirem and a whole-body gamma dose of 0.05 millirem.

  – *Other Purges*. Dose estimates for decontamination operations in the containment building subsequent to September 1, 1980, were based on the source terms listed in Table 5.10 of the PEIS. The dose estimates that were based on airborne releases of principal radionuclides for the tritium lower (2 curies) and upper (750 curies) range were presented in the PEIS (refer to Tables 5.11 and 5.12 of the PEIS). The highest total location dose for the maximum exposed individual was 0.34 millirem to the total body in children at upper tritium range. The 50-mile whole-body cumulative population dose received by the human population during these activities was estimated to be 0.02 to 6 person-rem at the lower and upper tritium range, respectively.

- *Offsite Doses (Accident Condition)*. The type of accident postulated in the analysis that would result in offsite doses was the rupture of a HEPA filter during containment building decontamination. Table 5.11 of the PEIS lists the source terms for this accident scenario. The highest total dose for the maximum exposed individual ranged from 0.05 to 0.07 millirem to the bone in children and adults at the three locations analyzed (nearest garden, nearest milk goat, and nearest cow and garden; refer to Table 5.14 of the PEIS).

• ***Reactor Coolant System Inspections***. The objectives of RCS inspections were to determine the condition of the core and provide ongoing monitoring of important RCS parameters during the dismantling and cleanup processes. Inspections would be coordinated with the containment building decontamination; removal of the reactor vessel head; defueling; and primary coolant system decontamination. The NRC concluded that virtually no radioactive effluents would be released as a result of inspection activities. Offsite impacts would be indistinguishable compared to defueling.

• ***Removal of Reactor Vessel Head and Internals***. Removal of the reactor vessel head and internal components was necessary to gain access to the reactor core for defueling and to the bottom of the reactor vessel to remove fuel debris. Before the head was removed, other decontamination activities directly related to head and internals removal would have to be completed. These prerequisite activities included: (•) decontamination of primary water and reduction of the chloride concentration to a level sufficient to proceed with head and internals removal; (•) decontamination of necessary containment building support equipment and systems and putting them into satisfactory working condition; and (•) work area decontamination accomplished in accordance with ALARA principles. Also, the core could be inspected by insertion of viewing devices through a port in the head and internals before their removal.

- *Normal Releases (Gaseous)*. Before removal of the head, attachments would be made at various points on the head to remove all pockets of radioactive gas that could be trapped inside the reactor vessel. The vessel would be purged to sweep out these gas pockets, and the gas would be stored or vented to the environment as deemed advisable. The NRC estimated that this process would release a maximum of 100 curies of krypton-85 gas, essentially the only radioactive gas present. Such a release was not significantly different from previous low-level krypton-85 releases vented from TMI-2 from time to time.

- *Normal Releases (Tritium)*. During certain portions of these operations, the fuel transfer canal and possibly the spent fuel storage pool would be filled. One option was to fill these structures with the decontaminated water from processing of the liquids in the auxiliary and fuel handling building tanks, the containment building sump, and the RCS. If this processed water was used, the water would contain a tritium concentration of about 0.5 picocurie per milliliter. Evaporation from the surface of these storage pools would contain tritium and pass through the filters in the ventilation systems. For the containment building, with a ventilation flow of 50,000 cubic feet per minute, an evaporation rate of 0.001 pound per minute per square foot, and a surface area of 1630 square feet (transfer canal), the effluent concentration would be 2.6×10^{-7} picocurie per milliliter. The fuel handling building ventilation flow was 36,000 cubic feet per minute, and the storage pool area was 2350 square feet. Using the same evaporation rate, the effluent concentration of tritium would be 5.3×10^{-7} picocurie per milliliter. Based on these ventilation rates and effluent concentrations of tritium, the total tritium that would be released to the atmosphere from the containment building and fuel handling building for 1 year would be 500 curies.

- *Normal Release (Particulates)*. Because almost all the operations would be conducted underwater, few if any new sources of airborne particulate activity would be expected. However, depending on the level of decontamination achieved during cleanup of the containment building, some airborne particulate activity could be generated simply by workers and equipment moving around in the building during removal of the head and internals. The resulting effluents, if any, would be much less than during the original building decontamination effort.

- *Releases (Accident Conditions)*. The main concerns during the removal operations were drop accidents that could lead to a loss of water or mechanical damage to the core. Careful monitoring and control of boric acid concentration would ensure that the reactor remained shut down. The only fission product that could be released and become airborne as a result of mechanical damage to the core would be krypton-85. Although the cladding on many of the fuel elements was expected to have failed, a few of the elements could be intact and still contain trapped krypton-85. The inventory of krypton-85 in a single fuel element was about 1.5 curies, of which only a few percent was free gas in the gap that could be released by rupturing the cladding.

  The other possibility was a bubble of fission gas trapped in a damaged region of the core that was released by dropping something on top of the core. Since the pressure in such a bubble would be about 2 atmospheres (i.e., under about 30 feet of water) a bubble 1 liter in volume would contain about 0.1 moles of gas. The composition of fission gas from low-burnup fuel after 1 year of decay was 99 percent stable isotopes of krypton and xenon and about 1 percent krypton-85. Thus, a 1-liter bubble could contain up to 0.001 mole, or 35 curies, of krypton-85 based on pure fission product gas. Since the volume of the containment building was about 2 million cubic feet, the release of 35 curies of krypton-85 would result in a concentration of about 6×10^{-4} picocurie per milliliter inside the building. Assuming a ventilation rate of 50,000 cubic feet per minute, 90 percent of this krypton-85 would be discharged to the atmosphere in about 100 hours.

At this stage of the operation, a drop accident that caused a failure of the seal plate would result in drainage of the transfer canal to the containment building basement but would have no significant radiological consequences. This conclusion was based on the canal water containing very little radioactivity (0.01 picocurie per milliliter, exclusive of tritium) and there being no fuel in the transfer canal.

- *Offsite Doses (Normal Operations)*. The dose estimates in the PEIS for removal of the reactor vessel head and internals were based on the source terms for removal of the head (100 curies of kyrpton-85) and evaporation of water from the containment building and fuel handling building (500 curies of tritium per year). The dose from particulates was also considered.

  - *Krypton and Tritium*. The highest total dose for the maximum exposed individual was about 0.2 millirem to the total body at all three locations (refer to Table 6.6 of the PEIS). The dose estimates for all organs except bone (zero dose) were the same as that for total body. The total body population dose received by the human population within 50 miles due to these releases was estimated to be 4 person-rem.

  - *Particulates*. The offsite doses due to the particulate releases were expected to be less than those from decontamination of the containment building. The estimates from decontamination were significantly less than those due to the tritium source term of 500 curies per year; therefore, the exposures from particulate releases encompassed the cumulative dose from head removal activities.

- *Offsite Doses (Accident Conditions)*. The amount of krypton released in these accident scenarios was so small (1.5 curies per fuel element accident and 35 curies from a trapped bubble) that there would be negligible offsite doses. Other fission products that might be released would be in either a particulate or water-soluble form and would be entrained in the RCS water for subsequent water processing.

- ***Core Examination and Defueling***. The defueling and inspection operations would be conducted through the transfer canal and spent fuel pool water. The only releases expected during normal operations would be the escape of trapped fission gas bubbles, as pieces of the damaged core were pried apart and as evaporated tritiated water in the transfer canal and spent fuel pools. Soluble and particulate material released by these operations would be picked up by the transfer canal water and removed by the processing systems (refer to Section 7 of the PEIS).

- *Normal Releases (Gaseous)*. The concentration of krypton-85 in the containment building that might result from the release of 35 curies of fission gas was discussed in the accident scenario leading to mechanical damage of the core during the removal of the reactor vessel head and internals (refer to Section 6.3.4.2 of the PEIS).

- *Normal Releases (Particulates)*. Any airborne particulate activity in the effluent from the containment building during the defueling and inspection operations was expected to be much less (less than 20 percent) than during the initial decontamination of the building.

- *Normal Release (Tritium)*. Tritium releases from evaporation of the transfer canal and spent fuel storage pool water were discussed in the analyses of the removal of the reactor vessel head and internals (refer to Section 6.3 of the PEIS).

- *Releases (Accident Conditions)*. During the defueling activities, the main accident concern was related to loss of water from the transfer canal. This could occur if there was damage to the canal seal plate (e.g., with the vessel head). Failure of the seal plate would cause a loss of water in the portion of the transfer canal above the reactor vessel and in the adjacent shallow region. Water would not be lost from the deep part next to the transfer tubes, spent fuel storage pool, or reactor vessel. An accident that caused failure of the seal plate and the subsequent loss of water from the fuel transfer canal could involve the overheating of a subassembly if one were in the shallow part of the canal at the time of the accident. Although it was likely that a sizable leak in the seal plate would be immediately detected, the rate of water loss could be so rapid that a fuel transfer could not be completed before significant loss of shielding occurred. In this case, an uncooled fuel assembly could possibly be situated in the canal for an extended period of time.

  Fuel would not be present in the shallow region except during a transfer or while being placed in a container for a transfer. For the purpose of this accident analysis, a fuel assembly was assumed to be present in the shallow region when water was lost. [h] At the time of defueling, the decay heat generation rate in a single fuel assembly would be on the order of 500 watts or less; hence, not much cooling was needed. However, depending on the time the assembly remained uncovered and on the geometry, the fuel might heat up and release some fraction of the gaseous fission products (kryton-85) trapped in the fuel matrix or in the fuel-clad gap. The PEIS assumed that a maximum of 10 percent of the gaseous activity would be released in this manner. A fuel assembly could contain a maximum of 320 curies of krypton-85. The resultant concentration in the containment building would be comparable to the accident release during the removal of the reactor vessel head and the internals (refer to Section 6.3.4.2 of the PEIS).

[h] Editor's Note: When the PEIS was released in 1981, plans were to remove intact fuel assemblies in a manner similar to that used for a routine defueling outage. The results of the first Quick Look video inspection inside the reactor vessel core region changed defueling plans.

A criticality associated with the movement of fuel or control rods during the defueling could not occur as long as the boric acid concentration of the water was maintained above 3000 parts per million.

- *Offsite Doses (Normal Operations*). The offsite dose estimates for the maximum exposed individual from core examination and defueling operations were based on 35 curies of krypton-85, 20 percent of the particulate source terms from decontamination activities, and the assumption that no tritium was released. The total doses in all categories were in the range of 10^{4} to 10^{-5} millirem (refer to Table 6.8 of the PEIS). The total body population dose received by the human population within a 50-mile radius from these activities was estimated to be 2×10^{-3} person-rem.

- *Offsite Doses (Accident Conditions)*. The type of accident postulated in the analysis that would result in offsite doses included the uncovering of a fuel assembly and subsequent release of 32 curies of krypton-85. The offsite doses resulting from such a release were insignificant, and they were negligible in comparison to those from the routine decontamination operations. Consequently, the PEIS presented no numeric values for this accident scenario.

• ***Decontamination of Primary System Components***. Postaccident analyses suggested that fuel debris and perhaps other particulates were scattered throughout the primary system. Fission products were thought to have been carried from the exposed fuel and subsequently plated out on all primary system component inner surfaces. The objective of decontamination of the primary system components was to remove the fuel debris and other particulate matter from the system and to reduce the fission product plateout on internal surfaces to a level similar to that of operating reactors. [i] Decontamination of the primary system components was considered as being principally two distinct activities: flushing after draining and chemical decontamination.

- *Releases (Normal Operations)*. The decontamination operations would generate solid and liquid wastes. Section 8 of the PEIS described the solid wastes resulting from these operations. The solutions resulting from decontamination operations were of two types: relatively clear water from simple flushing operations and chemical reagents containing substantial solids. Section 7 of the PEIS described the treatment of these liquid wastes and discussed the resulting releases. The NRC expected that decontaminating reagents would be made from processed water; thus, the inventories of liquid in the system would not be increased. Although there would be fission products in the decontamination solutions, the potential airborne particulates would be negligible. However, tritium would be released by the evaporation of water from the spent fuel pool and would be less (less than 300 curies per year) than that estimated for the removal of the reactor vessel head and internals (500 curies per year).

[i] Editor's Note: When the PEIS was completed in 1981, the licensee had plans to restart TMI-2 following defueling and cleanup. The videos from the first Quick Look inspection changed these plans.

- *Releases (Accident Conditions*). To eliminate the possibility of criticality, before the boric acid concentration was reduced below 3000 parts per million, the RCS would need to be carefully inspected to ensure that any remaining fuel particle deposits would be much less than a critical mass. Another potential accident involved the spill of decontamination liquid from the reactor coolant into the containment building while the primary system pumps were operating. It was assumed that 10 percent (2000 curies) of the maximum activity content in the untreated liquids (refer to Appendix G, Table G.2 of the PEIS) was spilled before corrective action occurred. Because the solution would be at a temperature of about 200 degrees Fahrenheit, the fraction of activity that would become airborne was somewhat higher than the activity at ambient temperature. Accordingly, the analysis estimated that 0.1 percent (2 curies) of the activity was transmitted to the building HEPA filters, which released 0.1 percent (0.002 curie) to the atmosphere in the form of cesium and strontium (refer to Table 6.14 of the PEIS).

- *Offsite Doses (Normal Operations)*. The offsite dose estimates for the maximum exposed individual during the decontamination of primary system components were based on 300 curies per year of tritium. Appendix W to Volume 2 of the PEIS described the calculational models used for these estimates and the interpretation of their results.

  The total dose at all three locations was about the same at 0.1 millirem to the total body. The dose estimates for all organs except bone (zero dose) were the same as that for total body (refer to Table 6.16 of the PEIS). The total body population dose received by the human population within a 50-mile radius from these activities was estimated to be 3 person-rem.

- *Offsite Doses (Accident Condition)*. The type of accident postulated in the analysis that would result in offsite doses was a spill of decontamination liquid from the RCS into the containment building, subsequently releasing 0.002 curie of particulates. The highest total location dose was about 6 millirem to the total body in children at two locations, primarily due to vegetable consumption (refer to Table 6.17 of the PEIS).

- ***Liquid Waste Treatment***. Liquids involved in the TMI-2 decontamination would require further processing to permit their safe disposal in accordance with the NRC's proposed use of the effluent criterion in Appendix I to 10 CFR Part 50. These liquids included those directly generated during the March 28, 1979, accident (accident-generated water), as well as liquids contaminated during the cleanup operations.

- *Liquid Sources*. The sources of TMI-2 liquids and their estimated radioactivity inventories included the following: (●) auxiliary and fuel handling building chemical decontamination solutions (7000 gallons, 60 curies); (●) containment building sump water (700,000 gallons, 500,000 curies); (●) RCS water (96,000 gallons, 20,000 curies); (●) RCS flush and drain (250,000 gallons of processed water, 20,000 to 100,000 curies); (●) containment building water-based decontamination solutions (150,000 gallons of processed water, 90 curies); (●) containment building chemical decontamination solutions (40,000 gallons, 10 curies); (●) RCS water-based decontamination solutions (100,000 gallons of processed water, 2000 to 20,000 curies); and (●) RCS chemical decontamination solutions (500,000 gallons of processed water, 2000 to 20,000 curies).

- *Releases (Normal Operations)*. Table 7.11 of the PEIS showed releases attributable to airborne material (aerosol) resulting from liquid processing. The bases for the calculation of releases included the following:

  - *Pathway*. The liquid in the aerosol evaporates, leaving suspended residual solids. This airborne material was: (●) carried by subsequent movement of air in process vent systems; (●) combined with building ventilation air; (●) treated by the air cleaning system for the building; (●) monitored by radiation detectors; and (●) finally, released to the environment by controlled discharge. If necessary, releases to the environment could be stopped by shutting down the ventilation system. While in transit, larger particles settle out rapidly, and very small particles remain suspended and tend to agglomerate with each other to form larger particles. The air cleaning system included one or more stages of HEPA filters, which removed the bulk of the solid airborne material. Volatile materials, such as tritiated water vapor, were not affected by the air cleaning system.

  - *Aerosol Formation*. The gaseous effluents and releases from the liquid processing operations attributable to the formation of aerosols were dependent on the following factors: (●) concentration of the radionuclides in the liquid being treated; (●) processing rate; (●) fraction of the material processed that aerosolized; and (●) fraction of such materials that passed through the air cleaning system in an uncontrolled manner.

  - *Vapor Composition*. As a result of various processing operations involving liquid materials, small quantities of the liquids would become airborne and enter the vapor spaces above the liquid. The composition of such aerosols was identical (at the time of formation) to the liquid from which the aerosol was derived.

  - *Processing Rate*. To calculate effluents and releases, the NRC used the design-basis throughput of 10 gallons per minute. The time averaged throughput (i.e., gallons processed over several months) would be about 1/4 of this value. This rate was based on EPICOR II operating experience.

- *Aerosolization Fraction*. A conservative value of 1×10^{-4} was used as the fraction of material processed that was aerosolized and ultimately reached the air cleaning system. For the processing operations planned for TMI-2, actual values were expected to be much less than this conservative value based on previous operating experience with EPICOR II that generated less aerosols.

  - *Filter Efficiency*. The efficiency of the air cleaning system for removing solid airborne materials (aerosols) was assumed to be 99.9 percent (i.e., a penetration of aerosols through the system of 0.1 percent, or 0.001). Actual penetration values were expected to be much lower based on the theory of operation of air cleaning systems and on operating experience. The key element of the cleaning system for the removal of airborne materials was the HEPA filter. Two stages of HEPAs in series were present in the cleaning system. The principal characteristic of the HEPA filter was its great efficiency in collecting very small and very large particles. The specification for the removal efficiency of each installed HEPA filter for 0.3-micrometer particle sizes was 99.97 percent (corresponding to a penetration fraction of $3x10^{-4}$). This efficiency was demonstrated to be within specification by suitable testing. A second HEPA filter in series with the first would remove 99.97 percent of the material passing through the first HEPA. Therefore, it was evident that the assumed 99.9 percent efficiency for the system was conservative and readily achievable.

– *Calculations*. To calculate the release concentrations, expressed as picocuries per milliliter of gaseous effluent, the ventilation flow rate passing through the air cleaning system must be known. Because the locations of all processing operations considered in the PEIS were not known at the time, the appropriate ventilation rates for these calculations also were not known. Therefore, all calculations were normalized to a ventilation flow rate of 10,000 cubic feet per minute. If, for example, actual ventilation rates were doubled (i.e., 20,000 cubic feet per minute), the calculated concentrations would be 1/2 of those shown in Table 7.11 of the PEIS.

- *Releases (Accident Conditions)*. The failure of a HEPA filter was considered a credible accident event. The operation of the air cleaning system was constantly monitored, and a HEPA filter failure would be detectable in a matter of minutes. However, the analysis assumed that accident conditions would exist for a period of 15 minutes before remedial action was taken. Based on experience, when a HEPA filter failed, a fraction of its inventory of previously filtered materials would be released. A representative value of 0.001 was used for this fraction. Table 7.12 of the PEIS showed the releases calculated for the postulated accident.

- *Offsite Doses (Normal Operations)*. The dose estimates for the maximum exposed individual associated with the processing of accident and decontamination water were based on the source terms of the principal radionuclides in the various sources of wastewater (refer to Table 7.11 of the PEIS). The highest total location dose for the maximum exposed individual was 1.7 millirem to bone in children (refer to Table 7.14 of the PEIS). The 50-mile total body population dose associated with the processing was estimated to be about 2 person-rem.

- *Offsite Doses (Accident Conditions)*. The type of accident postulated in the analysis that would result in offsite doses was the rupture of a HEPA filter while processing accident or decontamination water. The estimated doses to the maximum exposed individual for each accident scenario were listed for each liquid source.

  The highest total location dose was 40 millirem to the liver in adults as the result of goat milk consumption. The liquid source for this case was from the containment building sump accident water (refer to Table 7.15 of the PEIS). The second highest case was about 30 millirem to the total body in children at two locations, primarily due to vegetable consumption. The liquid source for the second case was from the RCS flush and drain water (refer to Table 7.19 of the PEIS)

- ***Disposal of Processed Water***. Processed water was the liquid effluent arising from treatment of accident water in the auxiliary and fuel handling building, containment sump, and RCS. The PEIS presented alternatives for the disposition of processed water and associated analyses of liquid and gaseous effluents to the environment. The radionuclide effluents accounted for more than 95 percent of the calculated offsite doses.

In 1987, Supplement 2 of the PEIS, (25) “Final Supplement Dealing with Disposal of Accident-Generated Water,” was issued to update the environmental evaluation of accident-generated water disposal alternatives published in the original PEIS, using more complete and current information. Also, the supplement included a specific environmental evaluation of the licensee’s proposal for water disposition. [j]

10.3 Data Collection Activities

10.3.1 Axial Power Shaping Rod Insertion Test

- ***Purpose***. To individually move the eight axial power shaping rod assemblies in the core to provide insight into the extent of core and upper plenum damage. This early insight allowed time to factor this information into plans for subsequent inspections for the head, upper plenum, and core removal.

- ***Evaluation***: ***Radiological Release***. (26) The licensee’s safety evaluation stated that radiological aspects of this test were limited to the radiation exposure from attaching the test instrumentation to the control rod mechanisms and moving the video cameras. The actual test would be conducted from outside the containment building so no effluents would be generated from this test. The licensee’s standard radiological control procedures would be used for this test. The licensee estimated that the test would result in an exposure of less than 5.0 person-rem.

- ***NRC Review***. (27) Editor’s Note: The NRC’s safety evaluation report did not specifically address this evaluation topic.

[j] Editor’s Note: When the NRC Commissioners endorsed the PEIS in their policy statement in 1981, they wanted to study the final disposition of processed water further. Subsequently, Supplement 2 of the PEIS was issued. In 1989, the NRC issued an amendment to the facility’s operating license, modifying the technical specifications by deleting the prohibition on disposing of accident-generated water. Disposal was allowed in accordance with NRC-approved procedures.

10.3.2 Camera Insertion through Reactor Vessel Leadscrew Opening

- ***Purpose***. To insert a camera into the reactor vessel through a control rod drive mechanism leadscrew opening to provide the first visual assessment of conditions inside the reactor vessel. This activity was known as “Quick Look.”

- ***Evaluation: Radiological Release***. [28] The licensee’s safety evaluation stated that a Quick Look examination could result in an increase in radiological environmental releases because of the release of krypton-85 from the reactor coolant system (RCS). This potential release was evaluated using the following assumptions: (●) The entire RCS inventory of dissolved and free krypton-85 would be released in 1 hour into the containment building purge exhaust. (●) Krypton-85 would be diluted by a plant vent stack flow rate of 100,000 cubic feet per second. (●) No credit was taken for krypton-85 dilution in the containment.

The total quantity of krypton-85 in the RCS available for release was calculated to be about 30 curies. This estimate was based on the following: (●) Henry’s Law for dissolved gases versus free gases was assumed to apply. (●) RCS vapor space was conservatively assumed to be 190 cubic feet. (●) RCS krypton-85 activity was 0.07 microcurie per cubic centimeter based on sample results. (●) No processing of the RCS would take place.

Using the calculated release of 30 curies and the guidance in Regulatory Guide 1.109, the increased dose at the nearest residence was calculated to be 2.1×10^{-5} millirem (total body dose). Actual releases from the plant vent stack, including during times of containment building purging, were estimated to be a small fraction of the limits specified in the TMI-2 technical specifications. With the exception of RCS venting, the Quick Look inspection was not expected to result in an increase in releases of radioactivity to the environment. All releases from the containment building as a result of Quick Look would be within the limits of the technical specifications.

- ***NRC Review***: ***Radiological Release***. [29] Editor’s Note: The NRC’s safety evaluation report did not specifically address this evaluation topic.

10.3.3 Reactor Vessel Underhead Characterization (Radiation Characterization)

- ***Purpose***. To characterize radiation under the reactor vessel head to ensure that adequate radiological protection measures would be taken to keep radiation exposures ALARA during head removal. To achieve this objective, measurements were necessary with the reactor vessel head still in place.

- ***Evaluation***: ***Radiological Release***. [30] The licensee’s safety evaluation considered activities associated with the underhead radiation characterization with respect to radioactive releases to the environment. The safety evaluation concluded that these activities would not result in releases of radioactivity to the environment in excess of the releases described in the licensee’s safety evaluation report [31] for decontamination of the containment building. This conclusion was based on the following:

- *No Increase in Source Term*. The releases to the environment presented in the safety evaluation for ongoing decontamination was based on the containment building purge in continuous operation at a purge rate of 25,000 cubic feet per minute for 365 days. The source term was assumed to remain constant based on measurements of airborne activity in the containment building atmosphere during the decontamination experiment. The assumption of no reduction in the source term throughout the decontamination effort was conservative, based on experience gained during the decontamination experiment, which showed a reduction in airborne activity as decontamination activities proceeded. Since decontamination activities were ongoing during the development of the source term used for the release analysis, the licensee reasonably assumed a reduction of the source term.

- *No Increase in Airborne Activity*. The activities associated with the underhead characterization would not result in an overall increase in the airborne activity in the containment building atmosphere beyond what was assumed in the calculation of releases to the environment during decontamination activities. The activities would involve the opening of the reactor vessel to the containment building atmosphere, which could present a potential source of additional airborne activity. However, the opening was not expected to increase the airborne activity in the containment building atmosphere. The temperature of the water in the reactor vessel was expected to increase as noted in the safety evaluation of decay heat removal. However, since the reactor vessel head would be colder than the water, the reactor vessel head should remain wetted by condensation rather than drying out since the head was located above the hot water. This would tend to suppress airborne contamination. There was no planned forced mixing of the air in the reactor vessel, although a contingency system was available to draw gas out of the vessel to prevent hydrogen buildup, if needed. Since the workers would be appropriately dressed to protect against airborne activity, worker protection was ensured.

- *Conclusion*. The safety evaluation concluded that releases to the environment during the performance of the data acquisition tasks would be within the results presented in the safety evaluation report for ongoing decontamination activities. This conclusion was based on the following: (●) The source term used in the calculation of releases to the environment during continued decontamination activities was conservatively high. (●) Activities were not expected to increase the airborne activity in the containment building atmosphere beyond what was assumed in the calculation of releases to the environment during decontamination activities. (●) The purge rate would be the same as for ongoing decontamination.

  The NRC's safety evaluation report [32] for the containment building decontamination effort concluded that the rate of release of radioactivity to the environment, predicted in the safety evaluation for ongoing containment building decontamination activities, was within the TMI-2 technical specification limits. Since the releases to the environment from the underhead characterization would be within the results in the safety evaluation for ongoing decontamination activities, the licensee's safety evaluation for the underhead characterization concluded that the rate of release of radioactivity to the environment would be within the TMI-2 technical specification limits.

- ***NRC Review: Radiological Release***. (33) The NRC's safety evaluation considered the potential for an increase in airborne particulate activity. The NRC concluded that, based on partial vapor pressures and humidity, particles small enough to present a potential airborne problem would remain moist during the time that the interior of the reactor vessel was open to the containment building atmosphere. The NRC's stated position that unknown conditions under the reactor vessel head, coupled with an extended period when the reactor vessel interior surfaces would be exposed to air, warranted a contingency system. Such a system would ensure that any airborne particles generated in the reactor vessel would be segregated from the containment building ambient air. This system would protect the containment building surfaces from recontamination and would provide an additional level of protection from airborne radiation to workers and the general public. Accordingly, the NRC required that the licensee was to provide such a system before opening the reactor coolant system to the containment building atmosphere.

10.3.4 Reactor Vessel Underhead Characterization (Core Sampling) (NA)

10.3.5 Core Stratification Sample Acquisition

- ***Purpose***. To perform the activities associated with: (●) installation, operation, and removal of the core boring machine; (●) acquisition of the core samples; (●) transfer of the samples from the machine to the defueling canisters; and (●) viewing of the lower vessel region through the bored holes.

- ***Evaluation: Radiological Release***. (34) The licensee's safety evaluation stated that planned activities associated with the sampling operation were not expected to release any appreciable amounts of gaseous or particulate activity. Any potential releases of radioactivity were enveloped by the dose assessment performed for early defueling activities in the licensee's safety evaluation report (35) for early defueling of the reactor vessel. These analyses demonstrated that any potential release would be within allowable limits.

The planned core bore activity had the potential to release krypton-85 from the core debris and from the containment building. The evaluation addressed offsite radiological consequences from the potential release of krypton-85 and the measures employed to minimize exposure to workers in the containment building.

- *Quantity of Krypton-85*. Krypton-85 is a fission product with a half-life of 10.7 years, which decays primarily (99.6 percent) by beta emission at a maximum energy of 0.672 million electron volts. The amount of krypton-85 available for release from the core could not be quantified because the complete characterization of the core (the existence and extent of voids and intact fuel pins) was not available at the time of the safety evaluation. The amount of krypton-85 in the core was postulated to be as low as zero and as high as 31,300 curies. The amount of krypton-85 entrapped within voids in the core was also speculative. Therefore, the amount of krypton-85 available for release during the core bore activities could not be quantified.

- *Worst Case Scenario*. To assess offsite radiological consequences for a worst-case bounding analysis, this analysis in the licensee's safety evaluation report (36) examined heavy load handling over the reactor vessel. This analysis assumed the release of 31,300 curies of krypton-85 to the environment and showed that the radiological consequences were acceptable from an accident release perspective. The resulting dose given in the safety evaluation report was 9.7 millirem to the whole body. The corresponding dose to the skin was 810 millirem using the dose conversion factors given in Regulatory Guide 1.109. The atmospheric dispersion factor used to obtain the dose of 810 millirem was 6.1×10^{-4} second per cubic meter (from Appendix 2D to the preaccident TMI-2 final safety analysis report).

- *Controlled Release Scenario*. Accident release calculations took no credit for mitigating actions by control room operators. However, if a large release of krypton-85 from the core occurred, operator action could be taken to minimize the offsite doses. Upon indication from a portable detector or from the plant vent radiation monitor, the containment building purge could be secured. Purging the containment building would be limited to only favorable meteorology (i.e., weather that ensured a rapid dispersion) that would minimize the offsite dose due to a krypton-85 release. Based on a controlled release scenario, the expected exposure to the skin of an individual off site would be much less than 810 millirem since the atmospheric dispersion factor would be much less than the atmospheric dispersion factor used for accident assessments. Based on the TMI offsite dose calculation manual, (37) the largest atmospheric dispersion factor on an annual average basis would be 2.27×10^{-6} second per cubic meter in the southeast sector at the site boundary. Thus, the highest expected dose to the skin for an individual off site would be 3 millirem from the release of 31,300 curies of krypton-85. This dose was well within the 15-millirem limit given in 10 CFR Part 50, Appendix I.

- ***NRC Review***. (38) Editor's Note: The NRC's safety evaluation report did not specifically address this evaluation topic.

10.3.6 Use of Metal Disintegration Machining System to Cut Reactor Vessel Lower Head Wall Samples

- ***Purpose***. To remove metallurgical samples from the inner surface of the lower reactor vessel head using the metal disintegration machining (MDM) system. An international research program used these samples to study core melt and reactor vessel interactions. This program was conducted after the reactor vessel was defueled.

- ***Evaluation: Radiological Release***. [39, 40] The licensee's safety evaluation concluded that the use of the MDM equipment to cut the vessel samples did not pose a safety concern related to the release of radioactivity. The initial concern with the plasma torch operation was the high-energy (200 volts direct current, 900 amps) burning of metal and possible oxidation of fuel material to a vapor state. This was not a concern with the relatively low-energy (21 volts direct current, 100 amps) MDM cutting process.

- ***NRC Review: Radiological Release***. [41] Issues related to release of radioactivity fell within the bounds of the NRC's previous safety evaluation reports [42, 43, 44] for defueling activities.

10.4 Pre-Defueling Preparations

10.4.1 Containment Building Decontamination and Dose Reduction Activities

Purpose. To conduct decontamination and dose reduction activities in the containment building at elevation levels 305-feet (entry level) and above. Planned activities included: (●) flushing containment building surfaces with deborated water; (●) scrubbing selected components of the polar crane; (●) conducting hands-on decontamination of vertical surfaces; (●) decontaminating the air coolers; (●) flushing the elevator pit; (●) cleaning floor drains; (●) shielding the seismic gap and penetration; (●) decontaminating and shielding hot spots; (●) decontaminating missile shielding; (●) shielding reactor vessel head surface structure; (●) removing concrete and paint; (●) decontaminating cable trays; (●) decontaminating equipment; (●) remote flushing the containment building basement; and (●) testing remote decontamination technology.

- ***Evaluation: Radiological Release***. [45] The licensee's safety evaluation noted that a small fraction of the airborne radioactivity in the containment building could be transported to the environment through the purge system exhaust. Particulate radioactivity and tritium were the airborne contaminants considered in assessing the potential offsite doses due to releases from the containment building. The offsite doses that could be expected during 1986 as the result of normal decontamination operations were calculated for the safety evaluation.

  - *Source Term*. The airborne particulate concentrations of the predominant airborne contaminants in the containment building environment were determined for the period January–September 1985 (refer to page 35 of the safety evaluation report (SER)). The tritium concentration was determined by grab samples taken on the defueling work platform during defueling activities in January 1986. The maximum tritium value for the samples taken was used. The evaluation assumed that these concentrations would represent the average concentration throughout 1986. The SER stated that this was a reasonable assumption since decontamination activities could temporarily increase local concentrations of particulates. However, as decontamination tasks were completed, there would be a long-term reduction in general area airborne radioactivity.

    The purge exhaust vented air from the containment building through the plant's high-efficiency particulate air (HEPA) filters and to the environment. For calculating offsite doses, the analysis assumed that the purge exhaust flow rate was 25,000 cubic feet per minute and the purge would exhaust continuously for 1 year. The analysis further assumed that the filters removed 99.9 percent of all particulates but did not remove tritium. The total radioactivity that would be exhausted for the period of 1 year was calculated (refer to page 36 of the SER). The SER stated that the quantities of radioactive materials expected to be released to the environment were considered reasonable, yet conservative, for the following reasons: (●) Containment building purge was not normally operated continuously throughout the year. (●) Measured releases from the plant vent, including releases from the containment building and the auxiliary and fuel handling building, were less than these values for the period from 1982 through 1985. (●) Alternative decontamination techniques that would minimize the generation of airborne radioactivity or use of local ventilation control could be adopted to reduce the impact of the decontamination operations on general area airborne radioactivity levels.

The radionuclides considered in the SER (i.e., cesium-137, strontium-90, and tritium) were expected to represent the most significant airborne radioactivity sources. Gross alpha radioactivity was expected to constitute a small fraction of the total radioactivity released to the environment. Alpha radioactivity released to the environment was monitored and total radioactivity released to the environment would be maintained within the TMI-2 recovery technical specification limits.

Assuming that the calculated activities of cesium-137, strontium-90, and tritium were released, the offsite dose consequences were calculated using the Regulatory Guide 1.109 [(46)] methodology. The atmospheric dispersion factors and relative deposition per unit area values were calculated from information taken from the TMI offsite dose calculation manual. [(47)]

- *Results*. Table 5.1 of the SER provided the resulting offsite doses. This table showed that the maximum exposed individual with the highest organ dose was a child with a liver dose of 0.009 millirem per year. The maximum total body dose was 0.008 millirem per year. These offsite dose results were a small fraction of the effluent limits given in Appendix B (environmental requirements) to the TMI-2 recovery technical specifications. During decontamination operations, if the purge system was exhausting to the environment, the plant vent radiation monitors would alert operators to increases in environmental releases. The radiation monitors would alarm and automatically shift the purge system to recirculation mode at a level that would comply with technical specification limits.

- ***NRC Review: Radiological Release***. [(48)] The NRC's safety evaluation concluded that the average airborne radioactivity concentrations in the containment building were expected to be reduced by the continued decontamination of the building. During the previous decontamination and dose reduction effort, the effluent monitors did not detect any increase in particulate effluents. The NRC evaluated the offsite environmental impacts resulting from the ventilation of the containment building atmosphere through the building filtration system. Based on actual experience with containment building cleanup, the NRC expected releases and radiation doses to the public resulting from the containment building decontamination to be within the scope of the impacts assessed in the PEIS. These impacts were well within the technical specification limits.

10.4.2 Reactor Coolant System Refill (NA)

10.4.3 Reactor Vessel Head Removal Operations

10.4.3.1 Polar Crane Load Test

- ***Purpose***. To conduct load testing of the polar crane in preparation for reactor vessel head lift and removal. The polar crane load test required four major sequential activities: (1) relocation of the internals indexing fixture; (2) assembly of test load; (3) load test; and (4) disassembly of test load.

- ***Evaluation: Radiological Release (Normal Operations)***. [49] The licensee's safety evaluation considered offsite releases during load test activities. Environmental releases were prevented by the containment building pressure boundary. The containment building integrity, as required by the technical specifications, would be set during the load test. All containment penetrations that could be damaged by a load drop were isolated to outside the containment building.

Since the polar crane load test did not involve the use of any system containing radioactivity, and since containment building integrity would be set and maintained throughout the test, no release of radioactivity to the environment was expected. However, postulated occurrences identified and evaluated in the safety evaluation report could result in some slight release. Since the release pathway to the environment was through the containment building boundary, these postulated releases would be strictly controlled such that they would be bounded by the release estimate in the licensee's safety evaluation report [50] for containment building decontamination activities.

- ***Evaluation: Radiological Release (Load Drop)***. [51] The licensee's safety evaluation considered a radiological release due to a heavy load drop in accordance with the guidance from NUREG-0612, [52] "Control of Heavy Loads at Nuclear Power Plants," issued July 1980. The safety evaluation of a heavy load drop in the vicinity of the reactor core included the identification of loads, targets, and load/target interactions, as well as the evaluation of the worst credible consequence (refer to NUREG/KM Chapter 7 on load drop evaluations). The evaluation showed that NUREG-0612, Criterion I for radiological release, was met. The NUREG-0612 guidance described various alternative approaches that provided acceptable measures for the control of heavy loads. The objectives of these guidelines ensured that either the potential for a load drop was extremely small, or for each area addressed, the following evaluation criteria were satisfied:

- *Requirement (Radiological Release)*. Criterion I of NUREG-0612 stated that based on calculations involving an accidental drop of a postulated heavy load, releases of radioactive material that could result from damage to spent fuel produced doses that were well within 10 CFR Part 100, "Reactor Site Criteria," [53] limits of 300 rem thyroid and 25 rem whole body (analyses should show that doses were equal to or less than 25 percent of 10 CFR Part 100 limits).

- *Evaluation (Radiological Release)*. The impact of a missile shield block dropping onto the reactor vessel head and service structure could cause leakage of reactor coolant through the control rod drive mechanism motor tubes into the containment building. This liquid would be contained in the containment building; thus, the containment building would act as a physical barrier and prevent any liquid releases from escaping to the environment. Likewise, any gaseous releases caused by this postulated drop would be physically contained since the containment building integrity would be set and maintained throughout the load test. Containment building integrity would be further assured since there was no longer any energy source capable of producing a driving pressure that could transport this activity across the containment building boundary. Any gaseous activity released in the containment building would be directed through the high-efficiency particulate air (HEPA) filters and containment building purge exhaust system. Gaseous activity would eventually be released in a controlled manner so the release would not exceed the limits established in Criterion 1 of NUREG-0612. Further, any releases that could occur regardless of the factors presented above, would be only a small fraction of the calculated release presented for a loss-of-coolant accident in Chapter 15 of the preaccident TMI-2 final safety analysis report, thus meeting Criterion I.

• ***Evaluation: Confinement of Radioactive Material***. [54] The licensee's safety evaluation considered the impact of a load drop on the confinement of radioactive material. The mechanism for confinement of radioactive material at the time of this planned activity consisted of two major components: the physical barrier of the containment building and the lack of an energy source capable of moving radioactive material across this barrier. The reactor coolant system and especially the vessel itself also contributed to confinement but more in the sense of preventing further escape and dispersion within the containment building. No planned activity of the load test would involve breaching the physical barrier of the containment building or providing a source of energy capable of transporting radioactive material across this boundary. In addition, no postulated unplanned occurrence would yield a credible mechanism that could compromise the confinement of radioactive material within the containment building.

• ***NRC Review: Radiological Release (Normal Operations)***. [55] The NRC's safety evaluation noted that the requalification of the polar crane did not involve the use of any fluid systems that contained radioactivity. The requalification did involve the use and movement of materials and components (e.g., the polar crane, missile shields) with contamination on exposed surfaces. The NRC noted that the movement of the missile shields or other materials could somewhat increase the local airborne particulate radionuclide concentrations, relative to the ambient building concentrations, in the vicinity of the activity (the so-called "pigpen effect"), similar to the local increases detected by personnel performing other cleanup activities in the building. These increases did not result in any detectable increase in radioactive material releases to the environment, as the airborne radioactivity either resettled in the building or was swept into the building ventilation system and collected on the system filters. The licensee would operate one of the containment building ventilation filtration system trains in the recirculation mode (the other train would be operated in the purge mode) to increase the removal of particulates from the building atmosphere. Accordingly, the NRC concluded that the requalification of the polar crane would not perturb the already low levels (about 23 microcuries per year) of radioactive particulate material releases to the environment.

- ***NRC Review: Radiological Release (Load Drop)***. [56] The NRC's safety evaluation considered the entire load test sequence and the potential for an accident in relation to the required lifts and pathways selected for lift movement. This evaluation included a review of the heavy load drop analysis provided by the licensee to address the guidance in NUREG-0612. The guidance in NUREG-0612 was developed to address concerns related to dropping heavy loads in certain locations within the plant that could impact stored spent fuel, fuel in the core, equipment required to achieve safe shutdown, or equipment to remove decay heat from the core. While these were valid concerns at normal operating plants, the NRC indicated that they were of less concern at TMI-2.

There was no spent fuel stored in the refueling can, and there was no potential for impacting exposed fuel assemblies outside the reactor vessel as a result of a drop accident. However, there was the potential, with very low probability, for dropping a missile shield on the reactor vessel and service structure, thereby rearranging the physical distribution of the fuel debris in the reactor vessel. The NRC noted that virtually all of the noble gases and iodine radionuclides had already been released from the damaged fuel assemblies in the core or had decayed to insignificant levels so the potential for a large release of volatile gaseous radionuclides from a drop accident did not exist. Furthermore, any generation of airborne particulate activity would be contained inside the containment building and filtered by the building ventilation system HEPA filters before release, and any potential releases would result in doses that were well within the limits of 10 CFR Part 100.

10.4.3.2 First-Pass Stud Detensioning for Head Removal

- ***Purpose***. To perform the first-pass detensioning of the 60 reactor vessel studs and the removal of up to 5 reactor vessel studs to check for stuck nuts and to examine the condition of the removed studs.

- ***Evaluation: Radiological Release***. [57,58] The licensee's safety evaluation concluded that activities associated with first-pass detensioning and removal of up to five studs were not part of credible scenarios that could lead to valid concerns about this issue. For example, detensioning could cause a very slight movement of the plenum assembly, but this movement would be less than 0.007 inch. The evaluation did not consider it credible for a plenum movement of this magnitude to cause a gross rearrangement of the existing core configuration. Likewise, no credible mechanism was identified that could lead to releases of radioactivity as a consequence of this activity.

- ***NRC Review: Radiological Release***. [59] The NRC's safety evaluation concluded that first-pass stud detensioning posed an insignificant risk to the occupational workforce and offsite public. Additionally, the estimated environmental impacts from first-pass stud detensioning fell within the scope of those previously assessed in the PEIS.

10.4.3.3 Reactor Vessel Head Removal Operations

- ***Purpose***. To prepare for reactor vessel head removal to perform the lift and transfer of the head and conduct the activities necessary to place and maintain the reactor coolant system (RCS) in a stable configuration for the next phase of the recovery effort.

- ***Evaluation: Radiological Release (Normal Operations)***. [60] The licensee's safety evaluation considered the potential of radiological releases during this activity. The activities associated with reactor vessel head removal were reviewed with respect to radioactive releases to the containment building and the environment. The potential release of radioactivity to the environment through the airborne pathway because of these activities was considered. The tasks associated with the preparations for head removal were not significant in their potential for increasing airborne radioactivity as compared to ongoing work in the containment building. These tasks were not expected to increase the normal background airborne levels that were experienced in the containment building.

The actual lift and transfer of the reactor vessel head presented a significant deviation from ongoing work and increased the possibility of airborne contamination in the containment with particulates and tritium. Containment building integrity requirements, as documented in the recovery technical specifications, would be maintained during head lift and transfer to minimize the potential uncontrolled release of airborne radioactivity to the environment. In the case of a heavy load drop on the reactor vessel, there was the potential for krypton-85 release from the containment.

- *Particulate Releases in Containment*. To prevent a significant increase in particulate activity in the containment building, several precautions would be taken as needed based on data collected during the underhead characterization program. For example, the underside of the reactor vessel head could be bagged upon removal from the vessel, which would minimize the spread of contamination from the underside of the head. Additionally, after head removal, the open reactor vessel could be covered or wetted with a water spray to minimize the spread of contamination from the reactor vessel and the upper plenum.

  Even though particulate airborne radioactivity could increase inside the containment, the containment building integrity would be maintained throughout the head lift and transfer. Releases from the containment building through the purge exhaust were filtered by high-efficiency particulate air (HEPA) filters; therefore, no significant increase in release to the environment was expected.

- *Tritium Releases in Containment*. Opening the reactor vessel would expose air to the reactor coolant, the upper plenum, and the internal surfaces of the head. Since the tritium existed primarily as tritiated water, there was no significant deposition of tritium on the internal surfaces. However, some of the tritium in the reactor coolant would become airborne because of evaporation. The tritium concentration in the RCS was about 0.05 microcurie per milliliter. This was significantly less than the tritium concentration in the containment sump or the water being used for decontamination in the containment. Therefore, the safety evaluation concluded that opening the RCS and subsequent head lift activities would not significantly increase airborne tritium concentrations in the containment building. The safety evaluation concluded that any increase in environmental tritium releases would be unlikely.

- *Krypton-85 Releases*. During head removal activities, there was a remote possibility that any remaining krypton-85 in the reactor core could be released. A significant release of krypton-85 was postulated as the result of major core disturbance due to a head drop accident and was not a credible result of any planned activity. During head lift and transfer, the containment building integrity would be maintained to prevent an uncontrolled release of radioactivity from the containment building.

    – *Assumptions*. However, an analysis of the potential release to the environment was based on the following assumptions: (●) Krypton-85 inventory at shutdown (i.e., March 28, 1979) was 96,000 curies. (●) Known releases of krypton-85 inventory totaled 44,600 curies. This was the quantity released during the planned June–July 1980 containment building purge. All other releases were ignored for the purpose of making this calculation conservative. (●) The remaining krypton-85 was decayed to January 1, 1983. (●) Offsite doses were based on an instantaneous release of the remaining krypton-85 from the containment. (●) The accident atmospheric dispersion factor was 6.1×10^{-4} second per cubic meter (from Appendix 2D to the preaccident TMI-2 final safety analysis report).

    – *Results*. The analysis assumptions yielded a minimum release of 37,400 curies of krypton-85. Based on Regulatory Guide 1.109 methodology and activity-to-dose conversion tables, the maximum site boundary total body dose from gamma radiation was 12 millirem. The maximum site boundary skin dose from beta radiation was 980 millirem.

    – *Conclusion*. The criterion used to judge the acceptability of the resultant offsite doses was given in NUREG-0612. (61) Criterion I [k] was considered appropriate since the most likely cause for releasing the krypton-85 would be a drop of the reactor vessel head onto

[k] Criterion I in Section 5.1 of NUREG-0612 states: “Releases of radioactive material that may result from damage to spent fuel based on calculations involving the accidental dropping of a postulated heavy load produce doses that are well within 10 CFR Part 100 limits of 300 rem thyroid, 25 rem whole body (analyses should show that doses are equal to or less than 25 percent of Part 100 limits).”

the reactor vessel or plenum. Other activities associated with head removal were not expected to cause significant releases of krypton-85 since the core would not be disturbed.

Since the maximum boundary total body dose was only 12 millirem, compared to an acceptable limit of 6250 millirem, the evaluation concluded that the release of the entire remaining krypton-85 inventory would not result in unacceptable offsite doses. Results of the prior core topography indicated that few, if any, intact fuel assemblies were present in the core. The amount of krypton-85 retained in the core and available for release was expected to be significantly less than the calculated amount. Therefore, resulting offsite doses would be less than those calculated.

- ***Evaluation: Radiological Release (Load Drop)***. [62] The licensee's safety evaluation report also included an evaluation of radiological release due to a heavy load drop in accordance with the guidance from NUREG-0612. The safety evaluation of a heavy load drop in the vicinity of the reactor core included the identification of loads, targets, and load/target interactions, as well as the evaluation of the worst credible consequence (refer to NUREG/KM Chapter 7 on load drop evaluations). The evaluation showed that NUREG-0612, Criterion I for radiological release, was met. The NUREG-0612 guidance described various alternative approaches that provided acceptable measures for the control of heavy loads. The objectives of these guidelines ensured that either the potential for a load drop was extremely small, or for each area addressed, the following evaluation criteria were satisfied:

  - *Requirement (Radiological Release)*. Criterion I of NUREG-0612 stated that, based on calculations involving an accidental drop of a postulated heavy load, releases of radioactive material that could result from damage to spent fuel produced doses that were well within 10 CFR Part 100 limits of 300 rem thyroid and 25 rem whole body (analyses should show that doses were equal to or less than 25 percent of 10 CFR Part 100 limits).

  - *Evaluation (Radiological Release)*. The impact of the vessel head and service structure dropping onto the vessel could cause a release of airborne radioactivity into the containment building. An uncontrolled release from this activity to the environment would be prevented by the containment building integrity during head removal. For gaseous release (krypton-85), calculations showed that even for the worst credible release, the offsite doses would be a small fraction of the 10 CFR Part 100 limits. Particulate releases would be controlled by the containment building boundary, and if particulates were released, they would be processed through HEPA filters in such a manner so as not to exceed 10 CFR Part 100 limits. Containment building integrity was assured since there was no longer an energy source capable of producing a driving pressure that could transport the activity across the containment building boundary.

- ***NRC Review: Radiological Release (Normal Operations)***. [63] The NRC's safety evaluation considered the potential release of radioactivity to the environment due to activities associated with reactor vessel head removal. The only potential release of radioactivity to the environment was through the airborne pathway. The review indicated that activities associated with head removal would not result in significant increases in airborne radioactivity inside the containment building or in corresponding releases to the environment.

- *Evaporation*. Lifting the reactor head would expose the reactor coolant to the containment building environment. However, the reactor coolant would remain at near ambient temperatures so there would be no driving force to significantly evaporate the coolant that would cause the dispersion of the entrained radioactivity. The gross radionuclide concentration in the reactor coolant was less than 1 microcurie per milliliter, and there were no significant radioiodine isotopes or dissolved noble gases (e.g., krypton-85). Typical krypton-85 releases from the plant were less than 1 curie per day. The tritium concentration in the reactor coolant was about 0.05 microcurie per milliliter. This was significantly less than the tritium concentration in the containment building sump or the processed water used in the containment building for decontamination purposes. Therefore, evaporation of some of the tritium in the reactor coolant when the head was lifted would not likely cause any significant increases of tritium released into the containment building or to the environment. Typical tritium releases from the plant were less than 0.1 curie per day.

- *Particulates (Experimental Data)*. As part of the underhead characterization study conducted in 1983/1984, a series of air samples was taken under the reactor vessel head after the RCS was drained down to 1 foot below the plenum cover plate to simulate head lift conditions. Several samples were taken after periods of data acquisition manipulations that were expected to generate airborne activity. These samples did not show excessively high levels of airborne radioactive particulates, and the levels were in fact lower than typical levels currently found in the environment of workers conducting cleanup activities in the building.

- *Particulates (Contaminated Surfaces)*. The head removal activity would involve the movement of materials and components with contaminated surfaces (e.g., polar crane, reactor vessel head and service structure). A number of precautions would be taken to lessen the potential for generation of radioactive particulates from the contaminated surfaces. These precautions included the following: (●) installation of an after-spray system that could be used to keep the exposed plenum surfaces wet; (●) if feasible, installation of a contamination control cover for the undersurfaces of the head that could be installed before transfer of the head to the storage stand, consistent with minimizing occupational radiation exposures; (●) use of a contamination control cover to seal the undersurfaces of the head after the head was placed on the storage stand; and (●) appropriate personnel respiratory protective equipment for use by workers during head lift, which would be adequate for any reasonably expected increase in airborne particulate radioactivity.

- *Particulates (Perturbations)*. The NRC anticipated that the movement of people and contaminated materials and components could increase the local airborne particulate radionuclide concentrations relative to the ambient building concentrations, similar to the local increases generated by personnel performing other cleanup activities in the containment building (the so-called "pigpen effect"). These activities should not result in any detectable increase in radioactivity releases to the environment as the local airborne particulate radioactivity either resettled in the building or was swept into the building ventilation system and collected in the system filters. Accordingly, the NRC did not expect the head removal activities to perturb the already low levels (less than 1×10^{-7} curie per day) of radioactive particulate material released to the environment.

- ***NRC Review: Radiological Release (Load Drop)***. [(64)] The NRC's safety evaluation stated that the only perceived accident that could result in a significant release of radioactivity would be a major disturbance of the reactor core caused by a load drop on the reactor vessel head during head lift. Such an accident could cause the release to the containment building of krypton-85 that could still be trapped in the core. However, as a precautionary measure during head removal activities that involved the movement of heavy loads, specifically the reactor vessel head and internals indexing fixture assembly, the containment building would be isolated, and the purge secured. The maximum remaining inventory of krypton-85 that could still be in the reactor core was estimated to be 37,000 curies. This inventory was less than the 44,000 curies of krypton released to the environment during the planned containment building purge in June–July 1980. Therefore, even for an accident that caused the entire inventory of krypton-85 to be released into the containment, any subsequent controlled purging of krypton-85 would likely result in an environmental impact less than that of the 1980 krypton-85 purge. Based on the 1980 purge experience, the NRC estimated the maximum exposure to an individual off site as a result of a subsequent controlled purge of krypton-85 to be 3.7 millirem of beta dose to the skin and 0.04 millirem of gamma dose to the whole body.

The licensee's safety evaluation report stated that the impact of dropping the reactor vessel head and service structure onto the vessel could cause a release of gaseous radioactivity into the containment building environment. An uncontrolled release of this activity to the environment was precluded by containment building integrity during head removal. The worst case gaseous release (krypton-85) that would result in doses was a small fraction of the 10 CFR Part 100 limits. Although little airborne particulate material would be expected from a load drop, any material generated would be within the containment building boundary. Any of the airborne material that did not settle out would be processed through HEPA filters before reaching the environment. The NRC concluded that airborne particulates would have no impact on the doses resulting from postulated krypton-85 releases. Therefore, the NRC determined that the requirements of Criterion I in NUREG-0612 were met.

10.4.4 Heavy Load Handling inside Containment

- ***Purpose***. To ensure that handling of heavy loads in the containment building and spent fuel pool "A" within the fuel handling building was in accordance with the safety requirements of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," [(65)] issued July 1980.

- ***Evaluation: Radiological Release (Load Drop)***. [(66)] The licensee's safety evaluation considered a radiological release due to a heavy load drop in accordance with the guidance from NUREG-0612. [(67)] The safety evaluation of a heavy load drop in the vicinity of the reactor core included the identification of loads, targets, and load/target interactions, as well as the evaluation of the worst credible consequence (refer to NUREG/KM Chapter 7 on load drop evaluations). The evaluation showed that Criterion I for radiological release in NUREG-0612 was met. The NUREG-0612 guidance described various alternative approaches that provided acceptable measures for the control of heavy loads. The objectives of these guidelines ensured that either the potential for a load drop was extremely small, or for each area addressed, the following evaluation criteria were satisfied:

- *Requirement (Radiological Release)*. Criterion I of NUREG-0612 stated that based on calculations involving the accidental drop of a postulated heavy load, releases of radioactive material that could result from damage to spent fuel produced doses that were well within the 10 CFR Part 100 limits of 300 rem thyroid and 25 rem whole body (analyses should show that doses were equal to or less than 25 percent of 10 CFR Part 100 limits).

- *Evaluation (Radiological Release)*. Any releases of radioactivity caused by the load drops addressed in the safety evaluation report would be released within the containment building or in the fuel handling building. These buildings would act as a physical barrier and prevent any liquid releases from escaping to the environment. Likewise, any additional particulates that could become airborne would be removed by high-efficiency particulate air (HEPA) filters so as not to exceed the limits established in Criterion I of NUREG-0612. A bounding analysis was performed that assumed an instantaneous total release of the unaccounted for krypton-85 inventory from the reactor core. The amount released was assumed to be 31,300 curies of krypton-85 with the resulting dose estimated to be 9.7 millirem to the whole body for an individual located at the nearest site boundary and 1.8 millirem to the whole body for an individual located at the low-population zone boundary. The meteorological dispersion factors used were 6.1×10^{-4} second per cubic meter (sec/m^3) at the site boundary and 1.1×10^{-4} sec/m^3 at the low-population zone boundary (taken from the preaccident TMI-2 final safety analysis report).

  An additional analysis was performed in the licensee's safety evaluation report [(68)] for bulk defueling of the reactor vessel to determine the maximum offsite dose from airborne particulates that could pass through HEPA air filters following the drop of a defueling canister. This analysis used conservative assumptions and calculated a critical organ (teenager's bone) dose of 2.96 rem, which was less than 4 percent of the 75-rem acceptance criteria and 25 percent of the 10 CFR Part 100 dose guidelines. The bone dose was presented since the bone was determined to be the critical organ based on comparisons of dose conversion factors for several organs, including the lung, kidney, liver, and gastrointestinal tract, for the distribution of radionuclides available for release.

- ***NRC Review: Radiological Release (Load Drop)***. [(69)] The NRC's safety evaluation considered load drops postulated in the licensee's safety evaluation report against Criterion I of NUREG-0612.

- *Fuel Canister*. The licensee assumed that a canister drop would release the entire unaccounted for krypton-85 inventory (about 31,000 curies) of the core with 0.12 weight percent of the contents of a canister as particulate matter. The resultant offsite dose consequence was less than 4 percent of the acceptance criteria. The licensee's analysis presented a very conservative case from both a probabilistic and consequence standpoint. The drop was assumed to occur over a dry location, and the canisters would be retained by two separate, diverse mechanisms when lifted over dry areas. The drop assumed a dry powder was in the canisters, when, in actuality, they would be drained but wet with surface water.

- *Reactor Vessel*. The licensee performed a bounding analysis that assumed an instantaneous total release of the presently unaccounted for 31,300 curies of krypton-85. This gas was assumed to be trapped in the grain boundaries of fuel pellets and fuel rods. The resulting dose was 9.7 millirem to the whole body for an individual located at the nearest site boundary and 1.8 millirem to the whole body for an individual located at the low-population zone boundary. If a sump recirculation mode was necessary, any airborne particulate activity would be collected by HEPA filters in the ventilation systems for the containment building. The above doses were compared to the 6.25-rem limit of 10 CFR Part 100.

10.4.5 Heavy Load Handling over the Reactor Vessel

- ***Purpose***. To permit the handling of heavy loads (greater than 2400 pounds including rigging weight) in the vicinity of the reactor vessel. This included any load handling activity that could result in a heavy load drop onto or into the vessel either directly or indirectly (as the result of the collapse of structures or equipment installed over the vessel).

- ***Evaluation: Radiological Release***. [70] The licensee's safety evaluation considered radiological releases from normal and accident conditions.

- *Normal Conditions*. The safety evaluation identified no heavy load handling activities over the reactor vessel that would increase the release of radioactivity from the site during nonaccident conditions.

- *Accident Conditions*. The only significant source of radioactivity available for release as a consequence of a load drop accident into the vessel was the krypton-85 activity that was assumed to still be present in the reactor core and could be securely trapped in the grain boundaries of fuel pellets or in the intact fuel rods. The load drop impact could cause the release of some of the remaining krypton-85 inventory.

  A bounding analysis was performed that assumed an instantaneous total release of the unaccounted for krypton-85 inventory from the reactor core. The amount released was assumed to be 31,300 curies of krypton-85, with the resulting dose estimated to be 9.7 millirem to the whole body for an individual located at the nearest site boundary and 1.8 millirem to the whole body for an individual located at the low-population zone boundary.

The atmospheric dispersion factors used were 6.1×10^{-4} second per cubic meter (sec/m^3) at the site boundary and 1.1×10^{-4} sec/m^3 at the low-population zone boundary (based on the preaccident TMI-2 final safety analysis report).

During the containment building basement recirculation mode following the postulated load drop accident, some particulate radioactivity could become airborne from the evaporation of water on the containment building basement floor and the evaporation due to possible leakage to the auxiliary building from the decay heat removal system. Particulate activity that became airborne would be collected through the HEPA filters in the ventilation exhaust system serving the area or, in the event of filter failure, contained within the area by isolation of the ventilation systems. Consequently, this particulate activity would not contribute significantly to the offsite doses.

- ***Evaluation: Radiological Release (NUREG-0612 Evaluation)***. (71) The licensee's safety evaluation considered a radiological release due to a heavy load drop in accordance with the guidance from NUREG-0612. (72) The safety evaluation of a heavy load drop in the vicinity of the reactor core included the identification of loads, targets, and load/target interactions, as well as the evaluation of the worst credible consequence (refer to NUREG/KM Chapter 7 on load drop evaluations). The evaluation showed that Criterion I for radiological release in NUREG-0612 was met. The NUREG-0612 guidance described various alternative approaches that provided acceptable measures for the control of heavy loads. The objectives of these guidelines ensured that either the potential for a load drop was extremely small, or for each area addressed, the following evaluation criteria were satisfied:

  - *Requirement (Radiological Release)*. Criterion I in NUREG-0612 stated that, based on calculations involving the accidental drop of a postulated heavy load, releases of radioactive material that could result from damage to spent fuel produced doses that were well within the 10 CFR Part 100 limits of 300 rem thyroid and 25 rem whole body (analyses should show that doses were equal to or less than 25 percent of 10 CFR Part 100 limits).

  - *Evaluation (Radiological Release)*. Any activity releases caused by the load drops addressed in this safety evaluation report would be released within the containment. The containment building would act as a physical barrier and prevent any liquid releases from escaping to the environment. Likewise, any additional particulates that could become airborne would be removed by HEPA filters so as not to exceed the limits established in Criterion I. The analysis for krypton-85 release showed that even when using worst case assumptions (instantaneous total release with no containment), the maximum whole-body dose was 9.7 millirem compared to a limit of 6250 millirem (25 percent of 25 rem).

- ***NRC Review: Radiological Release***. (73) The NRC's safety evaluation noted that the bounding analysis performed by the licensee assumed an instantaneous total release of the presently unaccounted for 31,300 curies of krypton-85. This gas was assumed to be trapped in the grain boundaries of fuel pellets and fuel rods. The resulting dose was 9.7 millirem to the whole body for an individual located at the nearest site boundary and 1.8 millirem to the whole body for an individual located at the low-population zone boundary. If a sump recirculation mode was necessary, HEPA filters in the ventilation systems for the containment building would collect any airborne particulate activity. The above doses were compared to the 6.25-rem limit in 10 CFR Part 100.

10.4.6 Plenum Assembly Removal Preparatory Activities

- ***Purpose***. To confirm the adequacy of plenum removal equipment and techniques, the preparatory activities included: (●) video inspection of potential interference areas; (●) video inspection of the core void space; (●) video inspection of the axial power shaping rod (APSR) assemblies; (●) measurement of the loss-of-coolant accident restraint boss gaps; (●) measurement of the elevations of the APSR assemblies; (●) cleaning of the plenum and potential interference areas; (●) separation of unsupported fuel assembly end fittings; and (●) movement of the APSR assemblies.

- ***Evaluation: Radiological Release***. [74] The licensee's safety evaluation considered an inadvertent drop of tooling that dislodged fuel assembly end fittings and released particulate and gaseous radioactivity. Any potential release of particulates would be effectively scrubbed by the water in the internals indexing fixture (IIF). Even in the unlikely event of particulates becoming airborne, negligible particulate activity would be released to the environment since the containment building atmosphere was exhausted via the containment building purge system through its associated high-efficiency particulate air (HEPA) filtration system. The remaining gaseous radioactivity assumed to be within the reactor core (mostly krypton-85) could be securely trapped in the grain boundaries of fuel pellets or in the intact fuel rods. It was highly unlikely that the krypton-85 remaining in the core could be released from the dropping of tooling or the dislodging of a fuel assembly end fitting. However, the possible core disruption from the postulated collapse of the IIF platform could result in the release of the remaining krypton-85 core inventory. The analysis of releasing the krypton-85 core inventory was presented in the licensee's safety evaluation report (SER) [75] for the removal of the reactor vessel head. This analysis showed that a release of 37,400 curies of krypton-85 would result in 12 millirem to the whole body and 980 millirem to the skin at the site boundary.

- ***NRC Review: Radiological Release***. [76, 77] Editor's Note: The NRC issued two SERs; the first SER covered the first five activities (see the purpose of this section) and the second SER covered the remaining three activities. The first SER was brief and did not specifically address this topic; however, it did state that the first five activities were previously addressed in the NRC's SERs [78, 79] for Quick Look video inspection of the reactor core and for the reactor vessel underhead characterization. The NRC concluded that the licensee's prior experience with the conduct of core and plenum video inspections and other in-vessel activities (e.g., radiation measurements, reactor coolant sampling) demonstrated that these were benign activities (i.e., environmental impacts were very small), which posed little risk to the onsite workers or offsite public. The NRC further stated that the corresponding plenum inspection activities did not warrant further review.

The NRC's second safety evaluation for the remaining three activities stated that plenum removal preparatory activities would result in the movement of some fuel debris and materials within the reactor vessel. The only potential release of radioactivity to the environment during the proposed activities was from the airborne pathway, as the reactor coolant within the vessel would be in communication with the containment building atmosphere. The NRC review considered radioactive release of particulates, tritium, and noble gases, along with the mitigating measures needed.

- *Particulates*. The coolant would remain at a temperature around 97 degrees Fahrenheit so there was little possibility for the evaporation of the coolant and the dispersion of any entrained radioactivity. Additionally, the IIF processing system would be operated as necessary, to minimize the dissolved or suspended activity in the coolant, which could result from the separation of end fittings or disturbance of the core debris bed. Accordingly, the NRC did not expect plenum removal preparatory activities to perturb the already low levels (less than 1×10^{-7} curie per day) of radioactive particulate material releases to the environment.

- *Tritium and noble gas*. Tritium and noble gas releases to the containment building and the environment were not expected to deviate from currently typical releases because of the reactor's low coolant tritium concentration (0.03 microcurie per milliliter) and low dissolved noble gases coupled with the low evaporation rate. Typical tritium releases from the plant were less than 0.1 curie per day. Typical noble gas releases (krypton-35) from the plant were less than 1 curie per day.

- *Filtration*. The containment building ventilation/filtration system would be available for use if there was a significant concentration of airborne activity in the air space above the IIF during the proposed activities. This system was designed to create airflow down into the IIF through the IIF platform and then to discharge the air to the containment building atmosphere through a HEPA filter. The containment building purge system would provide additional filtration capability to further reduce the potential for a significant release of radioactivity to the environment.

- *Conclusion*. The NRC concluded that the proposed activities would not significantly increase airborne radioactivity inside the containment building or corresponding releases to the environment.

10.4.7 Plenum Assembly Removal

- ***Purpose***. To remove the plenum assembly from the reactor vessel to gain access to the core region for defueling. The plenum was moved to the deep end of the fuel transfer canal, which contained reactor coolant for shielding.

- ***Evaluation: Radiological Release***. [80] The licensee's safety evaluation stated that planned activities associated with the lift and transfer of the plenum assembly were not expected to release any appreciable amounts of gaseous or particulate activity. The remaining gaseous radioactivity assumed to be within the reactor core (mostly krypton-85) could be securely trapped in the grain boundaries of fuel pellets or in the intact fuel rods. The release of krypton-85 from the movement of the plenum assembly was considered highly unlikely. Even in the unlikely event that the particulates became airborne during plenum assembly movement, the licensee concluded that negligible particulate activity would be released to the environment because the containment building atmosphere would be exhausted via the containment building purge system through its associated high-efficiency particulate air (HEPA) filtration system.

The postulated load drops over the reactor vessel could impart a loading on the vessel that could cause the release of some of the remaining krypton-85 inventory in the core. A bounding analysis was presented in the licensee's safety evaluation report [81] for the removal of the reactor vessel head that assumed the total release of the remaining krypton-85 core inventory. The amount released was assumed to be 37,400 curies of krypton-85 with resulting doses estimated to be 12 millirem to the whole body and 980 millirem to the skin, for an individual located at the nearest site boundary. These doses were well within the acceptance criterion for radiological releases given in NUREG-0612. [82]

- ***NRC Review: Radiological Release (Load Drop)***. [83] In the licensee's previous safety evaluation report [84] for the removal of the reactor vessel head, the NRC also concurred with the licensee's bounding analysis for the potential release of krypton-85 due to fuel disruption resulting from a load drop accident. The licensee concluded that the maximum resulting offsite doses from such an unlikely event would be several orders of magnitude below the limits specified in 10 CFR Part 100.

10.4.8 Makeup and Purification Demineralizer Resin Sampling

- ***Purpose***. To obtain resin samples from the two makeup and purification demineralizers. Resin samples were required to characterize the present resin conditions for the development of a technically sound resin removal and disposal program.

- ***Evaluation: Radiological Release***. [85] The licensee's safety evaluation stated that sampling systems were designed to eliminate release pathways in the auxiliary and fuel handling building by suitable containment boxes and filtration devices. For the vacuum sampling method, the vacuum pump exhausted back to the demineralizer and did not discharge directly to the gas analyzer room. For the mechanical sampling method, the release of the resin to the sample bottle was accomplished inside a secondary containment receiver. Suitable bagging and sleeving of probes and tools ensured minimum spread of contamination.

- ***NRC Review***. Editor's Note: The NRC's safety evaluation was not located.

10.4.9 Makeup and Purification Demineralizer Cesium Elution

- ***Purpose***. To remove most of the radioactivity from the resins while they were in the demineralizers to the extent that standard resin sluice procedures could complete the task. The scope of this evaluation included only the first phase of a three-phase process for disposition of the makeup and purification of resins. This first phase included the rinse and elution of the demineralizer resins. The latter two phases would include the sluicing, removal, solidification or other packaging, and disposal of these resins. Separate safety evaluations would address the latter phases.

- ***Evaluation: Radiological Release (Normal Operations)***. [86] The licensee's safety evaluation stated that all process flow piping from the pump/eductor skid up to and including the hose would be hydrostatically tested, according to NRC Regulatory Guide 1.143, [87] "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants," and American Society of Mechanical Engineers Standard B31.1, [88] "Power Piping," to guard against line breaks and valve, pump, and flange leaks. The operating pressure of the system was not to exceed 150 pounds per square inch gauge (psig) during backflush. Therefore, the hydrostatic test would be performed at 225 psig for 30 minutes. Backflush of the stainless-steel filters would not increase the pressure in the demineralizer to more than 10 psig. (Demineralizer pressure would be maintained between 2 and 8 psig by the waste gas decay system during normal operation of the process system.) Additionally, the new process hardware would be tested to the extent practical with processed water before elution, to verify system operability and integrity. These tests were expected to identify system weaknesses so that they could be corrected to preclude a radioactive release during normal operations.

To reduce the potential for a large leak of 2700 curies per milliliter of water, the eductor was placed at 13 feet above the floor. The top of the demineralizers were 5 feet lower. Assuming a potential 8-psi overpressure in the demineralizer, a leak upstream of the eductor could not result in siphoning more than 60 gallons of water out of the demineralizer and onto the floor.

An existing radiation monitor was equipped with alarms that would sound if general area radiation levels increased. In addition, radiation control surveillance would be ongoing, as required.

- ***Evaluation: Radiological Release (Abnormal Operations)***. [89] The licensee's safety evaluation stated that properly placed and calibrated general area radiation detectors would monitor, indicate, and respond to any inadvertent releases due to valve misalignment, pipe or hose rupture, or demineralizer overflow. However, unmonitored release scenarios were contemplated without accounting for initial detection safeguards. The scenarios included demineralizer overflow and piping and valve leaks.

- *Demineralizer Overflow*. When liquid addition was attempted for the elution of the resins, a miscalculation and subsequent overaddition were postulated. Overflow from demineralizer chemical addition could result in a liquid release with an activity that might range as high as 1350 microcuries per milliliter. This overflow could potentially follow any of three pathways from each demineralizer:

  - *Scenario 1*. The liquid could simply fill the line up to and through the eductor given that the piping valves were open. The consequences of this could be safely mitigated with no actual release in the building, but higher than anticipated dose rates (less than 100 milliroentgens per hour) would be seen in the hose running to the neutralizer tanks. Radiation monitors on the hose would detect these rates so that process water could be flushed through the line to dilute the high-activity water in the neutralizer tanks. Additionally, building traffic control would be used during times of transfer to the neutralizer tanks to preclude inadvertent personnel exposure during such an event.

  - *Scenario 2*. The liquid could fill the off-gas line to the waste gas decay system. If the liquid were introduced to the waste gas header, moisture would be diverted by a pipe drain, and the moisture separators on the waste gas compressors, to the auxiliary building sump. It was unlikely that the introduction of small quantities of water to the sump would be detected, but the water in this sump was normally processed via the submerged demineralizer system and stored on site. No inadvertent release of this water would occur. Personnel exposure would be minimized since the sump was in a locked high-radiation area.

  - *Scenario 3*. The third overflow pathway was backflow through the demineralizers' normal operation inlet headers. Liquid passing through this line could be released from piping in the makeup filter cubicle. These makeup filters were removed, and their housings left open, so a release was postulated at this location. Liquid introduced to these filters would overflow the housing and spill onto the floor. There was a floor drain next to each filter housing that drained to the auxiliary building sump. It was unlikely that the introduction of small quantities of water to the auxiliary building sump would be detected, but the water in this sump was normally processed via the submerged demineralizer system and stored on site. No inadvertent release of this water would occur. Personnel exposure would be minimized since the sump was in a locked high-radiation area.

- *Piping and Valve Leaks*. Piping and valve leaks were possible in any area where process piping was located. The areas on the 305-foot elevation of the auxiliary and fuel handling building that contained elution equipment and piping included the demineralizer cubicles, gas analyzer room, and the makeup valve alley. The associated areas on the 280.5-foot elevation included the neutralizer tank cubicle and the open corridor areas in the northern half of the building. The floor drains in these areas led to the auxiliary building sump. Since this type of upset condition had the potential to divert larger quantities of process water to the sump as compared to demineralizer overflow conditions, this type of leakage was more likely to be detected in the sump. However, as water from the sump was normally processed through the submerged demineralizer system and stored on site, no release to the environment would occur.

A leak was also postulated at demineralizer isolation valves (normally left in the “bleed” position) that would introduce contaminated water to valve MU-V8 in the reactor coolant system letdown pathway. As a result, contaminated water would be directed to the “C” bleed tank, which was normally used for letdown. Liquid in the “C” bleed tank was processed through the submerged demineralizer system, so no release to the environment would occur.

- ***Evaluation: Radiological Release (Maximum Hypothetical Accident)***. [90] The licensee’s safety evaluation considered the potential onsite and offsite radiological consequences of an accident that would release the contents of the makeup and purification demineralizer. The makeup and purification demineralizers contained not only the greatest concentration of radioactive material but the greatest total amount of radioactive material of any location at TMI-2 outside of the containment building.

  - *Source Term*. The makeup and purification demineralizers were actively used at the time of the accident to finely divide the fuel material that was entrained in the reactor coolant system through these demineralizers. Therefore, any fuel materials and the associated transuranic alpha activities were contained by the demineralizer resins. This analysis assumed that this postulated accident occurred while the demineralizers contained their maximum (as of October 1984) amounts of radioactivity.

    The maximum hypothetical accident was the release of the contents of the demineralizer (22 cubic feet of resin and 200 gallons of contaminated water) to the demineralizer cubicle. The demineralizer that was chosen for analysis was the “B” demineralizer, which was estimated to contain a total of about 10,089 curies of mostly cesium-137, based on a number of analyses (refer to Attachment 1 of the safety evaluation report (SER) for a detailed listing).

    The postulated sequence of events included the following assumptions: (●) Release of the contents from the demineralizer would amount to $1.38 \times 10^{+6}$ cubic centimeters of material. (●) The floor drain inside the cubicle would be plugged by the resin material spilled from the tank. (●) The resultant spill would leave 50 percent of the resin inside the cubicle and 1 centimeter of contaminated water on the floor. (●) The remaining volume of resin (11 cubic feet) and the remaining contaminated water would spill into the area adjacent to the cubicles, which was essentially an area of 18 feet by 42 feet. (●) Water, other than a pool 1 centimeter deep in this area, would be drained to the auxiliary building sump by the floor drain in the area outside the cubicle.

  - *Release Pathway*. Releases of airborne activity to the auxiliary building during the spill would be negligible to the area outside of the cubicle. The wet resin and the dynamics of the spill would prevent all but very minor amounts of released activity from becoming airborne.

The auxiliary and fuel handling building ventilation system would remove any activity that did become airborne from the “B” demineralizer cubicle.

After the spill, the water remaining on the floor of the cubicle and adjacent area was assumed to eventually drain into the sump. Cleanup of the residual activity was assumed to take place sometime after drainage because of the high-radiation fields in the area. During the intervening time, the spilled activity associated with the degraded resin would again have to be wetted down to prevent significant airborne activity from being generated during the cleanup operations.

- *Radiological Consequences (Onsite)*. Onsite radiological consequences of the maximum hypothetical accident spill were analyzed to determine exposure rates for two conditions. The first condition examined the exposure rate at the opening of the enclosure adjacent to the cubicle due to the pool of contaminated water and resin on the floor, and the second considered the exposure rate from the auxiliary and fuel handling building exhaust duct high-efficiency particulate air (HEPA) filters due to capture of the released airborne activity.

  - *Pool Water Radiation Fields*. As stated in the assumptions, half of the resin volume and a pool of contaminated water 1 centimeter deep would remain within the demineralizer cubicle, and the remaining material would be in the area outside the cubicle. This area was basically a 42.9-feet by 18.3-feet enclosure that could be entered through a well shielded labyrinth. The area was served by two floor drains that more than likely would allow the spilled water to drain into the auxiliary building sump. For calculational purposes, this area of the floor was assumed to be left with a 1-centimeter-deep pool of water and resin. The area was modeled with a computer code for the purpose of calculating the radiation field due to this pool. Labyrinth walls were excluded from the model, which considered a rectangular area 42 feet by 18 feet that contained 4990 curies of the primary gamma-emitting isotope, cesium-137, and 217 curies of cesium-134.

    Radiation field values were calculated at the following three locations 3 feet above the pool: (●) over the center at 297 roentgens per hour; (●) at the midpoint of the side at 193 roentgens per hour; and (●) at the midpoint of the end at 163 roentgens per hour. If all the water were drained away leaving only the resin, the corresponding radiation fields would be 79 percent of those presented above.

  - *Auxiliary Building Exhaust Plenum HEPA Filter Radiation Fields*. As mentioned in the licensee’s SER [91] for the submerged demineralizer system (SDS), a dropped SDS liner was assumed to release 10^{-4} percent of the contained activity to the fuel handling building atmosphere. Because the spill considered for this analysis consisted of water-resin slurry, and because the spill was the result of an event less violent than a cask drop, only 10^{-6} percent of the activity was assumed to become airborne and entrained into the exhaust flow. Therefore, the resulting primary gamma-emitting isotopic quantities assumed to be loaded on the HEPA filter bank included 0.0902 millicurie of cesium-137 and 0.0061 millicurie of cesium-134.

In December 1981, the HEPA filter bank for the containment building purge exhaust duct was analyzed. At that time, this bank of HEPA filters was determined to contain a total of 7.4 to 9.1 millicuries of cesium-134 and 65 to 79 millicuries of cesium-137. The resulting radiation readings were 10 milliroentgens per hour on the plenum side, 25 milliroentgens per hour at the plenum top, and an average of 64 milliroentgens per hour inside the plenum at the face of the filter bank. At the time of this analysis, the ratio of cesium-137 to cesium-134 ratio equaled 8.8. When scaling the above analytical results to account for the activity released during the spill from the makeup and purification demineralizer, the 0.0902 millicurie of cesium-137 and 0.0061 millicurie of cesium-134 would add the following deferential increases to whatever was present on the filters at the time of the spill: (●) increase at the plenum top of 0.03 milliroentgen per hour; (●) increase at the plenum side of 0.01 milliroentgen per hour; and (●) increase at the filter bank face of 0.07 milliroentgen per hour. Therefore, these increases would not cause a significant increase in the exposure rate due to the HEPA filter bank.

- *Radiological Consequences (Offsite)*. Offsite radiological consequences of the maximum hypothetical accident spill were analyzed for two results at the exclusion boundary, which included the maximum permissible concentrations (MPCs) and the exposure doses for two individuals (adult and infant).

  – *Assumptions*. Assumptions for determining offsite radiological consequences that were applicable to a release included: (●) 10^{-6} percent of the demineralizer activity was released to the auxiliary building as airborne activity; (●) the release occurred over a period of 15 seconds; and (●) there was a decontamination factor of 100 for the HEPA filters in the auxiliary and fuel handling building exhausts.

    Two factors of conservatism were included. The first factor was taking credit for only one bank of HEPA filters even though the effluent actually passed through two banks of HEPA filters before being released to the astrosphere. The second factor was adopting a plume centerline value for the atmospheric dispersion factors used in the calculation. This value was higher than would be experienced at ground level at the 610-meter distance to the exclusion boundary.

  – *Results (MPC)*. Table 1 of Attachment 1 of the SER presented the offsite radioactivity release parameters. The MPC values were taken from 10 CFR Part 20. As noted in Table 1, the summation of the ratio of exclusion area concentrations and related MPCs for each radioisotope was 0.306 for this conservative scenario.

  – *Results (Inhalation Dose)*. In addition to the comparison with 10 CFR Part 20 offsite MPCs, doses were calculated for the inhalation pathway for two individuals assumed to be located at the exclusion area distance of 610 meters at the plume centerline for the full duration of the plume passage. The dose was calculated for an infant and an adult across seven radiologically significant isotopes (cesium-137, cesium-134, strontium-90, plutonium-238, plutonium-239, plutonium-241, and americium-241) to indicate the magnitude of the doses. The organs considered included bone, liver, total body, kidney, lung, and gastrointestinal tract.

Dose results were provided in two tables on page 5 (adult) and page 6 (infant) of Attachment 1 of the SER. Total dose to each organ was higher for the adult, ranging from 1.97×10^{-4} millirem (bone) to 1.5×10^{-6} millirem (gastrointestinal tract).

– *Assumptions* (*Inhalation Dose*). Additional assumptions used in the inhalation dose analysis included the following:

  - Breathing rates for the two individuals were calculated from Regulatory Guide 1.109 by dividing the annual rates by $3.15 \times 10^{+7}$ seconds per year. Total volumes of breaths during the plume passage were calculated as the respective breathing rate (cubic meters per second) times 15 seconds, the duration of the release.

  - Dose conversion factors for cesium-137, cesium-134, and strontium-90 for the respective individuals were as tabulated in the TMI offsite dose calculation manual, (92) Revision 4.

  - Doses for the plutonium (Pu) isotopes (Pu-238, Pu-239, and Pu-241) and americium-241 were calculated according to the International Commission on Radiological Protection Task Group on Lung Dynamics, as outlined in the Oak Ridge National Laboratory report, (93) "Calculation of Doses Due to Accidentally Released Plutonium from a Liquid-Metal Fast Breeder Reactor (LMFBR)."

  - Particle size for the inhaled plutonium particles was assumed to be 0.3 micron, a size that could theoretically pass through HEPA filters.

  - In all cases concerning plutonium or americium, the particle size was assumed to be 0.3 micron and the retention factor was assumed to be 30 percent.

  - All doses were time integrated to give dose commitments to 70 years of age. For the adult, assumed to be 20 years of age, the dose was a 50-year dose commitment, and for the infant, the dose was a 70-year dose commitment.

– *Conclusion (Inhalation Doses from "A" Demineralizer)*. Although the microcuries of the cesium isotopes and strontium inhaled were orders of magnitude greater than the plutonium and americium isotopes inhaled, the higher dose conversion factors for plutonium and americium created a dose roughly 10 percent of those due to cesium and strontium. The dose was not calculated for each isotope; however, the dose was calculated for the seven radiologically dominant isotopes. The highest dose was the adult bone dose of 1.97×10^{-4} millirem. Therefore, all doses calculated for the exclusion area were within acceptable limits. Doses beyond this location would be lower than the tabulated values.

As determined under the assumptions of this analysis, significant radiological concerns for such a hypothetical demineralizer spill involved the personnel doses that were incurred during cleanup. Offsite exposures for such an event were relatively insignificant. Unless a demineralizer tank rupture was determined to be a significant possibility, radiological concerns were not a driving force in the disposition of the demineralizer contents.

– *Conclusion (Inhalation Doses from "A" Demineralizer)*. Because of the higher transuranic and strontium-90 inventory of the "A" demineralizer, offsite inhalation doses were calculated assuming that the maximum hypothetical accident involved the "A" demineralizer. Similar to the calculation of the dose calculations for the "B" demineralizer, dose results for the "A" demineralizer were provided in two tables on page 6 (adult) and page 7 (infant) of Attachment 1 of the SER. Total dose to each organ was higher for the adult, ranging from 1.65×10^{-3} millirem (bone) to 4.57×10^{-6} millirem (kidney). Even though these doses were slightly higher than those potentially resulting from the "B" demineralizer, they were still within acceptable limits.

- ***NRC Review: Radiological Release***. [94] The NRC's safety evaluation stated that the most probable leakage pathway could result in releases of relatively high activity water from the system to various locations in the auxiliary building cubicles. However, the system was located in an accessible area such that radiation monitors and periodic visual inspections would quickly detect any system leaks. Suitable personnel access controls would prevent undue radiation hazards to workers in the event of leaks. System leakage would be contained within existing plant drainage sumps, precluding adverse environmental impacts. Any airborne activity that might be generated from liquid leakage or gas space venting would be controlled and processed before discharge through the normal auxiliary and fuel handling building ventilation filters and monitors.

10.5 Defueling Tools and Systems

10.5.1 Internals Indexing Fixture Water Processing System

- ***Purpose***. To provide reactor coolant water processing capability during head removal until plenum removal. The internals indexing fixture (IIF) processing system was selected based on the limited reactor coolant processing capability in the drained-down condition and the desire to provide adequate water cleanup capacity to minimize radiation dose rates around the IIF.

- ***Evaluation: Radiological Release (Failure Assessment)***. [95] The licensee's safety evaluation stated that during normal operations, the IIF processing system would function to reduce dose rates on the IIF cover in order to minimize worker exposures. To ensure that system failures would not result in unacceptable radiological conditions, the consequences of system failures were examined. Failures that could potentially result in significant changes in radiological conditions included hose leakage, hose blockage, pump failure, overflow failure, overpumping, and valve failure.

- *Hose Leakage*. The safety evaluation traced the hoses from the containment building penetration to the IIF. Leakage from the hose lines would either collect in the refueling canal or on the 347-foot elevation (operating level). To prevent possible leakage from the hoses, all discharge hoses and pipes would be leak tested in accordance with American Society of Mechanical Engineers Standard B31.1, [96] "Power Piping." The hose would also be periodically inspected visually to assess its condition. The pressure ratios of the hose and couplings were higher than the IIF processing pump shutoff head (about 150 pounds per square inch gauge pressure). An emergency shutoff switch was provided to stop the system in case of hose breakage. Although precautions were taken to prevent hose leakage, it was determined that even significant leakage of reactor coolant onto the floor or in the canal would not increase radiation dose rates enough to prevent access to the area for cleanup of the spill. Conservative analyses indicated that large puddles of reactor coolant would not result in large increases in general area dose rates around the spill. These analyses assumed cesium-137 concentrations greater than 1 microcurie per milliliter.

- *Hose Blockage*. If the IIF discharge hose were blocked, the effect would be similar to closing a downstream valve. This would result in the pump shutting down either on thermal overload or on high-level trip. In either case, no unacceptable consequence would result.

- *Pump Failure*. The pump used in the IIF processing system was a commercially available pump that had been shown to be highly reliable and was expected to operate successfully for the operational lifetime of the IIF processing system. However, in the event of failure of the IIF pump, the bubbler system would act to prevent overflow of the IIF. In addition, the pump installation was designed such that the pump could be easily removed for repair or replacement while minimizing worker exposures. In the event of pump failure, the water processing capability would be greatly reduced, and some increase in radioactivity in the coolant could result. However, since the pump was easily removable and commercially available, this reduction in water processing capability would be minimized.

- *Overflow Failure*. For the reactor coolant system level to go outside of the alarm setpoint levels, two independent instrumentation failures had to occur simultaneously in the bubbler system during the automatic level control mode. Based on the reliability of these instruments, this was extremely unlikely. Overflow of the IIF would not cause radiological conditions that would prevent access to the area for cleanup operations. The refueling canal would contain an overflow from the IIF. If the IIF overflow was indicated by level monitoring instrumentation, the video system could easily monitor the canal area from outside the containment building, so that IIF overflow could be verified and terminated. If a large overflow did occur, the fuel transfer canal drain system could be started to pump out the canal. Reactor coolant in the canal would not preclude entry into the containment building to prepare the canal drain system. Airborne radioactivity would be no worse than that experienced when high-activity water was present in the containment building basement. An emergency stop switch located in the containment building could be used to terminate overflow from the IIF.

- *Over Pumping*. The pumping of the IIF down to the pump suction point would reduce the water shield over the tops of the control rod guide tubes. The top of the pump suction was located at about 1 foot above the top of the control rod guide tubes. Therefore, there was about 12 inches of water above the guide tubes. The 12 inches of water along with the IIF cover, which consisted of 0.75 inch of steel and about 1 inch of lead, provided adequate shielding of the plenum source to permit access to the IIF platform.

- *Valve Failure*. Closure of the valves would automatically terminate the process.

- ***NRC Review***. (97) Editor's Note: The NRC's safety evaluation report did not specifically address this evaluation topic.

10.5.2 Defueling Water Cleanup

10.5.2.1 Defueling Water Cleanup System

- ***Purpose***. To remove organic carbon, radioactive ions, and particulate matter from the reactor vessel and fuel transfer canal/spent fuel pool "A" (FTC/SFP-A). The defueling water cleanup system (DWCS) actually comprised two independent subsystems (sometimes called trains): the reactor vessel cleanup system and the FTC/SFP-A cleanup system. The DWCS was totally contained within areas that had controlled ventilation and could be isolated. This limited the environmental impact of the system during normal system operations, shutdown, or postulated accident conditions.

- ***Evaluation: Radiological Release (Onsite)***. (98) The licensee's safety evaluation considered the operation and design of the DWCS with respect to radioactive releases. No direct radioactive release paths to the environment existed for the system. Local spillage of contaminated water from the DWCS would result in a local contamination problem. Since the specific activity of the water was essentially that of the FTC and SFP-A, no significant radioactive releases above those from the open pools could occur when processing pool water. Defueling activities had the potential to significantly increase the specific activity of the reactor vessel water. To preclude any significant releases during these periods, the operating procedures associated with processing reactor vessel water would include requirements to ensure isolation of the system, should there be a line break or massive system leakage.

During shutdown of the DWCS filter trains, radiolytic decomposition of the water in the post filters and filter canisters caused the production of hydrogen and oxygen. To prevent the overpressurization of the filter canisters as a result of radiolytic decomposition, a pressure relief valve, which was certified by the American Society of Mechanical Engineers (Section VIII), was installed in the outlet pipe from each of the four DWCS filter canisters. Radiolytic decomposition of water in the post filters was minimized by the limited holding capacity of the filters.

Additionally, the frequent operation of DWCS filtration trains prevented accumulation of combustible gases in the post filters.

The filter canisters would not normally be isolated for extended periods; however, if extended isolation occurred, the maximum rate of hydrogen and oxygen generation within the canister based on conservative assumptions was estimated to be 0.029 standard cubic foot (scf) per day. (Note that later analyses resulted in a maximum hydrogen generation rate that was lower by about a factor of 10). At this rate of gas generation, the pressure inside the canister would not reach the canister design pressure (150 pounds per square inch gauge pressure) for at least 90 days. The relief valve would release the pressure buildup before this pressure was exceeded with about 0.3 scf of hydrogen and oxygen released from each canister. The relief valves would continue to relieve pressure at about 15-day intervals, releasing a maximum of about 0.3 scf hydrogen and oxygen per canister per relief. The relief valves discharged to the open volume of the containment building above the FTC or to the operating level of the fuel handling building. Since both of these areas were continuously or regularly vented and since the maximum volume of hydrogen released would be small, a buildup of hydrogen to a combustible concentration was not credible. Any particulate released during the operation of the relief valves would be bounded by the line breaks discussed in the safety evaluation report (SER).

- ***Evaluation: Radiological Release (Off Site)***. (99) The licensee's safety evaluation stated that the operation of the DWCS could reduce the offsite doses, whereas if the system was not available, higher offsite doses would result. Without the operation of the DWCS, specific activity of the water in the pools would slowly increase. This could lead to an increase in the local airborne concentration available for release via the plant ventilation system. However, operation of the DWCS would maintain the reactor and fuel pool water at very low specific activity, thereby minimizing this as a potential release source. Since the source available for release from the submerged demineralizer system (SDS) greatly exceeded the source available from the DWCS, the offsite dose analysis provided in the licensee's technical evaluation report (100) for the SDS bounded those of the DWCS.

- ***Evaluation: Radiological Release (Load Drop)***. (101) The licensee's safety evaluation considered load handling inside the containment building, including the transfer of the DWCS filter canisters from the deep end of the FTC to the fuel handling building via the fuel transfer system. Load handling within the fuel handling building consisted of the movement of SDS ion exchange liners, DWCS liners, DWCS filter canisters, and transfer casks. The heavy load drop analysis for the SDS shipping casks was provided in a previous SER (102) for the control of heavy loads. The DWCS liners would be moved using the existing liner transfer casks for the EPICOR II system. The handling of heavy loads in the containment building and in the fuel handling building was addressed in a previous SER (103) for the control of heavy loads inside the containment building.

The radiological concerns associated with a load drop of the SDS ion exchange liners and the DWCS liners were bounded by the analysis in the technical evaluation report (104) for the SDS. The radiological concerns associated with a load drop of a filter canister were bounded by the accident analysis in the previous SER (105) for the bulk defueling of the reactor vessel. These analyses showed that public health and safety would not be endangered as a result of these hypothetical accidents.

- ***NRC Review: Radiological Release***. [106] The NRC's safety evaluation considered the DWCS design for potential radioactive material releases to the environment and to the containment building as a result of system leakage, line breaks, or releases of gases.

  - *Design Features*. The DWCS design provided for the following features to prevent or reduce the consequence of a release: (●) Process hoses were used in several portions of the system. Hoses were steel armored to minimize the chance of accidental damage or breakage. (●) In the event of breakage, the hose routing was such that leakage would back into the FTC/SFP-A where no net loss of water inventory would occur or go to locations where the leakage could be collected in either the auxiliary and fuel handling building sumps or the containment building sump. (●) Loss of water inventory resulting from hose or pipe breaks would be detected by redundant level monitors, which were required by the recovery technical specifications, in the reactor vessel, FTC, and SFP-A. (●) Loss of water level would activate alarms. In the case of a decreasing level in the reactor vessel, the DWCS pumps would trip on indication of low water level. (●) Siphoning of water from the reactor vessel and FTC/SFP-A as a result of a line break was prevented by use of siphon breaks in the reactor vessel suction and return lines and also by use of check valves in the FTC/SFP-A return lines.

  - *System Leak*. Spillage of water from the DWCS would result in local contamination of areas within the containment building and fuel handling building but would not result in liquid leakage to the environment. The activity levels of the water would be low enough that contaminated areas would be accessible for cleanup and repair activities without undue hazard to the workers. Airborne activity resulting from spills would be contained in areas with controlled and monitored ventilation pathways. The potential airborne activity released to the building atmosphere was within the bounds of those previously analyzed in the safety evaluations for internals indexing fixture water processing and SDS processing.

  - *Conclusion*. The activity level of the fluids to be handled during defueling water cleanup was less than the activity levels of the water initially processed by the SDS and comparable to activity levels handled during internals indexing fixture processing. The design features of DWCS contributed to the relatively small probability of significant leakage or spillage from the system, and any spills or leaks would be contained within controlled areas. The NRC concluded that there was low risk of radioactive material releases from the system and that the consequences of any release would not pose an undue hazard to the public, the environment, or plant workers.

10.5.2.2 Cross-Connect to Reactor Vessel Cleanup System (NA)

10.5.2.3 Temporary Reactor Vessel Filtration System

- ***Purpose***. To restore and maintain the visibility in the reactor vessel to acceptable levels to ensure the continuation of the early defueling operations. Operation of the defueling water cleanup system revealed that a differential pressure across its filter canisters would increase rapidly as the result of microorganism growth in the reactor coolant. Consequently, the defueling water cleanup system was able to process only a relatively small amount of reactor coolant before the maximum design pressure was reached and the filter canister had to be replaced. These developments created the need to design and operate a temporary filter system while a permanent program to control this phenomenon was being developed.

- ***Evaluation: Radiological Release***. [107] The licensee's safety evaluation considered the consequence of radiological spills inside the containment building due to filler material spill, liquid spill, and canister drop.

  - *Filter Material Spill*. Within a couple of months following the startup of the temporary reactor vessel filtration system, a knockout canister was used to dispose of the diatomaceous earth filter residue, eliminating the use of 55-gallon drums. The safety evaluation postulated a spill of diatomaceous earth during the transfer from the filter to a knockout canister. In this case, the transfer water and about 6 pounds of diatomaceous earth, along with the filtrate, were spilled onto the surface of the north end defueling work platform. If such a spill occurred, a portion of the platform would be contaminated with up to 2.1 curies of strontium/yttrium-90 and 0.1 curie of cesium-137. If the spill spread to cover a depth of 1/8 inch (3 millimeters), an area of about 500 square feet would be contaminated. Dose rates attributable to this contamination would be in the range of 1.2 rads per hour at 10 centimeters above the floor.

    A resuspension factor of 1×10^{-4} was used to estimate airborne radioactivity levels in the range of 3.1×10^{-7} microcurie per cubic centimeter. Similar consequences would be obtained at higher reactor coolant contamination levels since the majority of the dose consequences were from the spent diatomaceous earth. The spill was postulated during filter backwash that was assumed to occur when the radiation level on the surface or the shield housing was 50 millirem per hour. At these levels, the local area airborne radioactivity monitors would alarm within 2 seconds of the spill. In 5 minutes, this environment would result in 33 maximum permissible concentration-hours for the involved isotope, assuming no protection factor and assuming that the airborne activity was equally distributed in the canal.

  - *Liquid Spill*. A liquid-only spill was also considered, such as a pipe break at the pump discharge, which had the potential to spill liquid from the internals indexing fixture (IIF) onto the 322-foot elevation of the fuel transfer canal floor. This event could be detected using the IIF level monitoring system. This liquid would then drain to the sump of the canal floor on the southeast corner of the upper canal, where the spill would collect and be pumped to a staging or processing location. With the suction limited to 2 feet below the surface of the water in the IIF, this represented about 4000 gallons of reactor coolant water. Such an event was not expected to significantly increase the radiation exposure to workers on the defueling work platform.

- *Dropped Canister*. Modified versions of the temporary reactor vessel filtration system used knockout and filter canisters. The effects of a dropped canister on worker doses were addressed in the licensee's safety evaluation report (108) for early defueling.

- ***NRC Review***. (109, 110, 111) Editor's Note: The NRC's safety evaluation report did not specifically address this evaluation topic.

10.5.2.4 Filter-Aid Feed System and Use of Diatomaceous Earth as Feed Material

- ***Purpose***. To add a feed material into the filter canisters to promote the buildup of cake on the filter media, thereby significantly improving the performance of the defueling water cleanup system (DWCS) filter canisters. A filter-aid feed system that used diatomaceous earth as the feed material was installed as an ancillary system to the DWCS.

- ***Evaluation***. (112) Editor's Note: The licensee's safety evaluation report did not specifically address this topic.

- ***NRC Review: Radiological Release***. (113) The NRC's safety evaluation concluded that the equipment modification necessary for the addition of body-feed/precoat material did not present the potential for any releases of radioactive material not previously analyzed in the review of the licensee's technical evaluation report (114) for the defueling canister and the licensee's safety evaluation report (115) for the DWCS.

10.5.2.5 Use of Coagulants

- ***Purpose***. To demonstrate the use of coagulants and body-feed material to improve the performance of the defueling water cleanup system (DWCS) filter canisters in maintaining water clarity. Operating experience with the DWCS had not achieved the desired clarity in the reactor coolant system (RCS) water to support defueling operations within the reactor vessel. The DWCS filters required changeout because of high differential pressure without the expected high filter throughput. The root cause of shortened filter canister life was expected to be the presence of hydrated metallic oxides in colloidal suspension within the RCS that were plugging the filter media. The addition of the coagulant with body-feed was expected to agglomerate the colloids to filterable sizes, thus forming a filter cake on the filter media.

- ***Evaluation: Radiological Release***. (116,117) The licensee's safety evaluation considered a postulated rupture of a pipe or hose during DWCS processing that would result in spillage within the auxiliary and fuel handling building or the containment building.

- *Leak Detection*. The quantity of spillage would depend on the size of the rupture and the time required to detect, identify, and terminate the leakage. A leakage could be detected during the batch-type mode by one of the following methods: (●) reduction in the discharge pressure of either the waste transfer pump or the booster pump; (●) increasing sump level; (●) increasing airborne activity; (●) unanticipated water level changes in the internals indexing fixture (IIF); or (●) a mismatch in the hourly level checks in the reactor coolant bleed tanks.

- *Recovery Time*. Considering the hourly level checks in the reactor coolant bleed tanks, system leakage was unlikely to continue undetected for more than 1 hour. Operator actions to secure system operation and to terminate the leakage were conservatively assumed to require another hour. Therefore, with the expected operating flow rate of 30 gallons per minute, about 3600 gallons of RCS water could be released to the auxiliary and fuel handling building or the containment building from a postulated rupture during the batch-type mode of processing. For the online processing mode, the RCS was being processed as a closed-loop system, and the level changes within the IIF would indicate a system leakage. An observed reduction in discharge pressure of the DWCS pump would also indicate a system leakage.

- *Spillage*. The licensee's safety evaluation report [(118)] for the DWCS showed that a line break could cause a siphon of the RCS water in the reactor vessel. The resulting leakage would be limited to 4000 gallons because of siphon breakers installed about 2 feet below the normal operating water level in the IIF.

- *Offsite Dose (Assumptions)*. The analysis of the offsite radiological consequences from a postulated rupture inside the containment building or the auxiliary and fuel handling building used the following assumptions: (●) the quantity of spillage was 4000 gallons; (●) the airborne release fractions were 0.001 for particulates and 1.0 for tritium; (●) only one of two trains of the high-efficiency particulate air (HEPA) filters was available; (●) each HEPA filter train would have a removal efficiency of 99 percent for particulates instead of the nominal 99.9 percent, and the removal efficiency for tritium was zero; (●) the accident atmospheric dispersion factor was 6.1×10^{-4} second per cubic meter (given in Appendix 20 to the preaccident TMI-2 final safety analysis report); and (●) the maximum radionuclide concentration in the RCS was observed during core drilling operations (refer to Table 1 of the safety evaluation report).

- *Offsite Dose (Results)*. The maximum dose to the bone of an exposed offsite individual was less than 4 millirem. This estimated dose was much less than the dose limits given in 10 CFR Part 100. The radionuclide concentrations in the RCS during DWCS processing were not expected to be greater than assumed in this assessment. In addition, the conservatisms employed in this assessment would adequately bound any potential for higher radionuclide concentrations in the RCS.

- ***NRC Review***. [119] Editor's Note: The NRC's safety evaluation report did not specifically address this evaluation topic.

10.5.2.6 Filter Canister Media Modification (NA)

10.5.2.7 Addition of a Biocide to the Reactor Coolant System

- ***Purpose***. To evaluate the effects of adding hydrogen peroxide as a biocide to the reactor coolant system (RCS) to aid the removal of biological contamination. This biological growth reduced water clarity and fouled the water processing system filters.

- ***Evaluation: Radiological Release***. [120] The licensee's safety evaluation considered the effects of hydrogen peroxide (H_2O_2) as a disinfectant in the RCS and whether the oxidation capability of H_2O_2 could cause an increase in radionuclide releases. Radionuclide release rates from plant systems (the RCS and fuel transfer canal) and INEL tests of core debris grab samples from RCS grade water with and without hydrogen peroxide were reviewed. These results indicated that H_2O_2 could increase the release rates by about a factor of 2 to 10 times above presently observed accumulation rates. The factor of 2 was observed by INEL and represented a realistic engineering estimate. The factor of 10 increase was observed for the fuel transfer canal and was a worst case estimate to establish the RCS cleanup capability after adding H_2O_2. Based on these increased accumulation rates, the addition of 200 parts per million (ppm) H_2O_2 to the RCS, would slightly increase fission product activity in the RCS, but this increase should have no measurable effect on containment building airborne activities, gaseous release rates, or offsite effects.

- ***NRC Review***. [121] Editor's Note: The NRC's safety evaluation report did not specifically address this evaluation topic.

10.5.3 Defueling Canisters and Operations

10.5.3.1 Defueling Canisters: Filter, Knockout, and Fuel (NA)

10.5.3.2 Replacement of Loaded Fuel Canister Head Gaskets

- ***Purpose***. To replace head gaskets on the loaded fuel canisters in spent fuel pool "A" because of excessive leakage. The replacement of gaskets involved: (●) moving a loaded canister from the spent fuel pool "A" storage rack to the dewatering station rack with the canister handling bridge; (●) removing the canister head; and (●) transferring the head to a worktable. At the worktable: (●) the head was rotated for access to the gaskets; (●) the metallic gaskets were removed; (●) new synthetic gaskets were installed; and (●) the head was inspected before reinstallation.

- ***Evaluation: Radiological Release***. [122] The licensee's safety evaluation stated that any radioactivity releases off site resulting from the gasket replacement activities were bounded by the evaluations performed in the licensee's safety evaluation report [123] for bulk defueling.

- ***NRC Review***. [124] Editor's Note: The NRC's safety evaluation report did not specifically address this evaluation topic.

10.5.3.3 Use of Debris Containers for Removing End Fittings

- ***Purpose***. To use modified fuel canisters as "debris containers" for removing fuel assembly upper end fittings, control component spiders, or other structural material from the reactor vessel. This activity was performed to expedite access to the vacuumable fuel and debris in the core. The modified fuel canister did not have internal neutron absorbing plates, concrete filler, recombiner catalyst, dewatering capability, or a relief valve. After the debris containers were loaded, they would be closed and stored in the spent fuel pool "A" racks until final dispositioning of the containers and their contents. There were no plans to use these debris containers for shipment. Since these canisters would not have relief valves installed (a prerequisite for shipping), they could be easily identified.

- ***Evaluation: Radiological Release***. [125] The licensee's safety evaluation stated that venting the container to the fuel pool water presented a potential for contamination of the pool water. However, the large dilution afforded by the 230,000 gallons in the pool greatly reduced the significance of this potential. If all the water in a container was released to the fuel pool, the cesium-137 activity in the pool water would increase less than 2.2×10^{-4} microcurie per milliliter per container. This was based on an assumed activity of 0.5 microcurie per milliliter in the reactor coolant system coolant water entrained in the container. Although the planned activities were not within the scope of the previously approved safety evaluation report for the defueling canisters, the licensee concluded that planned activities were bounded by these evaluations.

- ***NRC Review***. [126] Editor's Note: The NRC's safety evaluation report did not specifically address this evaluation topic.

10.5.3.4 Fuel Canister Storage Racks (NA)

10.5.3.5 Canister Handling and Preparation for Shipment

- ***Purpose***. To transfer defueling canisters from spent fuel pool "A" (SFP-A) to the shipping cask for offsite shipment. Defueling canisters were transferred to locations within the containment building and fuel handling building (FHB) using a transfer shield. The transfer of canisters to the shipping cask used a different device called a "fuel transfer cask."

- ***Evaluation: Radiological Release***. [127] The licensee's safety evaluation considered radiological releases during normal operations and accident conditions.

  - *Normal Operations*. The licensee concluded that activities within the scope of this safety evaluation report (SER) would not present a credible source of radioactive liquid release to the environment. The activities within the scope of this report were not expected to increase airborne radioactivity in the fuel handling building and would have a negligible effect on offsite releases. This conclusion was based on the following considerations: (●) The contents of the canisters were completely sealed during all activities except during connection to or disconnection from the disconnect fittings. (●) During these activities, the canisters were at least partially dewatered so that catalysts in the canister were activated. This prevented the buildup of gas pressure and the relieving of internal canister pressure to the fuel handling building atmosphere. (●) Airborne radioactivity potentially created by argon gas flowing through a canister during dewatering activities was vented to the submerged demineralizer system off-gas system. (●) A decontamination spray ring was used at the fuel transfer cask loading station to reduce any loose surface contamination adhering to the canister. (●) Fuel transfer cask bottom doors, which created an enclosure, minimized the creation of airborne radioactivity from the canister during transfer to the truck bay.

    Tritium existed primarily as tritiated water. Because of evaporation and the use of a spray ring, some of the tritium in SFP-A would become airborne. However, canister handling operations would not create new sources of tritium in the water or increase tritium releases to the environment. Any tritium release was monitored and maintained within technical specification limits.

  - *Accident Conditions*. Postulated accidents had been previously evaluated for potential offsite releases. The licensee's SER [128] in support of exemption from seismic design requirements presented a wide range of potential accident scenarios and source terms, including releases from a defueling canister. The maximum offsite dose involving releases from a defueling canister dropped in the dry defueling canal and crushed by nonseismic equipment was calculated to be 12 rem. This was the dose to the bone of an adolescent due to a 2-hour unfiltered release. The offsite releases presented in the seismic design requirements SER were bounding for the activities within the scope of this evaluation.

- ***Evaluation: Radiological Release (Impact on Unit 1)***. [129] The licensee's safety evaluation stated that although the FHB crane and the truck bay were shared by the two plant units, their use by one unit or the other would be determined by operational considerations on a case-by-case basis. Because the Unit 1 and Unit 2 FHB both joined a common area in the truck bay, the activities described in this SER were evaluated for the possible radiological impact on Unit 1. The activities described in this SER did not present a credible potential for radioactive liquid release to Unit 1. Any transfer of liquid, such as by the dewatering system, was controlled and maintained within the Unit 2 FHB. Since the activities described in this SER were not expected to generate significant quantities of airborne radioactivity, no increase in airborne radioactivity in Unit 1 was expected.

- ***NRC Review: Radiological Release***. [130] The NRC's safety evaluation concluded that proposed activities would not increase airborne radioactivity in the FHB nor would they present any greater potential for spills of radioactive liquids than those previously analyzed. Since the activities were performed within the FHB with the normal ventilation system in operation, planned canister handling and preparation for shipment activities would not present the potential for any abnormal environmental releases. The analysis of a dropped fuel canister in a previous NRC safety evaluation report [131] for the defueling canister bounded the worst case handling accident postulated during the program. The previous analysis determined that the worst case offsite dose commitment from this accident would be less than 20 percent of the limits of 10 CFR Part 100.

10.5.3.6 Canister Dewatering System

- ***Purpose***. To remove and filter the water from submerged defueling canisters and to provide a transfer path to the defueling water cleanup system for processing. The dewatering system also provided the cover gas for canister shipping.

- ***Evaluation***. Editor's Note: The licensee's safety evaluation of the canister dewatering system was provided in the safety evaluation report [132, 133] for canister handling and preparation for shipment.

- ***NRC Review***. Editor's Note: The NRC's safety evaluation of the canister dewatering system was provided in the safety evaluation report [134, 135] for canister handling and preparation for shipment.

10.5.3.7 Use of Nonborated Water in Canister Loading Decontamination System (NA)

10.5.4 Testing of Core Region Defueling Techniques

- ***Purpose***. To use hydraulic heavy-duty defueling tools for limited bulk defueling operations on the hard crust layer of the damaged core.

- ***Evaluation: Radiological Release***. [136] The licensee's safety evaluation stated that the effects of the proposed activity on krypton releases was evaluated in the licensee's safety evaluation report (SER) [137] for early defueling. The effects of the proposed activity on the collapse of the hard crust region were evaluated in the licensee's SER [138] for the use of a hydraulic impact chisel to separate fused material.

- ***NRC Review: Radiological Release***. [139] The NRC's safety evaluation concluded that potential releases of radioactivity and hydrogen due to breaking of the crust were bounded by previous analyses and would be mitigated by defueling procedures and equipment, as discussed in the NRC's SER [140] for early defueling.

10.5.5 Fines/Debris Vacuum System (NA)

10.5.6 Hydraulic Shredder (NA)

10.5.7 Plasma Arc Torch

10.5.7.1 Use of Plasma Arc Torch to Cut Upper End Fittings

- ***Purpose***. Use of a plasma arc torch to cut the upper end fittings inside the reactor vessel to allow the fittings to be placed directly in fuel canisters for transfer from the reactor vessel to the fuel canal. This was the first use of the plasma arc torch inside the reactor vessel.

- ***Evaluation: Radiological Release***. [141] The central zone of the plasma arc torch reached temperatures of 20,000 to 50,000 degrees Fahrenheit (degrees F) and was completely ionized. The plasma arc torch heated and melted the metal by transfer of energy from the high-temperature, high-energy arc between the electrode and work piece. The licensee's safety evaluation expected that any fuel on these metal surfaces would be heated to the liquid or vapor state and immediately oxidize, transferring its heat to the surrounding water, then resolidify, and sink into the reactor vessel. Soluble isotopes trapped in the fuel matrix could dissolve in the water. Any increase in the isotope concentration of the water was expected to be less than the increases noted during the core bore program.

- ***NRC Review***. [142] Editor's Note: The NRC's safety evaluation report did not specifically address this evaluation topic.

10.5.7.2 Use of Plasma Arc Torch to Cut the Lower Core Support Assembly (NA)

10.5.7.3 Use of Plasma Arc Torch to Cut Baffle Plates and Core Support Shield (NA)

10.5.7.4 Use of Air as Secondary Gas for Plasma Arc Torch (NA)

10.5.8 Use of Core Bore Machine for Dismantling Lower Core Support Assembly

- ***Purpose***. To use the core bore machine, in conjunction with the automatic cutting equipment system, to dismantle the lower core support assembly and facilitate defueling by providing access to the reactor vessel lower head.

- ***Evaluation: Radiological Release***. [143] The licensee's safety evaluation stated that the safety concerns related to this operation, such as the potential releases of radioactive material, criticality within the reactor vessel, and the potential for a pyrophoric event, were previously addressed in its safety evaluation report [144] for core (bore) stratification sample acquisition.

Use of the core bore machine on the lower core support assembly did not alter the consequences of these issues. Consequently, operating procedures would incorporate the following restrictions: (●) A water level instrument would be incorporated for the internals indexing fixture/reactor vessel. (●) The internals indexing fixture/reactor vessel water level would be measured and logged every hour. (●) The borated water storage tank would be maintained at a minimum of 390,000 gallons and 4950 parts per million boron. (●) The containment building sump would be limited to a maximum of 70,000 gallons of water. (●) Weight on the drill bit would be limited to 9000 pounds.

- ***NRC Review***. [145] Editor's Note: The NRC's safety evaluation report did not specifically address this evaluation topic.

10.5.9 Sediment Transfer and Processing Operations

- ***Purpose***. To collect sediment from tanks and sumps in the auxiliary and fuel handling buildings, and also from the containment building basement and sump, in order to transfer the sediment to the spent resin storage tanks and treat or process the sediment (for disposal).

- ***Evaluation: Radiological Release***. [146] The licensee's safety evaluation stated that an inadvertent release of liquid from the sediment transfer or processing equipment could result from valve misalignment, hose rupture, or tank overflow. Any leakage during transfer of the containment building basement liquid/sediment onto the 305-foot elevation (entry level) of the containment building would simply drain back to the basement floor and would remain within the building. Personnel exposure was minimized since containment building access was controlled. No inadvertent environmental release of this water would occur. Inadvertent release scenarios in the auxiliary and fuel handling building (AFHB) were contemplated without allowance for initial detection. The scenarios and their consequences included: (●) spent resin storage tank overflow; (●) hose and valve leaks; (●) transfer and process hose rupture; (●) valve misalignment; and (●) spent resin storage tank rupture.

  - *Spent Resin Storage Tank (SRST) Overflow*. Liquid levels in the SRST were continuously monitored by automatic sonic level detectors. An overfill of the SRST was considered in the event of a postulated instrument failure or an operational error. As a result of overfill, liquid/sediment would fill the SRST and overflow through the existing 2-inch-diameter overflow line that would then flow to a floor drain and finally to the AFHB sump. Other tank exit pathways were isolated during all sediment transfer operations. Since the overflow line ended 1 inch above the funnel-shaped floor drain, a minor amount of slurry could spill onto the cubicle floor. Any increase in the AFHB sump level caused by an SRST overflow would be detected at the radwaste panel digital readout, which was monitored during sediment transfer operations. No inadvertent release to the environment would occur, and personnel exposure would be negligible since the AFHB sump was in a locked, high-radiation area, and the SRST cubicles had limited access during process operations depending on discharge dose rates.

- *Hose and Valve Leaks*. Hose and/or valve leaks were possible in areas where piping and hoses were routed. The safety evaluation report (SER) listed various cubicles that were susceptible to leakage. Leakage could also occur from hoses routed in corridors on all elevations of the AFHB. However, floor drains in each of these areas drained to the AFHB sump. Since, in this situation, small quantities of liquid could be diverted to the sump (i.e., quantities small enough to go undetected by the flow rate detectors and pressure detectors), this type of leakage would most likely be detected by an increase in sump level indication. The air monitoring system would detect an increase in airborne activity. Leakage potential was further minimized by the use of polybags or sleeves placed at each of the hose and valve connections. All hose and pipes in the system were leak tested in accordance with American National Standards Institute Standard B31.1. Inadvertent release from the AFHB to the environment would be very minor and well within the release analyzed for an SRST rupture.

- *Transfer and Process Hose Rupture*. Releases caused by a hose rupture or a break were minimized by instrumentation and control circuits in the transfer and process systems. In the event of a hose rupture in the transfer system, a pressure detector on the discharge side of the transfer pump would sense low pressure and trip the pump. If there were a similar break in the process system, the mass flow detector on the discharge side of the solids handling pump would detect a rate change and activate an alarm on the status and control panel, notifying the operator to secure process operations. A fluid detector would monitor the suction side of each of these pumps. When this monitor detected a loss of fluid, such as a hose rupture, the pump would trip. Similarly, the transfer hose from the containment penetration to the SRST was protected from a system rupture by a pressure detector on the discharge side of the containment building sediment removal pump. When this instrument detected a loss of pressure, the sediment removal pump would trip. Based on the above information, an inadvertent release from a hose rupture would result in a spill of about 50 gallons.

  The evaluation assumed that this 50-gallon spill was released in the corridor on the 280-foot elevation within the AFHB, and half of the spill would drain to the AFHB sump through floor drains, while the other half would remain on the corridor floor. The resulting dose rate exposure at the center of the spill would be 895 millirem per hour at 1 foot. The amount of airborne activity released from this type of liquid spill would be very minor. Any releases of airborne activity would be within the AFHB and removed by the high-efficiency particulate air (HEPA) filters. Offsite airborne releases would be insignificant and far within the bounds of that postulated in an SRST rupture.

- *Valve Misalignment*. The closed-loop configuration of the transfer and process system design virtually eliminated the possibility of a misaligned valve causing an inadvertent release of sediment slurries. The exceptions to this fact were in two valve locations in the processing system plant design. These valves were the 0.5-inch local sample connection valves (located in the spent resin transfer pump room) and the 2-inch resin transfer valve (located in the reclaimed boric acid pump room). To minimize an inadvertent release by these valves, their operation was administratively controlled by system operational procedures. Any release of radioactivity from a valve misalignment would occur within the containment building or the AFHB. The containment building or AFHB would act as a physical barrier and prevent any liquid release from escaping to the environment. Any airborne release and dose consequence were bounded by the SRST rupture accident.

- *SRST Rupture*. The possibility of an accident during the sediment transfer and processing operations was remote. However, environmental releases, even under accident conditions, would be controlled and filtered. The potential onsite and offsite radiological consequences were evaluated for a release of the contents from an SRST to the AFHB environment. The containment building basement and sump, along with the AFHB tanks and sump, could contain significant amounts of fission products, some fuel material, and sediment. This evaluation assumed that the accident occurred while the SRST contained the maximum estimated amounts of radioactivity (i.e., containment building sump sediment). Table 4 of the SER presented the sediment radionuclide distribution of the contents of the SRST. The predominant radionuclides were cesium-137 (3600 curies), strontium-90 (2900 curies), and cesium-134 (200 curies), which was based on 43-percent weight sediment concentrated in the SRST.

  The SRST rupture was considered to release 2400 gallons containing 43 percent by weight concentrated sediment to its respective cubicle. As an example, SRST "A" and its associated cubicle were used. The floor drain inside the cubicle could be assumed to be plugged with material spilled from the tank. The result of the spill was assumed to leave 50 percent of the sediment inside the cubicle with a layer of water 1 centimeter deep. The remaining sediment and contaminated water were assumed to spill into the area adjacent to the cubicles, which included the SRST "B" cubicle and the spent resin transfer pump cubicle. Water, other than a pool 1 centimeter deep in this area, was assumed to drain to the AFHB sump by floor drains.

  Release of airborne activity to the AFHB during a spill would be negligible in the areas outside the affected cubicle. The wet sediment combined with a water layer would prevent all but very minute amounts of the released activity to become airborne. The AFHB ventilation system would remove from the cubicle any activity that did become airborne.

- *Radiological Consequence of Tank Rupture (On Site)*. Onsite radiological consequences of the tank rupture were analyzed to determine exposure rates for two conditions: the exposure rate at the opening of the enclosure adjacent to the SRST "A" cubicle due to the pool of contaminated water and sediment on the floor, and the exposure rate from the AFHB exhaust duct HEPA filters resulting from its capture of the released airborne activity.

  – *Pooled Water Radiation Field*. This calculation was based on a worst case scenario that assumed that the SRST was filled with containment building sump sediment at 43 percent by weight. The exact sediment volume at each source location was unknown, although the optimum concentration for sediment transfer was between 1 and 5 percent by weight sediment. At this concentration, several sediment transfers would be required to achieve a 43 percent by weight sediment concentration. The SRST "A" cubicle was a

14-foot by 16.5-foot enclosure entered by way of a shielded door. The area was modeled in a shielding computer code to calculate the radiation field due to the pool of sediment and water.

Radiation field intensities were calculated at several distances above the surface of the pool and noted at: (●) 886 roentgens per hour on contact; (●) 559 roentgens per hour at 1 foot; (●) 271 roentgens per hour at 3 feet; (●) 133 roentgens per hour at 6 feet; and (●) $2.8x10^{-7}$ roentgen per hour outside the cubicle. The expected radiation levels at lower concentrations would be significantly less. This conservative assumption bounded any possible worst case dose consequences from an SRST rupture.

– *Auxiliary Building Exhaust Plenum HEPA Filter Radiation Fields*. As mentioned in the licensee's technical evaluation report for the submerged demineralizer system, a dropped liner was assumed to release $1.0x10^{-4}$ of the contained activity to the atmosphere of the fuel handling building portion of the AFHB. Similarly, this analysis assumed that $1.0x10^{-4}$ was released to the environment of the auxiliary building portion. This assumption was conservative because the spill being considered for this analysis consisted of a liquid-solids mixture, and the mechanism of the spills was less violent than a cask drop. Additionally, the airborne activity that could be released from a liquid-solids spill would be significantly less than that from a dry powder spill. The released airborne activity was expected to become entrained in the exhaust flow filter system. Therefore, the resulting primary gamma-emitting isotopic quantities of $3.6x10^{-3}$ curie of cesium-137 and $2.0x10^{-4}$ curie of cesium-134 were assumed to be loaded on HEPA filter banks.

- *Radiological Consequence of Tank Rupture (Off Site)*. The offsite radiological consequences from the postulated SRST rupture were evaluated using the following assumptions: (●) $1x10^{-4}$ of the sediment activity was released to the auxiliary building as airborne activity. (●) Auxiliary building HEPA filtration efficiency was 99 percent. (●) The SRST contained radionuclide concentrations as presented in Table 4 of the SER. (●) Atmospheric dispersion factors for the exclusion boundary and the low-population zone were $6.1x10^{-4}$ and $1.1x10^{-4}$ second per cubic meter, respectively.

  As previously mentioned, the tank contents would be wet at the time of the tank rupture. Consequently, the contents would not be expected to exhibit as great a tendency to become airborne. These conservative assumptions bounded any possible dose consequence from an SRST rupture accident. The offsite doses resulting from the postulated tank rupture were assessed using the dose conversion factors listed in NUREG-0172, "Age-Specific Radiation Dose Commitment Factors for a One-Year Chronic Intake," issued November 1977, [147] and the organ dose calculation methodology consistent with Regulatory Guide 1.109. Offsite doses for the whole body, thyroid, and bone were calculated for the exclusion boundary and low-population zone. The dose for whole body, thyroid, and bone were $5.16x10^{-4}$ rem, $4.28x10^{-11}$ rem, and $8.38x10^{-3}$ rem, respectively, at the exclusion boundary, and $9.31x10^{-5}$ rem, $7.71x10^{-12}$ rem, and $1.51x10^{-3}$ rem, respectively, at the low-population zone.

The bone dose was presented since bone was determined to be the critical organ. The critical organ determination was made based on comparison of dose conversion factors for several organs, including the lung, kidney, liver, and intestinal tract.

- ***NRC Review: Radiological Release***. [148] The NRC's safety evaluation considered inadvertent releases of liquid and airborne contamination from the sediment transfer or processing equipment that could result from valve misalignment, hose rupture, tank overflow, or tank rupture. The NRC's safety evaluation stated that the agency agreed with the licensee's conclusion that the worst possible scenario of a ruptured SRST and instantaneous release of 2400 gallons containing 43 percent by weight concentrated sediment would be well within the dose guidelines of 10 CFR Part 100. The NRC also stated that these activities fell within the scope of activities previously considered in the PEIS.

10.5.10 Pressurizer Spray Line Defueling System (NA)

10.5.11 Decontamination Using Ultrahigh Pressure Water Flush

- ***Purpose***. To use ultrahigh pressure (UHP) water flush at 20,000 to 55,000 pounds per square inch to remove surface coatings and surface contamination inside the containment building.

- ***Evaluation: Radiological Release***. [149] The licensee's safety evaluation stated that a small fraction of the airborne radioactivity in the containment building could be transported to the environment though the purge system exhaust. Particulate radioactivity and tritium were the airborne contaminants considered in the assessment of the potential offsite doses due to releases from the containment building during decontamination activities. The offsite doses that could be expected from decontamination using UHP water flush operations were previously assessed in the licensee's safety evaluation report (SER) [150] for the containment building decontamination and dose reduction activities.

A temporary increase in airborne radioactivity could result during UHP flushing operations. If the purge system exhaust was operating, the plant vent radiation monitor would alarm and alert operators to increases in environmental releases. The plant vent radiation monitor would alarm and shut down the purge exhaust at a level that complied with technical specification limits for offsite releases.

To prevent the mist produced during the operation of the UHP water flush from spreading contamination to the 347-foot level (operating level) or the fuel transfer canal, the containment building purge exhaust would be operated during the use of the UHP water flush within the D-rings. Also, the purge exhaust could be operated during decontamination operations in other areas. The purge took suction from the "B" D-ring/basement area and had a design capacity of 25,000 cubic feet per minute per train. The purge operations would be procedurally limited to one train. The purge should provide a downdraft through the D-rings to draw the mist toward the basement. Airborne mist from the UHP water flush could be drawn into the purge exhaust. This should not decrease the life of the high-efficiency particulate air (HEPA) filters as any entrained moisture would be collected on the prefilters that were part of the containment building purge system. The SER did not expect any significant quantities of entrained moisture to reach the filters because of the flow velocity and complex pathway. The filters and prefilters would be changed out as necessary based on high differential pressure or low flow.

If the purge could not maintain a downdraft through the D-rings in the area being decontaminated, local ventilation control would be provided to prevent uncontrolled airborne releases in the containment building, as required. The radiological control group would specify local ventilation controls to be implemented. Other cross-contamination protection measures would be taken in accordance with the licensee's SER [151] for the containment building decontamination and dose reduction activities.

- ***NRC Review: Radiological Release***. [152] The NRC's safety evaluation stated that a small amount of airborne radioactivity, in the form of particulates and tritium, may be introduced into the containment building atmosphere during UHP water flush. Containment building effluents to the environment would be treated by the purge filtration system before release. The NRC estimated that any increase in airborne releases as a result of UHP water flushing would be a small fraction of the technical specification limits for offsite releases. The NRC concluded that the proposed operation was within the scope of the decontamination activities addressed in the PEIS.

10.6 Evaluations for Defueling Operations

10.6.1 Preliminary Defueling

- ***Purpose***. To allow movement of debris within the reactor vessel, to allow installation of defueling equipment, and to identify core debris samples for later removal and analysis by INEL. The movement of debris within the vessel was prepared for early defueling and required some rearrangement of debris in the reactor vessel before the actual removal of fuel. These activities included the loading of small pieces of core debris into debris baskets but did not include actual loading of fuel debris into fuel canisters.

- ***Evaluation***. [153] Editor's Note: The licensee's safety evaluation report did not specifically address this topic.

- ***NRC Review: Radiological Release***. [154] The NRC's safety evaluation concluded that there was little potential for a release of radioactivity significantly above the trace amounts discharged routinely from cleanup activities. The proposed preliminary defueling activities were similar to previous activities performed in the reactor vessel, and as such, were not expected to result in a significant increase in background radiation levels in the containment building or in a significant offsite release of radioactivity. During the separation of end fittings and partial fuel assemblies from the plenum, radiation levels did not increase significantly, despite the displacement of core debris.

The NRC concluded that adequate measures were in place to ensure that the potential for significant releases of radioactivity would be acceptably low during the proposed activities. These measures included the following: (●) All gaseous release pathways from the containment building to the environment would be filtered and monitored to prevent an uncontrolled release case. (●) The licensee had the capability to completely isolate the containment building from the environment. (●) Tools and equipment used for the preliminary defueling would be flushed as they were removed from the reactor vessel water to limit the spread of contamination and to prevent any uncontrolled removal of fuel. (●) Monitoring of alpha-emitting particulates at potential release points would comply with the plant environmental technical specifications. (●) The licensee estimated [155] that potential offsite releases of krypton-85 during normal defueling operations would result in doses less than 1 percent of the limits specified in 10 CFR Part 50, Appendix I. Potential kryption-85 release points were monitored, and an alarm indicating high levels was located in the control room.

10.6.2 Early Defueling

- ***Purpose***. To remove end fittings, structural materials, and related loose debris that would not involve removal of significant amounts of fuel material, to remove intact segments of fuel rods and other pieces of core debris, and to remove loose fuel fines (particles) by vacuuming operations.

- ***Evaluation: Radiological Release (Normal Operations)***. [156] Editor's Note: Refer to the licensee's safety evaluation report [157, 158] for bulk defueling for details.

- ***Evaluation: Radiological Release (Accident Conditions)***. [159] Editor's Note: Refer to the licensee's safety evaluation report [160, 161] for bulk defueling for details.

- ***NRC Review: Radiological Release***. [162] The NRC's safety evaluation stated that the licensee's safety evaluation report for early defueling described the potential for radionuclide releases (containment building atmosphere, fuel handling building (FHB) atmosphere, and the environment) during normal and accident conditions.

  - *Normal Conditions*. Potential releases to the environment would be in the form of gaseous effluents since early defueling activities would not create pathways for liquid effluents.

    – *Release Mitigation*. The following measures would be in place to limit effluents inside the containment building and FHB: (●) All gaseous release pathways to the environment from the containment building and FHB would be filtered and monitored, and building ventilation controls would be maintained in accordance with the technical specifications. (●) During defueling activities in the reactor vessel, defueling canisters, tools, and other equipment would be flushed as they were removed from the water to prevent the spread of radioactive contamination. (●) The off-gas system would be operated to filter particulates and to disperse radioactive gases that could collect under the defueling work platform. (●) The defueling water cleanup system would be operated as needed to limit particulate and ionic activity in the reactor coolant system, "A" spent fuel pool, and fuel transfer canal water. (●) Building ventilation filter systems would prevent a significant release of particulate activity to the environment.

– *Tritium Release*. New sources of tritium would not be produced by defueling activities, but a slight increase in tritium concentrations in the containment building and in tritium releases to the environment could result from an increased evaporation rate. These slight increases would not cause a significant increase in radiation doses to workers or the public.

– *Krypton Release*. The safety evaluation report considered the offsite dose contribution from a postulated release of krypton-85 during normal defueling activities. The calculated doses were several orders of magnitude below the dose limits required by the technical specifications.

- *Accident Conditions*. The licensee also analyzed potential offsite doses for two bounding accident scenarios, including an instantaneous release of all unaccounted for krypton-85 from the reactor vessel and a canister drop accident in which the entire canister contents were spilled on the dry fuel transfer canal floor.

– *Instantaneous Release*. Conservative analysis (discussed in this safety evaluation) of an instantaneous release of all unaccounted for krypton-85 (31,300 curies) yielded offsite doses several orders of magnitude below the accident limits specified in 10 CFR Part 100. The resulting offsite doses to the whole body would be less than 1 percent of the accident dose guidelines of 10 CFR Part 100.

– *Canister Drop*. The canister drop accident also yielded offsite doses well below the guidelines of 10 CFR Part 100. In addition, the canister drop accident was extremely unlikely because when a canister was transported over the dry portion of the fuel transfer canal, the canister would be held in place by the canister handling bridge grapple and by redundant canister retention mechanisms on the bottom of the canister transfer shield. The possibility of failure of both supporting mechanisms would be remote. Also, the design of the canister transfer shield, the canister itself, and the lift height made it unlikely that a postulated drop would result in both the failure of the canister pressure boundary and the entire contents of a canister being spilled onto the canal floor. However, this postulated accident represented the worst case since other postulated canister drops would occur over water and the consequences would be less severe.

- *Conclusion*. The NRC concluded that: (●) adequate methods would be implemented to minimize the release of radioactivity during normal early defueling activities; (●) the likelihood of potential accidents would be low; and (●) the offsite radiological consequences of postulated accidents would be within the guidelines specified in 10 CFR Part 100.

10.6.3 Storage of Upper End Fittings in an Array of 55 Gallon Drums

- ***Purpose***. To use shielded 55-gallon drums to store end fittings and other structural material removed from the reactor vessel.

- ***Evaluation: Radiological Release***. [163] The licensee's safety evaluation stated that the 55-gallon drum housing the end fittings was a metal container subject to corrosion. To prevent leakage, the drum would be inserted into an NC-90 shipping container. Leakage through this container was highly unlikely since the container was constructed of fiberglass and was designed to contain contaminated water.

- ***NRC Review***. [164, 165] Editor's Note: The NRC's safety evaluation report did not specifically address this evaluation topic.

10.6.4 Defueling (Also Known as "Bulk" Defueling)

- ***Purpose***. To develop defueling tools and to conduct activities necessary to remove the remaining fuel and structural debris located in the original core volume. An additional activity was to address vacuuming of debris from the lower head of the vessel, a primary activity associated with lower head defueling.

- ***Evaluation: Radiological Release (Normal Operations)***. [166] The licensee's safety evaluation considered the radiological impact of radionuclide release from defueling activities to both the fuel handling building and containment building atmospheres, as well as to the environment.

  - *Overview*. During defueling, the containment building would be maintained in accordance with the requirements of the technical specifications. All gaseous release pathways to the environment from both the containment building and fuel handling building would be filtered and monitored, which would prevent an uncontrolled release of radioactivity to the environment. Radionuclides released to the environment would be in the form of gaseous effluents because defueling systems and activities would not introduce potential liquid effluent to release pathways. The tasks associated with the preparations for defueling were not significantly different in their potential for increasing airborne radioactivity than previous work in the containment building. Consequently, these tasks were not expected to increase the normal background airborne levels currently experienced in the containment building. Potential releases of krypton-85 and particulates would be monitored as follows:

    - *Krypton-85*. Releases of krypton-85 would be monitored, and an alarm indicating high krypton-85 levels at the release point was located in the control room. The radiological controls department would determine the need for local monitoring of krypton-85 in the containment building during defueling.

- *Particulates*. Monitoring was conducted at potential release points for alpha-emitting particulates to meet the requirements of the environmental technical specifications. Beyond this, vent samples for gross alpha emitters were analyzed weekly. Additionally, portable air samples and breathing zone air samples were routinely analyzed for alpha activity. If any of these samples began to show a significant increase in the frequency of alpha detection (i.e., in excess of levels expected for background) or if the presence of plutonium or other alpha-emitters was suspected, the level of analytical scrutiny for the alpha-emitters would be increased appropriately to address the situation.

- *Particulate Release (Normal Operations)*. The licensee's evaluation considered particulate sources and release mitigation.

  - *Particulate Sources*. Sources of particulate contamination in the water or air included the following:

    - *Defueling Tools*. All surfaces and equipment, including defueling canisters that could come in contact with fuel fines, would either remain underwater during the defueling of the reactor vessel or be decontaminated (i.e., flushed), as required for radiological control, while being removed from the water. The canisters and defueling tools were designed to be easily decontaminated where practicable. This minimized the potential for fuel being removed from the vessel in an uncontrolled manner.

    - *Canister Relief Valve*. Before transfer, the defueling canisters could be partially dewatered in the reactor vessel to ensure that the catalytic recombiner was uncovered. This action would minimize hydrogen pressure buildup leading to the opening of the canister relief valves. If the relief valves lifted, particulates could be released to the fuel transfer canal or spent fuel pool "A" water. In the unlikely event that a relief valve lifted during the time when the canister was within the canister transfer shield, the resulting release was expected to be minimal.

    - *Others*. Other defueling activities that could result in releases of particulates to the fuel transfer canal or spent fuel pool "A" included the removal of fuel canister heads to perform gasket replacement and the use of vented canisters or debris containers.

  - *Release Mitigation*. Cleanup systems that were available to remove contamination from a release during normal operations included the following:

    - *Airborne*. An off-gas system was located under the shielded defueling work platform to remove particulates that could become airborne from the reactor coolant system during defueling. These radioactive particulates would be filtered out of the gaseous effluents and would not be available for release in the environment under normal operational conditions.

- *Waterborne*. If there was a release of particulates, they would either settle on the bottom of the pools, or they would be entrained in the water, which would be monitored for this type of contamination. The defueling water cleanup system, or an alternative water cleanup system, would be used as necessary to keep contamination to acceptable levels.

- *Tritium Release (Normal Operations)*. Tritium existed primarily as tritiated water. Because of evaporation, some of the tritium in the reactor coolant would become airborne. Although defueling would not create new sources of tritium in the water, the operation of the off-gas system and the additional heat introduced by the underwater lighting or other equipment, could increase the evaporation rate of the reactor coolant water. Thus, a slight increase in the tritium release rate to the containment building atmosphere was possible. However, the off-gas system would dilute the tritium as the gas was released to the containment building. Therefore, tritium concentrations would not reach unacceptable levels in the containment building, nor would tritium releases from the containment building have any unacceptable effect on the health and safety of the public.

- *Krypton-85 Release (Normal Operations)*. Krypton-85 is an inert gas and would not be removed by the containment building or fuel handing building filter systems. There was a possibility that the krypton-85 that remained in the reactor core could be released as a result of defueling activities. The offsite doses from postulated krypton-85 released to the environment were analyzed for normal operations.

  – *Assumptions*. This analysis included all defueling activities (i.e., canister filling, canister transfer, canister relief valve opening in spent fuel pool or fuel transfer canal, and dewatering). The range of values for krypton-85 readily available for release was estimated to be 0 to 100 curies; the most likely value [(167)] was about 30 curies. The safety evaluation report assumed that 100 curies of krypton-85 were available for release during defueling activities. The maximum annual average atmospheric dispersion factor from the TMI offsite dose calculation manual [(168)] was 2.27×10^{-6} sec/cm^3 and occurred in the southeast sector at the site boundary. The methodology and dose conversion factors from Regulatory Guide 1.109 were used in the calculation.

  – *Results*. The maximum offsite total body dose was 1×10^{-4} millirem per year, skin dose was 0.01 millirem per year, gamma air dose was 1×10^{-4} millirad per year, and beta air dose was 0.014 millirad per year. These doses were less than 1 percent of the dose limits of 10 CFR Part 50, Appendix I.

- ***Evaluation: Radiological Release (Accident Conditions)***. [(169)] The licensee's safety evaluation concluded that the possibility of an accident during the defueling activities was remote. However, environmental releases, even under accident conditions, would be controlled and filtered. The offsite dose consequences from two postulated scenarios were evaluated. These scenarios represented the worst credible accidents. Therefore, their offsite dose consequences would be the most severe, and all other postulated accidents were expected to result in offsite doses that were lower than those presented. The two postulated accidents included an instantaneous release of all unaccounted for krypton-85 and a canister drop accident onto a dry fuel transfer canal floor.

- *Krypton-85 Release (Accident Condition)*. An analysis of the offsite doses from a postulated release of krypton-85 to the environment was made for the worst case scenario that involved the instantaneous release of all unaccounted for krypton-85 in the reactor core.

  - *Assumptions*. This accident assumed an instantaneous release of 31,300 curies of krypton-85, which represented the unaccounted for krypton-85 inventory remaining after the controlled containment building purge of June–July 1980 and decayed to July 1, 1985. The accident atmospheric dispersion factors were 6.1×10^{-4} second per cubic meter (sec/m^3) for the site boundary (from Appendix 2D to the preaccident TMI-2 final safety analysis report) and 1.1×10^{-4} sec/m^3 for the low-population zone at 3218 meters (from Chapter 15.1.21 of the preaccident final safety analysis report). The methodology and dose conversion factors from Regulatory Guide 1.109 were used in the calculation.

  - *Results*. The maximum offsite whole-body dose at the site boundary was 9.7 millirem. The low-population zone whole-body dose was 1.8 millirem. These doses were less than 1 percent of the dose guidelines for accidents in 10 CFR Part 100.

- *Canister Drop (Accident Condition)*. When a canister was being raised into or lowered from the canister transfer shield (CTS), a failure of the grapple could result in dropping the canister. This could occur over the reactor vessel, the deep end of the fuel transfer canal (FTC), or spent fuel pool "A." In all cases, the canister would be dropped into water having a boron concentration of 4350 parts per million or greater. Therefore, subcriticality would be ensured under any leakage condition. Should the canister leak, any particulate activity would remain in the water and would not be released to the environment. Any krypton-85 that could be released from the canister would result in offsite doses less than the doses described in the above krypton-85 release evaluation.

  - *Canister Design Features*. The CTS was designed with diverse means for preventing a canister drop accident while the canister was being transported from the reactor vessel to the deep end of the FTC. Since multiple failures would be required for a canister drop accident to occur over the dry portion of the FTC, such an event was considered extremely unlikely. However, if multiple failures did occur, and a canister was dropped onto the dry portion of the refueling canal, there would be the potential for canister leakage. Potential canister leakage would be limited by the following features:

    - *Canister-CTS Clearance*. Limited space was available for leakage of canister contents due to the small inner diameter of the CTS. The maximum annular space width was estimated at 0.5 inch. The small clearance between the canister and the shield would provide structural support along the length of the canister and prevent a total circumferential rupture of a canister. Therefore, leakage would be expected to occur only at the extreme ends of the canister.

- *Bottom Head Integrity*. Vertical drop tests [170] showed that the bottom head of the defueling canisters could withstand a drop from heights exceeding the drop heights for canisters in the containment building with only minor deformation and no observed cracking. This impact load exceeded the calculated impact load for a canister drop in the containment building. Therefore, the bottom head of the canister would not be expected to crack or rupture.

- *Lift Height*. By design, the lift height of the load was restricted so that the canister would not fall completely out of the transfer shield in the event of a canister drop over the dry portion of the FTC. This ensured that any impact would occur on the canister bottom head.

- *Leak Path Clearances*. The most likely leakage path was the top portion of each canister. Under normal conditions, the canister vent and drain connections on the upper head could offer a leakage path from the canister during connect/disconnect operations. These quick disconnects included integral shutoff valves and were capped before shipping. If leakage should occur, the release was expected to consist of fuel fines, gases, and water vapor. The clearances in the fittings and the connecting tubes would not pass large fuel particles, such as fuel pellets.

- *Top Head Protection*. The upper closure head nozzles on the canisters were protected by a steel skirt. Under postulated drop accidents, direct impact loads on the canister would be minimized. There was no defined mechanism for dropping something inside the skirt that would directly impact the nozzles. Therefore, leakage from the canister due to a direct impact on the nozzles was determined to be not credible.

– *Release Amount*. The design features of the canister and the handling equipment had the potential for a very small leak. It was expected, under design drop conditions, that no leakage would occur, but for the purposes of this safety evaluation, leakage from the canister was assumed. Since the amount of leakage could not be quantified, the analysis conservatively assumed that the entire contents of the canister would leak onto the dry FTC floor.

– *Offsite Consequence (Canister Drop)*. To assess the offsite exposure consequences of the postulated canister drop, the evaluation estimated the fraction of the canister contents that became airborne into the containment building atmosphere and released to the environment. To evaluate this fraction, a literature review was conducted of experimental and calculated suspension factors. Only suspended particles were assumed to be available for offsite release.

- *Release Quantity (Canister Drop)*. NUREG/CR-2139, [171] "Aerosols Generated by Free Fall Spills of Powders and Solutions in Static Air," issued December 1981, reported on experiments that determined the weight percent of a dry powder that would become airborne after a spill. The powders used in the experiments were dry titanium dioxide and depleted uranium dioxide. These powders were released in a free fall spill through static air. Particle sizes ranged up to 75 microns, with about 98 percent of the powders having particle sizes 20 microns or less. The results of these experiments suggested that 0.12 weight percent of the particles would become airborne during a spill. The study also showed that particles less than 10 microns in diameter accounted for about 40 percent of the airborne mass. This suggested that larger particles had less of a tendency to become airborne than smaller ones. Therefore, although these tests did not cover the entire range of particle sizes of interest, they did cover the lower end of the range where particles had the greatest tendency to become airborne.

  Additional data from a 1976 report [172] confirmed the 0.12 weight percent estimate referenced in NUREG/CR-2139. Exposed to a 20-mile-per-hour (mph) wind, 1-micron particles on a stainless-steel surface resulted in 0.29 weight percent of the total mass becoming airborne. The airflow inside the containment building and fuel handling building was much less than 20 mph, so a more applicable experiment could be one with 1-micron particles on a stainless-steel surface in a 2.5-mph wind. For this case, 0.075 weight percent of the total mass became airborne. Therefore, following a spill of dry powder (i.e., particles less than or equal to 75 microns), a reasonable estimate of the percentage of the powder becoming airborne was 0.12 weight percent.

- *Release Source (Canister Drop)*. The filter canister would be expected to contain mostly fuel fines with sizes in the range of 0.5 to 140 microns, which would exhibit a tendency to become airborne and result in a 0.12 weight percent airborne fraction. The fuel canister was expected to contain large pieces of core debris, and the knockout canister was expected to contain debris ranging in size from 140 microns up to particles larger than whole fuel pellets. The fines in the filter canisters were not free but were contained within the filter media and would not become as readily airborne as dry powder. Additionally, when the canister was over the dry portion of the FTC, the canister contents were wet so when compared to dry powders, they would not be expected to exhibit as great a tendency to become airborne.

- *Assumptions (Canister Drop)*. The 0.12 weight percent airborne release fraction for dry powders (i.e., particles less than or equal to 75 microns) discussed above was used to assess the offsite exposure consequences resulting from the postulated canister drop. This conservative assumption bounded any possible dose consequences from a canister drop accident.

  The offsite doses resulting from the postulated canister drop in the dry portion of the refueling canal were based on the following assumptions and methods:

(●) 100 percent of the canister content released; (●) 0.12 weight percent of the particulates released; (●) 100 percent of the airborne gases released; (●) maintenance of the containment building integrity; (●) 99-percent efficiency of the high-efficiency particulate air (HEPA) filter; (●) 100-percent dose conversion factors from NUREG-0172 [(173)] ; (●) organ dose calculation methodology consistent with Regulatory Guide 1.109; (●) whole-body dose calculation methodology consistent with Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors" [(l)] ; (●) and radionuclide inventory of the canister based on 1 percent of the core inventory as given in GEND-INF-019, [(174)] "Estimated Source Terms for Radionuclide and Suspended Particulates During TMI-2 Defueling Operations Phase II," decayed to July 1, 1985, and applying a peaking factor of 1.9.

- *Results (Canister Drop)*. The offsite doses for the total body, thyroid, and bone were estimated to be $4.3\text{x}10^{-4}$ rem, $1.9\text{x}10^{-3}$ rem, and 2.96 rem, respectively, at the exclusion boundary, and $7.7\text{x}10^{-5}$ rem, $3.5\text{x}10^{-4}$ rem, and 0.53 rem, respectively, at the low-population area. The bone dose was presented since bone was determined to be the critical organ. The critical organ determination was made based on comparison of dose conversion factors for several organs, including the lung, kidney, liver, and gastrointestinal tract, for the distribution of radionuclides available for release.

  This accident scenario (i.e., entire canister contents spilled on dry surface) represented a worst case accident. For other canister accidents (e.g., stuck-open relief valve) the amount of the canister content released would be smaller, and thus the corresponding offsite doses would be smaller.

- ***NRC Review: Radiological Release***. [(175)] The NRC's safety evaluation noted that the proposed core region defueling activities would involve the displacement of core debris to a greater extent than earlier defueling activities. Based on previous defueling experiences, the NRC did not expect significant increases in the containment building radiation levels or in offsite releases. This determination was based on the following: (●) Systems equipment and procedures used to minimize the potential for, and consequences of, a release of radiation during early defueling activities would continue to be used during bulk defueling. (●) All gaseous release pathways to the environment would be monitored and filtered, and all containment building exhaust points could be isolated as needed. Monitoring for krypton-85 and alpha-emitting particulates would be conducted. (●) The off-gas system would be operated, as necessary, to filter particulates and disperse gases that could collect under the defueling work

[l] Editor's Note: Regulatory Guide (RG) 1.4 was withdrawn December 2016. The guidance contained in RG 1.4 was updated and incorporated into RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," and RG 1.195, "Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors." RG 1.183 provided guidance for new and existing light-water reactor plants that have adopted the alternative source term, and RG 1.195 provided guidance for those that have not adopted the alternative source term.

platform. (●) All equipment and tools would be flushed upon removal from the reactor vessel to limit the spread of contamination. (●) A water cleanup system would be operated to maintain the reactor coolant system at an acceptable activity level. (●) Analysis approved in the NRC safety evaluation report [176] for early defueling for potential releases of radiation during normal and accident conditions were also bounding for any release scenario associated with bulk defueling.

The NRC concluded that potential releases of radioactivity within the containment building and to the environment as a result of bulk defueling activities during normal or postulated accident conditions would be maintained at acceptable levels in compliance with applicable regulatory limits.

10.6.5 Use of Core Bore Machine for Bulk Defueling (NA)

10.6.6 Lower Core Support Assembly Defueling

- ***Purpose***. To dismantle and defuel the lower core support assembly and to partially defuel the lower reactor vessel head. Structural material removed from the lower core support assembly included the: (●) lower grid rib assembly; (●) lower grid forging; (●) distribution plate; and (●) in-core guide support plate. The lowest elliptical flow distributor and the gusseted in-core guide tubes would not be removed under this proposed activity. The use of the core bore machine, plasma arc cutting equipment, cavitating water jet, robot manipulators, automatic cutting equipment system, and other previously approved tools and equipment was included in this activity and associated safety evaluations.

- ***Evaluation: Radiological Release***. [177] The NRC's safety evaluation noted that the central zone of the plasma arc reached temperatures of 20,000 to 50,000 degrees Fahrenheit and was completely ionized. However, this high energy was quickly dissipated and primarily heated the conductive metal. Fuel on the metal surfaces was expected to be heated to the liquid or vapor state. Most fuel so heated would immediately oxidize, transfer its heat to the surrounding water, resolidify, and sink. Soluble isotopes trapped in the fuel matrix could become dissolved in the water. This possible increase in the concentration of radioactivity was not expected to be prohibitive or exceed that observed in the core drilling program. Safety concerns associated with the release of radioactivity from the reactor vessel to the environment were bounded by the licensee's safety evaluation report [178] for bulk defueling.

- ***NRC Review: Radiological Release***. [179] The NRC's safety evaluation concluded that the proposed activities could be accomplished without significant risk to the health and safety of the public, provided that they were in accordance with the limitations stated in the licensee's submittals and in the NRC's safety evaluation report. The limitations related to the control of any radiological release included ensuring that effluent from the defueling work platform off-gas system be routed to the vicinity of the containment building purge system suction point and also that the defueling work platform off-gas system and containment building purge system be operating whenever plasma arc cutting was performed.

10.6.7 Completion of Lower Core Support Assembly and Lower Head Defueling

- ***Purpose***. To remove the elliptical flow distributor and gusseted in-core guide tubes of the lower core support assembly and to defuel the reactor vessel lower head.

- ***Evaluation: Radiological Release***. (180) The licensee's safety evaluation concluded that safety concerns associated with the release of radioactivity from the reactor vessel to the environment were bounded by its safety evaluation report (181) for bulk defueling. The central zone of the plasma arc gas would reach temperatures of 20,000 to 50,000 degrees Fahrenheit and would completely ionize. However, this high temperature would be quickly dissipated and primarily heated the conductive metal. It was expected that fuel on the metal surfaces would also be heated to the liquid or vapor state. Most fuel that would be heated would immediately oxidize, transfer heat to the surrounding water, resolidify, and remain within the reactor vessel. Soluble isotopes trapped in the fuel matrix could become dissolved in the water; however, this possible increase in the concentration of radioactivity was not expected to be prohibitive or exceed that observed in the core bore drilling program during the now completed bulk defueling.

- ***NRC Review***. (182) Editor's Note: The NRC's safety evaluation report did not specifically address this evaluation topic.

10.6.8 Upper Core Support Assembly Defueling

- ***Purpose***. To cut and move the baffle plates and to defuel the upper core support assembly. This evaluation addressed the following activities: (●) cutting the baffle plates for later removal; (●) removing bolts from the baffle plates; (●) removing the baffle plates; and (●) removing core debris from the baffle plates and core former plates.

- ***Evaluation: Radiological Release***. (183) The licensee's safety evaluation concluded that safety concerns associated with the release of radioactivity from the reactor vessel to the environment were bounded by its safety evaluation report (184) for bulk defueling. The central zone of the plasma arc gas would reach temperatures of 20,000 to 50,000 degrees Fahrenheit and would completely ionize. However, this high temperature would be quickly dissipated and primarily heated the conductive metal. It was expected that fuel on the metal surfaces would also be heated to the liquid or vapor state. Most fuel that would be heated would immediately oxidize, transfer heat to the surrounding water, resolidify, and remain within the reactor vessel. Soluble isotopes trapped in the fuel matrix could become dissolved in the water; however, this possible increase in the concentration of radioactivity was not expected to be prohibitive or exceed that observed in the core bore drilling program during the previously completed bulk defueling.

Previous cutting operations in the lower core support assembly caused minor changes in radiation levels through the off-gas ventilation system because of krypton-85 gaseous releases. Defueling personnel were protected from direct or concentrated krypton-85 releases via off-gas system operation during cutting. Releases outside the containment building would be maintained within the limits given by the safety evaluation report for bulk defueling.

- ***NRC Review***. [185] Editor's Note: The NRC's safety evaluation report did not specifically address this evaluation topic.

10.7 Evaluations for Waste Management

10.7.1 EPICOR II

- ***Purpose***. To decontaminate accident-generated, intermediate-level radioactive wastewater being held in tanks in the auxiliary building. Later, the system was used to polish effluents from the submerged demineralizer system during the cleanup of highly radioactive water from the containment building sump, reactor coolant system, and reactor coolant drain tanks. Following the decommissioning of the submerged demineralizer system, EPICOR II was used to clean residual wastewater from decontaminating the structures and systems.

- ***Evaluation***. Editor's Note: The licensee's safety evaluation was not located.

- ***NRC Review***. The NRC's formal review was documented in the environmental assessment NUREG-0591, "Environmental Assessment on the Use of EPICOR II at Three Mile Island Unit 2," [186] issued October 1979. An updated environmental assessment that applied to EPICOR II and all other cleanup activities was documented in the PEIS, which was issued March 1981. Refer to these reports for details of the radiological release evaluation.

10.7.2 Submerged Demineralizer System

10.7.2.1 Submerged Demineralizer System Operations

- ***Purpose***. To decontaminate the containment building sump water and reactor coolant system (RCS) water using the submerged demineralizer system (SDS), followed by effluent polishing with the EPICOR II system.

- ***Evaluation: Radiological Release (Normal Operations)***. [187] The licensee documented its safety evaluation in a technical evaluation report (TER). [m] The estimate of offsite radiological exposures comprised analyses for the maximum individual dose and population dose.

  - *Source Terms for Liquid Effluents*. Liquid source terms were not required for this evaluation because effluent from the SDS would be returned to plant tankage for further deposition.

  - *Source Terms for Gaseous Effluents*. The plant vent system carrying airborne radioactive material was the first potential pathway for gaseous release. The second pathway was the

[m] Editor's Note: The licensee typically documented its safety evaluation of a system design and operations in what it called a "technical evaluation report."

ventilation system vent in the chemical cleaning building where the EPICOR II system was located. (EPICOR II was sometimes used as a process stream polisher.) Radioactive nuclides in the gaseous effluent came from water sampling and also from entrainment during the transfer of contaminated water to various tanks, filters, and ion exchange units.

– *Entrainment Factors*. Gaseous effluent source terms (in microcuries per second released to the atmosphere) were developed by assuming evaporation in the system. For this reason, an entrainment factor of 10^{-6} was assumed for the particulate radionuclides escaping from liquid to vapor. An entrainment factor of 0.0075 was assumed for iodine-129 per NUREG-0017, [188] "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors (PWR-GALE Code)," issued April 1976. In the case of evaporation by boiling, there was a higher rate of radionuclide release with off-gas vapors than would be expected from routine operation of pumps, valves, and water transfer. Therefore, these entrainment factors were considered to be conservative for the solution vent system during the transfer of pump water.

– *Tritium Release Rates*. The release of tritium from the plant vent (through the SDS) was calculated by assuming the air discharge from the vent was saturated with water vapor at 80 degrees Fahrenheit. At this temperature, 650 cubic feet per minute (cfm) of air would carry 500 grams of water vapor and correlate to 2.66×10^{-5} microcurie per cubic centimeter of tritium. The release of tritium from the chemical cleaning building (through the EPICOR II ventilation vent) was calculated from the evaporation of water at 100 degrees Fahrenheit from the tank inside the building. At this temperature, the tritium concentration in the discharge of the ventilation system was 7.15×10^{-9} microcurie per cubic centimeter at 8000 cfm.

– *Release Points*. Several vent systems were part of the final off-gas stream in the fuel handling building (FHB), some with lesser potential for contamination. However, for conservatism, the evaluation assumed that the total 650 cfm was in contact with water on the containment building basement floor, which at the time of this evaluation contained the highest specific activity of radionuclides. The tank vents in the EPICOR II system were the primary release point for airborne radioactive material from the chemical cleaning building.

– *Decontamination Factors*. A decontamination factor (DF) of 100 was assumed for particulates for the SDS off-gas system. No effluent treatment (i.e., a DF of 1) was assumed for tritium. The off-gas flow rate in the SDS off-gas system was 650 cfm. Radionuclides in the off-gas of the SDS were further diluted as they were mixed with existing gaseous effluent at TMI-2, giving a total off-gas volume flow rate of 100,650 cfm (plant vent stack). The evaluation further assumed that particulates passed through installed high-efficiency particulate air (HEPA) filters to give an additional DF of 100. However, no further effluent treatment was assumed for tritium or iodine-129. Therefore, the total DF for particulate including both the SDS off-gas system and treatment previously existing at TMI-2, was 10,000, while the DF for tritium and iodine-129 was 1.

In the chemical cleaning building, the EPICOR II tanks and the building ventilation system were equipped with HEPA filters and charcoal absorbers. Therefore, the total DF for particulates was assumed to be 10,000. For iodine-129, a DF of 20 was assumed and for tritium the DF was 1.

– *Particulate Off-Gas Rate*. The TER used the actual concentration of the containment building sump water and effluent water from the SDS to the EPICOR II system (refer to Table 6.1 of the TER). The data were based on the measured values listed in the TER (refer to Table 1.1 of the TER) and the expected performance of the SDS (refer to Section 3 of the TER). The pumping rate of water through the cleanup system was assumed to be 10 gallons per minute. From the assumed entrainment factor, the amount of radioactivity introduced into the off-gas was $[(f_i)(0.03785)]$ curie per minute, where f_i was the activity of an isotope per milliliter.

– *Release Results*. In the development of the iodine-129 source term, the results from the above method yielded a plant vent concentration of $2.09\text{x}10^{-10}$ microcurie per cubic centimeter. The contribution from evaporation increased this concentration to $6.15\text{x}10^{-10}$ microcurie per cubic centimeter.

– *SDS Effluents*. The TER listed (refer to Table 6.2 of the TER) the concentration of radionuclide source terms in the off-gas (from SDS effluents) following treatment by the SDS off-gas system and the existing plant effluent treatment system. Release rates for the various radionuclides were also provided. The concentrations in the plant effluent were below detectable levels for all isotopes except tritium.

– *EPICOR II Effluents*. The TER listed (refer to Table 6.3 of the TER) the concentration of radionuclide source terms in the chemical cleaning building (EPICOR II effluents) ventilation system following treatment. Release rates for the various nuclides were also provided. The concentrations in the effluent from the chemical cleaning building were below detectable levels for all isotopes except tritium.

o *Dose Assessment Methodology*. The radiological impact of the SDS was assessed by calculating radiation doses to the maximum individuals and populations living in the vicinity of the TMI site. Potential pathways for internal and external exposure of a person from radionuclides released to the atmosphere included: (●) inhalation; (●) ingestion of contaminated foods; (●) ingestion of contaminated water; (●) exposure from contaminated surfaces; and (●) exposures from immersion in the plume.

– *Exposure Periods*. The radiological impact was estimated using the methodology proposed in Regulatory Guide 1.109, Revision 1. The dose from a specified intake of a radionuclide to a reference organ was calculated over the remaining lifetime of an individual. The exposed person was assumed to be an adult (20 years of age) at the time of intake who lived to age 70. Thus, the accumulated dose was calculated by integrating the dose rate over a 50-year period (called the 50-year dose commitment).

- *Atmosphere Dispersion Factor*s. To calculate dose to the maximum exposed individual and to the population from the operation of the SDS, atmosphere dispersion factors were taken from previously published data and updated to 1980. The data were calculated for a semielevated point of release, including building wake effects for the SDS off-gas system. For the chemical cleaning building ventilation system vent (from EPICOR II effluents), data were calculated for a ground release. The values for atmosphere dispersion factor for each of the 16 sectors of the compass and downwind distance from the point of release were listed in the TER (refer to Table 6.4 of the TER). The data indicated that the point of maximum exposure to a hypothetical individual living near the site was 2413 meters away in the north-northeast direction since the most significant radiation release was from the plant vent stack (from SDS effluents).

- *Dry Deposition and Scavenging*. Radioactive particulates were removed from the atmosphere and deposited on the ground through mechanisms of dry deposition and scavenging. Dry deposition represented an integrated deposition of radioactive materials by gravitational settling, absorption, particle interception diffusion, and chemical-electrostatic effects. The deposition rate from the atmosphere for radioactive material was calculated using the methods described in Regulatory Guide 1.109 and Regulatory Guide 1.111, Revision 1. Scavenging of radionuclides in the plume was the process by which rain or snow washed out particles or dissolved gases and deposited them on the ground or water surfaces. The assessment included the effects of scavenging based on the methodology proposed in Regulatory Guide 1.111.

- *Internal organs*. Organ doses varied considerably for internal exposure from ingested versus inhaled materials because some radionuclides concentrated in certain organs of the body. This assessment calculated the dose to four organs: total body, bone, thyroid, and gastrointestinal tract. Radiation doses to the internal organs of children in the population varied from those received by an average adult because of differences in metabolism, organ size, and diet. Differences between the organ doses of a child and those of an average adult of more than a factor of 3 would be unusual for all pathways, except the atmosphere-pasture-cow-milk pathway for iodine-129, as this pathway contributed to the thyroid dose. Therefore, the dose to the thyroid for both the infant and child was estimated separately.

- *Other Assumptions*. Total dose commitments were calculated for the specified amount of each isotope released after 50 days of continuous release. Several conservative assumptions were made that tended to make dose commitments higher than would actually occur. For example, usage factors for the maximum exposed individual were taken from Table E-5 of Regulatory Guide 1.109. The assessment also assumed that all vegetables, both leafy and nonleafy, were grown at the point where dose was calculated and that an individual lived outdoors at the reference location 100 percent of the time. Since there were no releases from liquid effluent, the analysis assumed that the dose from ingestion of contaminated water was negligible. Regulatory Guide 1.109 contained additional details on the assumptions and methodology.

- *Results of Maximum Individual Dose*. The maximum dose to a hypothetical individual adult was calculated for the four organs and was based on the processing of 710,000 gallons of water. The estimated dose exposure levels, based on simultaneous releases from the plant vent (from SDS effluents) and the chemical cleaning building ventilation vent (from EPICOR II effluents), were as follows: (●) total body, 9.47×10^{-3} millirem; (●) bone, 3.57×10^{-4} millirem; (●) thyroid, 4.65×10^{-1} millirem; and (●) gastrointestinal tract, 8.82×10^{-3} millirem. This level of exposure to the total body represented about 0.2 percent of the allowable dose exposure (5 millirem) recommended in 10 CFR Part 50, Appendix I.

  - *Organ Dose*. The TER listed the contribution of the various exposure pathways to the dose of each organ considered (refer to Table 6.5 of the TER). Ingestion of contaminated foods was the primary mode of exposure, contributing 78 percent of the dose to total body, 98 percent to the bone, 99 percent to thyroid, and 76 percent to the gastrointestinal tract. Inhalation was the second most important pathway, while external exposure contributed less than 1 percent to each organ.

  - *Radionuclide Contribution*. The TER listed the contribution from each radionuclide to the total dose (refer to Table 6.6 of the TER). Tritium contributed about 93 percent of the dose to the total body and 99 percent of the dose to the gastrointestinal tract. Iodine-129 contributed 98 percent of the dose to thyroid and 58 percent of the dose to bone.

  - *Dominant Release Source*. Contribution to the individual organ doses was primarily from the SDS off-gas releases. The contribution to the dose within the chemical cleaning building from EPICOR II releases was less than 1 percent.

  - *Infant and Child Thyroid Doses*. Because of the possible age dependency of the dose to certain organs, the dose to the thyroid of an infant and child were estimated separately. This calculation yielded a dose of 0.12 millirem for an infant thyroid and 0.336 millirem for the thyroid of a child.

  - *Conclusion*. Even with the conservative assumptions incorporated into this assessment, results showed that the estimated dose to the maximum exposed individual was acceptable and met the recommended criteria for exposure to the public.

- *Results of Population Dose*. The estimated radiological exposure to the population from continuous operation of the SDS for 50 days was calculated using the methodology outlined above and as specified in Regulatory Guide 1.109. The population distribution was based on 1980 demographic data to a radius of 50 miles from the TMI site. The population integrated dose was calculated to be 0.15 person-rem total body and 4.42 person-rem for the thyroid.

- ***Evaluation: Radiological Release (Accident Analysis)***. [(189)] The licensee's safety evaluation stated that inherent safety features of the SDS and maximum use of existing site facilities minimized potential accidents involving the release of radionuclides to the environment. Hypothetical accidents during system operations were evaluated in the accident analysis, which assumed that zeolite beds were loaded to 40,000 curies. Should higher radiological loadings be determined to be appropriate, the accident analysis would be reassessed using the higher radiological loadings. [n] Scenarios that were evaluated included: (●) inadvertent pumping of containment sump water into the spent fuel pool; (●) pipe rupture on filter inlet line; (●) inadvertent lifting of SDS prefilter above pool surface; (●) inadvertent lifting of zeolite ion exchange vessel above pool surface; and (●) SDS shipping cast drop.

- *Inadvertent Pumping of Containment Sump Water into the Spent Fuel Pool*. A leak that was not immediately detected was postulated to occur in the effluent line from the final filter.

  - *Assumptions*. The analysis used the following assumptions: (●) Contaminated water would be released into the pool at a rate of 30 gallons per minute for a period of 15 minutes (i.e., 450 gallons with 340 curies). (●) Total activity was made up of 47 curies of cesium-134 and 293 curies of cesium-137, based on the measured concentrations as reported in Section 1 of the TER. (●) Uniform mixing of 233,000 gallons of pool water resulted in pool water contamination levels of 0.39 microcurie per milliliter.

  - *Occupational Exposure Effects*. The dose rate was calculated to an individual on the walkway at a point 6 feet above the surface of the water using equations for an infinite slab source and using published radionuclide decay data. The depth of water in the pool was 38 feet. The calculated maximum exposure rate at 6 feet above the surface was 116 milliroentgens per hour.

  - *Offsite Effects*. Airborne contamination releases as a result of this hypothetical accident were a small fraction of the limits specified in 10 CFR Part 20, Appendix B. [o, 190]. No significant increases at the site boundary from direct gamma exposure level were expected as a result of this hypothetical accident because of the spent fuel pool configuration, inherent shielding properties of the pool walls, and the distance to the site boundary.

  - *Conclusion*. This hypothetical accident was evaluated under conservative assumptions. Even though the analysis of this hypothetical accident indicated a radiation field of 116 milliroentgens per hour at a level 6 feet above the pool surface, area radiation monitor alarms would indicate its presence. Personnel would be evacuated to ensure that occupational exposures were limited. Potential offsite radiological consequences resulting from this hypothetical accident would be insignificant.

- *Pipe Rupture on Filter Inlet Line (above Water Level)*. A pipe rupture was postulated to occur above water level, in the inlet line to the filters at the southeast corner of the spent fuel pool.

---

[n] Editor's Note: Later TERs assumed a maximum loading of 60,000 curies based on actual load estimates during the early operations of the SDS.

[o] Editor's Note: 10 CFR Part 20, Appendix B, "Concentrations in Air and Water Above Natural Background," was subsequently revised and retitled, "Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposures; Effluent Concentrations; Concentrations for Release to Sewerage."

- *Assumptions*. The analysis made the following assumptions: (●) The leak proceeded for 15 minutes before the pump was stopped. (●) Contaminated water sprayed from around the lead brick shielding. (●) A total of 75 gallons of water was spread onto a surface area of 200 square feet and 675 gallons of contaminated water was drained into the pool. (●) Contaminated water contained 0.77 curie per gallon of activity, as cesium-134 and cesium-137 in the same concentration ratios that were assumed for the hypothetical accident (inadvertent pumping of sump water into the spent fuel pool).

- *Occupational Exposure Effects*. As a result of this hypothetical accident, the following significant effects were postulated: (●) Maximum gamma exposure rate at the surface of the contaminated floor area was calculated to be 8.64 rem per hour. (●) Maximum beta exposure rate at a point 3 feet above the surface of the contaminated floor area was estimated to be 384 rads per hour. (●) Exposure rate from the surface of the contaminated spent fuel pool water, at a point 6 feet above the surface, would be about 174 millirem per hour.

- *Offsite Effects*. Airborne contamination releases at the site boundary from this hypothetical accident were below those limits specified in 10 CFR Part 20, Appendix B, and would not be significant because of the shielding characteristics of the FHB walls and the distance to the site boundary.

- *Conclusion*. This hypothetical accident and its consequences would pose no threat to the public health and safety, and there would be no accumulation of occupational radiological exposure. Even though high surface contamination levels existed on the floor area and the spent fuel pool water was contaminated such that the total body could be exposed to relatively high radiation levels, the area radiation monitors would indicate the presence of high radiation. Personnel would be evacuated from the area to ensure that occupational exposures were limited.

- *Inadvertent Lifting of SDS Prefilter above Pool Surface*. The SDS prefilter was installed to filter out solids in the untreated contaminated water before the water was processed by the SDS ion exchange vessels. This cartridge-type filter, which was the first of two in series, was housed in a containment enclosure and submerged in the spent fuel pool.

  - *Assumptions*. This postulated accident assumed that the overhead crane moved toward the loading bay after pulling one expended SDS filter to the maximum height of 8 feet below the pool surface because of a failure in the crane control system. As the crane moved toward the bay, the handling tool hits the end of the pool and the filter was dragged from the water, exposing operating personnel. Analysis of the accident was performed by using a point source approximation and calculating the dose rate at a distance of 15 feet from the filter. The calculated dose rate was 21 rem per hour and was based on an assumed filter loading of 1000 curies.

  - *Occupational Exposure Effects*. As the filter assembly neared the surface of the spent fuel pool water area, radiation monitor alarms would be sounded announcing the presence of high-radiation fields. Personnel would be evacuated from the area to ensure that occupational exposures were limited.

- *Offsite Effects*. Airborne contamination as a result of this hypothetical accident would not occur, since the particulate activity was fixed on the filter elements contained within the filter housing. The increase in the radiation level at the site boundary would not be significant because of the shielding characteristics of the fuel building walls and the distance to the site boundary.

- *Conclusion*. The public health and safety would not be compromised as a consequence of this hypothetical accident.

o *Inadvertent Lifting of Zeolite Ion Exchange Vessel above Pool Surface*. The first part of the SDS ion exchange system consisted of up to six underwater vessels (24.5 inches in diameter by 54.5 inches long). Each vessel contained about 8 cubic feet of homogeneously mixed zeolite ion exchange media. Zeolite media volumes and mixtures could be changed to reflect different processing scenarios.

- *Assumptions*. This postulated accident assumed that because of multiple failures, a zeolite vessel was lifted from the spent fuel pool resulting in the exposure of plant operating personnel. The accident was analyzed by modeling the zeolite ion exchanger bed in a cylindrical geometry and calculating the dose rate at a distance of 20 feet from the surface of the zeolite ion exchanger. The calculated dose rate was about 297 rem per hour based on an estimated zeolite ion exchange bed loading of about 5390 curies of cesium-134 and about 34,600 curies of cesium-137.

- *Occupational Exposure Effects*. As the zeolite vessel neared the surface of the spent fuel pool water, area radiation monitor alarms would sound automatically, announcing the presence of high-radiation fields. Personnel would be evacuated from the area to reduce occupational doses. Airborne contamination would not occur since the activity was fixed on the zeolites.

- *Offsite Effects*. The increase in the radiation level at the site boundary would not be significant because of the shielding provided by the FHB walls and the distance to the site boundary.

- *Conclusion*. The public health and safety would not be endangered as a result of this hypothetical accident. Occupational exposures would be minimized by evacuation of the area.

o *Inadvertent Drop of SDS Shipping Cask*. The spent SDS vessel or filter was loaded into a shipping cask (transportation package) inside the spent fuel pool. The loaded cask was raised by a crane from the spent fuel pool to the truck transporter parked in the FHB truck bay. Pool water provided the shielding necessary to reduce radiation levels during cask loadings.

- *Assumptions*. This postulated accident assumed that a failure in SDS shipping cask handling equipment caused an SDS cask containing a zeolite ion exchange vessel to be dropped from the FHB crane to the floor at the 305-foot elevation. The analysis made the following assumptions: (●) The SDS shipping cask was dropped from the maximum crane lift height. (●) Upon impact with the floor at the 305-foot elevation, the SDS shipping cask ruptured along with the zeolite vessel, exposing the dewatered zeolite resins to the FHB atmosphere. (●) Radiation sources on the zeolite ion exchange media included about 5390 curies of cesium-134 and about 34,600 curies of cesium-137. (●) Contribution from other isotopes on the zeolite media and residual containment building sump water in the ion exchange media was negligible. (●) A factor of 1×10^{-4} percent of the isotopes was instantaneously released to the FHB atmosphere. This assumption was conservative because the isotopes were absorbed by the zeolite media. (●) FHB HEPA filters were assumed to have an efficiency of 99 percent.

- *Occupational Effects*. Assuming that the SDS shipping cask ruptured completely and exposed the zeolite ion exchanger containing the activity as assumed above, the calculated dose rate was about 297 rem per hour at a distance of 20 feet. Upon the rupture of the cask, radiation monitors would sound announcing the presence of high-radiation fields. Personnel would be evacuated from the area to reduce radiation exposures. Airborne contamination would not occur if the zeolite ion exchange vessel remained intact. Even with the assumption that the vessels ruptured and radioactive material became airborne, the airborne activity would be reduced to acceptable levels by the FHB ventilation system before atmospheric release.

- *Operational Effects*. Impacts on systems, structures, and components that could adversely affect the ability to operate both reactor units safely, to transfer a load or unload fuel safely, or to maintain both plants in a safe cold-shutdown condition were considered. The analysis demonstrated that a postulated SDS cask drop along the proposed travel path would not adversely affect either Unit 1 or Unit 2.

- *Offsite Effects*. The increase in radiation level at the site boundary would not be significant because of the shielding provided by the FHB walls and the distance to the site boundary. If the SDS cask ruptured exposing the zeolite ion exchanger, with the assumption that radioactive material escaped, the whole-body dose due to the released activity at the site boundary would be less than 1 millirem for both beta and gamma radiation.

- *Conclusion*. The public health and safety would not be compromised as a consequence of the hypothetical accident.

- ***NRC Review: Radiological Release (Normal Operations***). [(191)] The NRC evaluated the capability of the SDS to keep the levels of radioactivity in gaseous effluents ALARA, in accordance with 10 CFR Part 20, the existing Appendix B [(192)] to the operating license (technical specifications for environmental protection), and the criteria specified in Appendix R [193] of the PEIS that proposed additional requirements to the environmental technical specifications. The NRC documented its safety evaluation in NUREG-0796, [194] "Operation of the Submerged Demineralizer System at Three Mile Island Nuclear Station, Unit No. 2," issued June 1981.

The NRC's safety evaluation considered three release sources: (●) direct discharge from the SDS to the fuel handling building ventilation system (FHBVS); (●) direct leakage from SDS piping and valves to the FHB atmosphere; and (●) evaporation of tritiated water. Offsite consequences from the first release source were conservatively estimated in the PEIS and summarized in the NUREG-0796. The analysis performed in the PEIS was based on a generic water processing system. The updated analysis in NUREG-0796 was based on the final SDS design and actual radiological samples of the containment building sump water and RCS water. Both evaluations are presented below.

- *Release Sources*. There were no liquid discharges of radioactive material from the SDS and no direct gaseous discharges, but the SDS interfaced directly with the FHBVS and was housed in the FHB and serviced by the FHBVS. The FHBVS discharged to the environment through the plant stack along with other waste streams. There were three potential sources of radioactive gaseous influents to the FHBVS resulting from the operation of the SDS. The first source was the direct discharge from the SDS to the FHBVS. The second was potential leakage into the FHB from SDS piping and valves that were not submerged in the fuel pool. The third source was the evaporation of tritiated water from the pool. Any leakage from SDS piping and valves that were submerged in the pool would mix with pool water in the underwater containment boxes and would be processed by the leakage containment ion exchangers.

- *Gaseous Release: Direct Discharge from the SDS to the FHBVS (PEIS)*. The original PEIS evaluation considered the potential for leakage from underwater liquid waste processing system components and directly to the FHBVS. The PEIS did not credit a separate off-gas system for the liquid waste processing system; however, an air filtration system was modeled for the FHBVS.

  - *Release Model (PEIS)*. The PEIS release model conservatively assumed that $1x10^{-4}$ of the curie inventory in the influent liquid waste stream to the processing system (i.e., the SDS for this case) would become aerosolized during processing. The model further assumed that this gaseous waste stream was carried by subsequent movement of air in the process vent system, then ultimately combined with building ventilation air (i.e., the FHBVS for this case), treated by the air cleaning system for the building, monitored by the radiation detectors in the plant stack, and finally released to the environment by controlled discharge. The PEIS model did not assume that there was an air filtration system in the liquid waste processing system; therefore, no credit was given for the SDS air filtration unit in estimating the gaseous effluents to the environment.

  - *Release Estimates (PEIS)*: Regarding the discharge from the SDS, the NRC conservatively estimated in the PEIS the radioactive gaseous effluents to the environment that would result from the processing of containment building sump and RCS water. The principal radionuclides in these releases from the processing of containment building sump water were: (●) 6.4×10^{-3} curie for cesium-134; (●) 4.1×10^{-2} curie for cesium-137; (●) 6.9×10^{-4} curie for strontium-90; and (●) 2.5×10^{-1} curie for tritium. Releases from the processing of RCS water were much lower, except for strontium-90 for which the releases were similar to those for the processing of sump water.

- *Dose Estimates (PEIS)*. The dose estimates for the maximum exposed individual that pertained to these releases when summed across release pathways (inhalation, ground shine, vegetable/milk use) for each offsite location (nearest garden, nearest milk goat, and nearest cow and garden) ranged from 1 to 0.01 millirem for total body and for bone (refer to Table 15 of the SER). Doses were calculated for four age groups (adults, teenagers, children, and infants). The table provided the highest dose estimates for each age group considered.

o *Gaseous Release: Direct Discharge from the SDS to the FHBVS (Revised)*. The NRC reevaluated the PEIS results of the potential for leakage from underwater SDS components through the SDS gaseous waste treatment subsystem (known as the SDS off-gas system) and directly to the FHBVS.

- *Release Model (Revised)*. The NRC reevaluated the PEIS estimates of radioactive gaseous effluents to the environment resulting from wastewater processing in the SDS, based on the filtration capability of the SDS and the availability of more recent information on the radionuclide distribution in the containment building sump and RCS water. The NRC's modeling assumptions in the reevaluation included the following: (●) The PEIS model was used for the fraction (1×10^{-4}) of inventories that became an aerosol during SDS process operations. (●) The movement of air in the gaseous waste treatment subsystem (GWTS) of the SDS carried this gaseous waste stream, and it was treated by the GWTS air filtration unit. (●) For conservatism, an overall particulate DF of 10 (i.e., a removal efficiency of 90 percent) was applied for both of the unit HEPA filters (i.e., two in series) even though they were each tested to a removal efficiency greater than 99.95 percent. (●) The PEIS model was applied for particulate removal in ventilation air filtration systems, and an overall particulate DF of 1000 was applied for the FHBVS. (The gaseous effluent from the GWTS air filtration unit was directed to the FHBVS and treated by the ventilation system HEPA filters.)

- *Release Estimates (Revised)*. Based on the source term model, as described above, the principal radionuclides in the effluents from processing containment building sump water were: (●) 5.0×10^{-4} curie for cesium-134; (●) 3.7×10^{-3} curie for cesium-137; (●) 1.3×10^{-4} curie for strontium-90; and (●) 2.0×10^{-1} curie for tritium. Effluents from the processing of RCS water were lower (refer to Table 16 of the SER).

- *Dose Estimates (Revised)*. The dose estimates for the maximum exposed individual that pertained to the effluents when summed across release pathways (inhalation, ground shine, vegetable/milk use) for each offsite location (nearest garden, nearest milk goat, and nearest cow and garden) ranged from 10^{-2} to 10^{-3} millirem for total body and for bone (refer to Table 17 in the SER). Doses were calculated for four age groups (adults, teenagers, children, and infants). The highest dose estimates for each age group considered were listed. These dose estimates were considerably smaller in every case than the corresponding PEIS values.

- *Gaseous Release: Direct Leakage from SDS Piping and Valves to the FHB Atmosphere*. The NRC considered the potential for direct leakage from SDS piping and valves that were not submerged in spent fuel pool "B" and off-gassed through the GWTS. This postulated leakage was evaluated to determine if there was a significant potential for generating airborne contamination in the FHB atmosphere. SDS design features that prevented leakage of components above water included the piping integrity design and the use of containment manifolds.

  – *Piping Design*. The SDS piping was constructed with a design pressure of 150 pounds per square inch gauge (psig) pressure and successfully pressure tested at 120 to 150 percent of its design pressure (i.e., tested at 180 to 225 psig). The SDS was a relatively low-pressure system that operated at ambient temperature with a normal operating pressure of about 75 psig, and the likelihood for leakage was correspondingly minimized at the low operating pressures.

  – *Containment Manifolds*. The principal valves in the SDS that were used to control process operations were housed in containment manifolds (i.e., RCS cleanup manifold, high-radiation filter manifold, feed pump manifold, ion exchanger manifold, and beta monitoring manifold). These manifolds were maintained at negative pressure by the GWTS. Thus, any leakage that became airborne in the valve containment manifolds would be processed in the GWTS. This source of activity was adequately bounded by the NRC's conservative assumption of the fraction of the effluent activity (i.e., 1×10^{-4}) that became airborne during SDS process operations.

  – *Conclusion*. The NRC concluded that the design provisions for system integrity and leak-off collection precluded a significant potential for the generation of airborne activity in the FHB from the leakage of ex-pool SDS piping and valves.

- *Gaseous Release: Evaporation of Tritiated Water*. The NRC also considered the potential impact of the evaporation of processed tritiated water from the "B" spent fuel pool. The "B" spent fuel pool was filled with 240,000 gallons of processed accident-generated water that had a tritium concentration of about 0.15 microcurie per milliliter. This volume corresponded to a tritium inventory of 136 curies. In the PEIS evaluation, the NRC estimated the annual rate of evaporation of tritiated water from the plant that resulted from the use of processed, accident-generated water to shield a submerged ion exchange processing system in the "B" spent fuel pool. Based on an average tritium concentration of 0.13 microcurie per milliliter in the existing inventory (i.e., 740,000 gallons) of processed water in onsite storage, the NRC estimated the rate of tritium evaporation and subsequent loss to the environment to be 50 curies per year.

Based on this release rate, the NRC estimated that the doses for the maximum exposed individual when summed across release pathways (inhalation, ground shine, vegetable/milk use) for each offsite location (nearest garden, nearest milk goat, and nearest cow and garden) ranged from 10^{-2} to 10^{-3} millirem for total body (refer to Table 18 of the SER).

- *Conclusion*. The NRC concluded that the SDS was capable of keeping the levels of radioactivity in gaseous effluents ALARA, in accordance with 10 CFR Part 20, the existing Appendix B technical specifications, and the limits specified in Appendix B to the PEIS.

• ***NRC Review: Environmental Release (Normal Operations)***. [195] The NRC evaluated the ability of the SDS to maintain releases of radioactive materials in gaseous effluents to unrestricted areas below the limits in 10 CFR Part 20, Appendix B, Table II, Column 1, during periods of process operations, at the maximum flow rate in the SDS. The existing Appendix B to the technical specifications implemented the requirements of 10 CFR Part 20. The delivery and filtration of the containment building sump water to the upper tank farm would occur at a flow rate varying from 10 to 30 gallons per minute.

- *Assumptions*. The NRC calculated the instantaneous release rate (microcuries per second) of the principal radionuclides discharged to the environment for comparison with the limits in Appendix B to the technical specifications. Assumptions for this analysis included: (●) maximum process flow rate of 30 gallons per minute; (●) partitioning of $1x10^{-4}$ of the effluent activity to the SDS GWST; (●) GWTS particulate DF of 10; (●) GWTS discharge flow rate of 1000 cfm; (●) FHBVS particulate DF of 1000; (●) FHBVS flow rate of 40,000 cfm; and (●) total plant stack flow rate of 100,000 cfm.

- *Results*. The calculated release rates of 0.003 microcurie per second for particulates and 0.3 microcurie per second for tritium were well within the technical specification limits of 0.3 microcurie per second for particulates and 30,000 microcuries per second for tritium.

- *Conclusion*. The NRC concluded that the SDS was capable of maintaining releases of radioactive materials in gaseous effluents to unrestricted areas below the limits in 10 CFR Part 20, Appendix B, Table II, Column 1.

• ***NRC Review: Radiological Release (Accident Analysis)***. [196] The NRC's safety evaluation stated that quantities of radioactive materials that could be released during postulated accidents associated with SDS operations were based on the licensee's TER [197] and system description [198] for the SDS. After reviewing these documents, the NRC considered the following postulated accidents as bounding: (●) shipping cask drop; (●) HEPA filter failure during processing of containment building sump or RCS water; and (●) failure of the 60,000-gallon feed tank system to the SDS. As described in the following subsections, the NRC analyzed the accidents that could occur during the operation of the SDS. The NRC analyses showed that offsite radiological consequences from each accident were within the limits of 10 CFR Part 20 or the guidelines of 10 CFR Part 100.

- *Shipping Cask Drop*. The worst possible accident would be a shipping cask drop that could occur during the transfer of a first-stage ion exchange vessel in a shipping cask within the FHB.

  – *Source Term*. The NRC stated in the SER that the radiological loading on the first-stage ion exchange resins would be administratively limited to 60,000 curies. For the purpose of this evaluation, the accident scenario was based on a loading of 120,000 curies (the maximum curie loading ever expected).

  – *Drop Height*. The cask was designed to retain its integrity during a 30-foot drop; however, at one point during the cask movement, the cask could fall a distance of 60 feet. For this 60-foot drop accident, the NRC's evaluation assumed that the ion exchange vessel in the cask and cask seal would be breached.

  – *Release*. This accident was evaluated in the PEIS where the NRC estimated that about 12 curies of mostly cesium would be released to the FHB atmosphere. This release was based on a curie loading of 120,000 curies of dewatered ion exchange media and a partition factor of 10^{-4}. Since the cask drop was assumed to occur in the FHB, the contaminated air would be treated by the FHBVS HEPA filters, and the estimated release to the environment was 0.012 curie.

  – *Offsite Doses*. Table 21 of the SER presented the resulting dose estimates for the maximum exposed individual as a result of this accident. The individual summed doses for total body, bone, and liver from multiple pathways (inhalation, ground shine, and vegetable/milk use) at various offsite locations (nearest garden, nearest milk goat, and nearest cow and garden) ranged from 5.4×10^{-4} to 9.4×10^{-3} millirem.

  – *Conclusion*. The NRC concluded that offsite radiological consequences were well within the guidelines of 10 CFR Part 100.

- *HEPA Filter Failure During Processing of Highly Contaminated Containment Building Sump or RCS Water*. In the PEIS evaluation, the NRC evaluated the consequences of a HEPA filter failure during the processing of the highly contaminated containment building sump water and RCS water.

  – *Assumptions*. In the PEIS evaluation, the radioactivity, which was estimated to become aerosolized during the processing of the containment building sump or RCS water, was assumed to be deposited on the FHBVS HEPA filter. The HEPA filter was subsequently assumed to fail and release a fraction (0.001) of its contents to the environment. The PEIS evaluation did not consider HEPA filters (i.e., two filters in series) as part of the SDS design; thus, the evaluation did not assume activity deposition in the SDS air filtration unit. In actuality, any radioactive material that became aerosolized during SDS process operations would collect on the first HEPA filter in the SDS GWTS. For any significant radioactivity to reach the environment during a postulated HEPA filter failure, the downstream HEPA filter in the GWTS and the two HEPA filters in the FHBVS would also have to fail.

– *Results*. Table 22 of the SER presented the estimated doses for the maximum exposed individual from this postulated accident. For processing of containment building sump water, the doses were 3.3 millirem for total body, 16 millirem for bone, and 12 millirem for liver. For processing RCS water, the doses were 1.5 millirem for total body, 6.0 millirem for bone, and 0.28 millirem for liver. Doses were calculated for four age groups (adults, teenagers, children, and infants). The table provided the highest dose estimates for each age group considered.

– *Conclusion*. The NRC concluded that this postulated event was well bounded by the PEIS evaluation and the dose consequences (listed above), which were well within the guidelines of 10 CFR Part 100.

o *Failure of the 60,000-Gallon Feed Tank System to the SDS or of a 77,000-Gallon Reactor Coolant Bleed Tank*. In the PEIS evaluation, the NRC evaluated the consequences of leaking the entire 700,000 gallons of water in the bottom of the containment building into the ground water. This leakage would ultimately reach the Susquehanna River and then the public through the drinking water dose transport pathway. This was the bounding accident involving a release of liquid radioactivity from TMI-2. The NRC's evaluation indicated that concentrations of the principal radionuclides (i.e., strontium-90, cesium-137, and tritium) at the nearest drinking water intake were well below the concentration limits in 10 CFR Part 20. The peak concentrations in the Susquehanna River, based on leakage of 700,000 gallons from the containment building of about 500,000 curies of the principal radionuclides, were as follows: (●) 5.2×10^{-7} microcurie per milliliter for tritium, (●) 5.1×10^{-10} microcurie per milliliter for cesium-137, and (●) 5.1×10^{-8} microcurie per milliliter for strontium-90. The maximum permissible concentrations from 10 CFR Part 20 for tritium, cesium-137, and strontium-90 were 3.0×10^{-3}, 2.0×10^{-5}, and 3.0×10^{-7}.

In the case of the SDS, a similar type of accident would involve the rupture of the four interconnected SDS feed tanks (15,000-gallon capacity each, total capacity 60,000 gallons) in the "A" spent fuel pool containing containment building sump water of a 77,000-gallon reactor coolant bleed tank, containing RCS water. The rupture of these tanks could subsequently release water into the ground and eventually to the public via the water transport pathway. However, since the potential volume released was considerably less than that analyzed in the PEIS, the consequences of such an accident would be correspondingly lower.

The NRC concluded that this postulated event was well bounded by the results of the PEIS evaluation of containment building sump water leakage and within the limits of 10 CFR Part 20.

- *Conclusion*. Based on the results of the above accident evaluations, the NRC concluded that the postulated failure of the feed tanks to the SDS would not result in radionuclide concentrations in the Susquehanna River that exceeded the limits in 10 CFR Part 20. Further, the consequences resulting from the postulated drop of a zeolite vessel with high specific activity or the failure of a HEPA filter during SDS process operations were well within the guidelines of 10 CFR Part 100.

10.7.2.2 Submerged Demineralizer System Liner Recombiner and Vacuum Outgassing System

- ***Purpose***. To eliminate the potential of a combustible hydrogen and oxygen mixture existing in the submerged demineralizer system (SDS) liners and to facilitate the ultimate shipment and burial of the SDS liners. The liner recombiner and vacuum outgassing system (LRVOS) was designed to remove moisture by evaporation from the zeolite beds of SDS spent liners. This operation dried the beds but did not remove the water in the zeolite.

- ***Evaluation: Radiological Release***. [199, 200] The licensee's safety evaluation stated that the release of fission products from the zeolite beds due to vacuum pumping was not expected to occur as stated in the system performance evaluation. Release of significant amounts of resin fines to the environment was highly unlikely because of the vacuum system filtering capability and the SDS ventilation system that consisted of a roughing filter, high-efficiency particulate air (HEPA) filter, charcoal filter, and another HEPA filter. In addition, the exhaust of the SDS ventilation was monitored by a radiation sampler, then sent to the auxiliary and fuel handling building (FHB) ventilation system. Since no release of fission products was expected because of the filtering trains involved, the evaluation assumed that gaseous and particulate releases from the vacuum system operation would be below those activities attributed to normal SDS spent liner dewatering. The licensee concluded that the operation of the LRVOS would comply with the technical specification requirements.

- ***NRC Review: Radiological Release***. [201] The licensee's analysis showed that no measurable releases of radioactive material from the SDS zeolite bed were expected during the vacuum pumping through the LRVOS. The conclusion was based on the type of water removal (i.e., water vapor transport under vacuum) and the extremely low airflow rates. The NRC review concluded that, during normal system operation, minor amounts of particulate entrainment would occur and the potential for releases to the FHB were negligible. The two major entrainment controls were the water separator/mechanical filters in the LRVOS and the SDS off-gas system, which contained a roughing filter, HEPA filter, charcoal filter, and final HEPA filter. As part of the functional test, the LRVOS, including the catalyst loading tool and vent hose connections, were leak tested to ensure that the SDS liner gas space did not leak directly to the spent fuel pool or atmosphere in the FHB. If leakage should occur during operations, early indications would be available to the system operators (e.g., loss of vacuum, increased water level in knockout drum, high flow rate), and the LRVOS and SDS liner would be isolated according to operating procedures.

The licensee's safety assessment of accidents addressed an event in which the catalyst did not recombine the hydrogen and oxygen gases generated, and then liner pressure and combustible gas control would be reestablished by use of the SDS off-gas header. All other credible accidents were enveloped by the licensee's previous technical evaluation report (202) for the SDS and the associated safety evaluation report (203) by the NRC. The NRC concurred with these analyses and concluded that the additional vacuum drying and recombiner loading, as a preparation for shipment, did not measurably increase the amount of liner movements and handling operations; therefore, preparations did not increase the risk of an SDS liner accident. Additionally, an SDS liner loaded with a catalytic recombiner system at subatmospheric pressure (less than 5 pounds per square inch absolute pressure) provided a lower energy source and potentially smaller consequence than non-recombiner-loaded SDS liners. The NRC also concluded, based on the LRVOS designs and safety features, the SDS off-gas filter and monitoring systems, and the FHB ventilation filter system, that there was substantial defense in depth for radioactive material controls and that the health and safety of the public was adequately protected.

10.8 Endnotes

Reference citations are exact filenames of documents on the Digital Versatile Discs (DVDs) to NUREG/KM-0001, Supplement 1, (204) "Three Mile Island Accident of 1979 Knowledge Management Digest: Recovery and Cleanup," issued June 2016, DVD document filenames start with a full date (YYY-MM-DD) or end with a partial date (YYYY-DD). Citations for references that are not on a DVD begin with the author organization or name(s), with the document title in quotation marks. Contents of the DVDs are available at the Idaho National Laboratory Web site (https://tmi2kml.inl.gov).

---

[1] NUREG-0683, Vol. 1, PEIS-Decontamination and Disposal of Radioactive Wastes Resulting from TMI-2 (1981-03)

[2] DOE-ID-10276, TMI-2 Lessons Learned by the DOE 1979-1990 (1990-03)

[3] NUREGCR-1250V2PT2, TMI, A Report to The Commissioners And to The Public (Rogovin Report) (1980-01) [See Page 348]

[4] (1979-10-01) GPU, Water Storage Assessment

[5] (1979-05-25) NRC Policy, Direct Staff to Prepare Environmental Assessment for Decontaminate and Dispose Waste Water

[6] (1980-02-28) NRC, Report of Special Task Force on TMI Cleanup

[7] USNRC, "Interim Criteria for Radiological Effluents from TMI-2 Data Gathering and Maintenance Operation," Commission Paper SECY-80-175, April 4, 1980 [Available on NUREG/KM-0001 DVD-2]

[8] NUREG-0662V1, Environmental Assessment for Decontamination of TMI-2 Reactor Building Atmosphere, Final Report (1980-05)

[9] (1980-06-13) NRC Order (6-12-1980), Orders Release Krypton-85 from the Reactor Building Atmosphere by Controlled Purge

[10] (1979-11-21) NRC Policy, Notice of Intent to Prepare Programmatic EIS (TMI-2)

[11] NUREG-0683, Vol. 1, PEIS-Decontamination and Disposal of Radioactive Wastes Resulting from TMI-2 (1981-03)

[12] (1981-04-27) NRC Policy, Programmatic Environmental Impact Statement of the Cleanup of TMI-2 (46 FR 24764)
[13] (1979-04-19) NRC, TMI-2 Ground Water Study.pdf
[14] NUREG-0662V1, Environmental Assessment for Decontamination of TMI-2 Reactor Building Atmosphere, Final Report (1980-05)
[15] NUREG-0681, Environmental Assess of Radiological Effluents from Data Gathering and Maint Operation on TMI-2 (1980-04)
[16] NUREG-0683, SUPP. 2, PEIS, Disposal of Accident-Generated Water (1987-06)
[17] NUREG-0683, SUPP. 3, PEIS, Post-Defueling Storage and Cleanup (1989-08)
[18] USNRC, "1981 NRC Annual Report," NUREG-0920 [Available on NUREG/KM-0001 DVD-1]
[19] USNRC, "Three Mile Island Accident of 1979 Knowledge Management Digest: The Cleanup Experience: A Literature Review," NUREG/KM-0001, Supplement 2, December 2020 [Available at nrc.gov]
[20] NUREG-0683, SUPP. 1, PEIS, Occupational Radiation Dose (1984-10)
[21] USNRC, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Regulatory Guide 1.109, Rev. 1, October 1977 [Current and past regulatory guides are available at nrc.gov]
[22] USNRC, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors," Regulatory Guide 1.111, July 1977 [Current and past regulatory guides are available at nrc.gov]
[23] USNRC, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," Regulatory Guide 1.145, August 1979 [Current and past regulatory guides are available at nrc.gov]
[24] USNRC, "Design, Testing, and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Absorption Units of Light Water Cooled Nuclear Power Plants," Regulatory Guide 1.140, Rev. 1, October 1979 [Current and past regulatory guides are available at nrc.gov]
[25] NUREG-0683, SUPP. 2, PEIS, Disposal of Accident-Generated Water (1987-06)
[26] (1982-06-15) GPU Safety Evaluation, Axial Power Shaping Rod Insertion Test
[27] (1982-05-17) Order Amendment
[28] (1982-07-06) GPU Safety Evaluation, Insertion Camera Through Reactor Vessel Leadscrew Opening, Rev. 2
[29] (1982-07-13) NRC Review, Control Rod Drive Mechanism Quick Look Camera Inspection (re 07-06-1982) (2)
[30] (1983-05-19) GPU Safety Analysis, Radiation Characterization Under Reactor Vessel Head (No Polar Crane), Rev. 0
[31] (1982-09-23) GPU Safety Evaluation, Ongoing Containment Building Decontamination Activities, Rev. 0
[32] (1982-09-24) NRC Safety Evaluation, Extended Containment Building Decontamination Effort
[33] (1983-07-13) NRC Review, Reactor Vessel Underhead Characterization (re Various Letters)
[34] (1986-06-11) GPU Safety Evaluation, Core Stratification Sample Acquisition, Rev. 4
[35] (1985-10-10) GPU Safety Evaluation, Early Defueling of the TMI-2 Reactor Vessel, Rev. 4
[36] (1985-04-19) GPU Safety Evaluation, Heavy Load Handling Over Reactor Vessel SER
[37] (1993-11-12) GPU, Proposed POL Submittals-TMI Offsite Dose Calculation Manual [Note: This version is Revision 7, dated November 1993, provided as an example.]
[38] (1986-05-05) NRC Safety Evaluation, Core Stratification Sample Acquisition SER, Rev. 1 (re 08-30-1985, 12-31-1985)

[39] (1989-08-18) GPU Safety Evaluation, Remove Metallurgical Samples from Reactor Vessel SER
[40] (1989-10-20) GPU Safety Evaluation, Remove Metallurgical Samples from Vessel, Rev. 1 (re 08-18-1989) (effective pages)
[41] (1989-11-28) NRC Safety Evaluation, Reactor Vessel Lower Head Metallurgical Sampling (re 08-18-1989)
[42] (1985-10-29) NRC Safety Evaluation, Commencement of TMI-2 Preliminary Defueling Operations (re 10-24-1985)
[43] (1985-11-12) NRC Safety Evaluation, Early Defueling of the TMI-2 Reactor Vessel (10-10-1985)
[44] (1986-07-24) NRC Review, Core Region Bulk Defueling Operations Limited to the Core Region (re 05-15, 07-23-1986)
[45] (1986-03-28) GPU Safety Evaluation, Reactor Building Decontamination and Dose Reduction Activities for 1986, Rev. 0
[46] USNRC, "Calculation of Annual Dose to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Regulatory Guide 1.109 [Current and past regulatory guides are available at nrc.gov]
[47] (1993-11-12) GPU, Proposed POL Submittals-TMI Offsite Dose Calculation Manual [Note: This version is Revision 7, dated November 1993, provided as an example.]
[48] (1984-02-03) NRC Safety Evaluation, Containment Decontamination and Dose Reduction (re 09-29, 01-23-1983)
[49] (1983-02-18) GPU Safety Evaluation, Polar Crane Load Test
[50] (1982-09-23) GPU Safety Evaluation, Ongoing Containment Building Decontamination Activities, Rev. 0
[51] (1983-02-18) GPU Safety Evaluation, Polar Crane Load Test
[52] USNRC, "Control of Heavy Loads at Nuclear Power Plants," NUREG-0612, July 1980 [Available at nrc.gov]
[53] *U.S. Code of Federal Regulations*, "Reactor Site Criteria," Part 100, Chapter I, Title 10, "Energy"
[54] (1983-02-18) GPU Safety Evaluation, Polar Crane Load Test
[55] (1983-11-18) NRC Safety Evaluation, Reactor Building Polar Crane Load Test
[56] (1983-11-18) NRC Safety Evaluation, Reactor Building Polar Crane Load Test
[57] (1983-09-29) GPU, Plans for First Pass Stud Detensioning for Head Removal
[58] (1984-01-30) GPU Response to NRC, First Pass Stud Detensioning SER (re 09-29, 12-06-1983)
[59] (1984-02-17) NRC Safety Evaluation, First Pass Vessel Stud Detensioning (re 09-29, 12-06-1983, 01-30-1984)
[60] (1984-03-09) GPU Safety Evaluation, Head Removal, Rev. 5
[61] USNRC, "Control of Heavy Loads at Nuclear Power Plants," NUREG-0612, July 1980 [Available at nrc.gov]
[62] (1984-03-09) GPU Safety Evaluation, Head Removal, Rev. 5
[63] (1984-07-17) NRC Safety Evaluation, Reactor Pressure Vessel Head Lift (re -3-09-1984)
[64] (1984-07-17) NRC Safety Evaluation, Reactor Pressure Vessel Head Lift (re -3-09-1984)
[65] USNRC, "Control of Heavy Loads at Nuclear Power Plants," NUREG-0612, July 1980 [Available at nrc.gov]
[66] (1986-06-02) GPU Safety Evaluation, Heavy Load Handling Inside Containment, Rev. 3
[67] USNRC, "Control of Heavy Loads at Nuclear Power Plants," NUREG-0612, July 1980 [Available at nrc.gov]
[68] (1986-05-15) GPU Safety Evaluation, Defueling SER, Rev. 10

[69] (1985-05-02) NRC Safety Evaluation, Heavy Load Handling Over the TMI-2 Reactor Vessel (re 04-19-1985)
[70] (1985-04-19) GPU Safety Evaluation, Heavy Load Handling Over Reactor Vessel SER
[71] (1985-04-19) GPU Safety Evaluation, Heavy Load Handling Over Reactor Vessel SER
[72] USNRC, "Control of Heavy Loads at Nuclear Power Plants," NUREG-0612, July 1980 [Available at nrc.gov]
[73] (1985-05-02) NRC Safety Evaluation, Heavy Load Handling Over the TMI-2 Reactor Vessel (re 04-19-1985)
[74] (1984-06-18) GPU Safety Evaluation, Preparatory Activities for Plenum Assembly Removal
[75] (1984-03-09) GPU Safety Evaluation, Head Removal, Rev. 5
[76] (1984-07-31) NRC Review, Preparatory Activities for Plenum Assembly Removal SER (re 06-18-1984)
[77] (1984-09-14) NRC Safety Evaluation, Plenum Assembly Removal (re various letters)
[78] (1982-07-13) NRC Review, Control Rod Drive Mechanism Quick Look Camera Inspection (re 07-06-1982)
[79] (1983-07-13) NRC Review, Reactor Vessel Underhead Characterization (re Various Letters)
[80] (1985-05-01) GPU Safety Evaluation, Plenum Removal and Transfer, Rev. 3
[81] (1984-03-09) GPU Safety Evaluation, Head Removal, Rev. 5
[82] USNRC, "Control of Heavy Loads at Nuclear Power Plants," NUREG-0612, July 1980 [Available at nrc.gov]
[83] (1985-05-07) NRC Safety Evaluation, Lifting, Transfer, and Storage of Plenum Assembly (re various letters)
[84] (1984-03-09) GPU Safety Evaluation, Head Removal, Rev. 5
[85] (1983-02-25) GPU Safety Evaluation, Makeup and Purification Demineralizer Resin Sampling
[86] (1984-07-19) GPU Safety Evaluation, Makeup and Purification Cesium Elution SER
[87] USNRC, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants," Regulatory Guide 1.143, Rev. 1, October 1979 [Current and past regulatory guides are available at nrc.gov]
[88] American Society of Mechanical Engineers, B31.1, "Power Piping," New York, NY [Revision not provided]
[89] (1984-07-19) GPU Safety Evaluation, Makeup and Purification Cesium Elution SER
[90] (1984-07-19) GPU Safety Evaluation, Makeup and Purification Cesium Elution SER
[91] (1984-07-19) GPU Technical Evaluation, SDS
[92] (1993-11-12) GPU, Proposed POL Submittals-TMI Offsite Dose Calculation Manual [Note: This version is Revision 7, dated November 1993, provided as an example.]
[93] Oak Ridge National Laboratory, "Calculation of Doses Due to Accidentally Released Plutonium from an LMFBR," ORNL-NSIC-74, Oak Ridge, TN, November 1972
[94] (1984-09-19) NRC Safety Evaluation, Makeup and Purification Demineralizer Cesium Elution (re 07-19, 08-29-1984)
[95] (1984-05-31) GPU Safety Evaluation, Internals Indexing Fixture Processing System, Rev. 1
[96] American Society of Mechanical Engineers, B31.1, "Power Piping," New York, NY [Revision not provided]
[97] (1984-07-24) NRC Safety Evaluation, Internals Index Fixture Water Processing System (re 05-31, 07-16-1984)
[98] (1989-05-12) GPU Technical Evaluation, Defueling Water Cleanup System, Rev. 12
[99] (1989-05-12) GPU Technical Evaluation, Defueling Water Cleanup System, Rev. 12
[100] (1987-09-28) GPU Technical Evaluation, SDS, Rev. 5
[101] (1989-05-12) GPU Technical Evaluation, Defueling Water Cleanup System, Rev. 12

[102] GPU Nuclear Corporation, "Control of Heavy Loads," GPU Letter LL2-81-0277 to NRC, September 30, 1981 [ADAMS Accession No. ML20010J673]
[103] (1986-06-02) GPU Safety Evaluation, Heavy Load Handling Inside Containment, Rev. 3
[104] (1985-08-16) GPU Technical Evaluation, SDS, Rev. 3
[105] (1986-05-15) GPU Safety Evaluation, Defueling SER, Rev. 10
[106] (1985-08-06) NRC Safety Evaluation, Defueling Water Cleanup System (re various letters)
[107] (1986-02-06) GPU Safety Evaluation, TMI-2 Temporary Reactor Vessel Water Filtration System
[108] (1985-10-10) GPU Safety Evaluation, Early Defueling of the TMI-2 Reactor Vessel, Rev. 4
[109] (1986-02-07) NRC Safety Evaluation, Temporary Reactor Vessel Water Filtration System
[110] (1986-03-18) NRC Review, Restricted Operations of Temporary Vessel Filtration System (re various letters)
[111] (1986-04-18) NRC Review, Temporary Reactor Vessel Filtration System, Rev. 2 (re various letters)
[112] (1986-06-30) GPU Safety Evaluation, Operation of Filter-Aid Feed System
[113] (1986-07-09) NRC Review, Operation of Filter-Aid Feed System, Use of Diatomaceous Earth (re various letter)
[114] (1985-11-05) NRC Safety Evaluation, Defueling Canisters (re various letters)
[115] (1985-08-06) NRC Safety Evaluation, Defueling Water Cleanup System (re various letters)
[116] (1986-12-15) GPU Safety Evaluation, Addition of Coagulants to the Reactor Coolant System, Rev. 0
[117] (1986-12-31) GPU, Use of Different Coagulants (re 12-15-1986)
[118] (1986-02-07) GPU Technical Evaluation, Defueling Water Cleanup System
[119] (1987-03-27) NRC Review, Criticality Safety Evaluation for Coagulants (re 12-11-1986, 02-20, 01-05-1987)
[120] (1986-05-05) GPU Safety Evaluation, Addition of a Biocide to the TMI-2 Reactor Coolant System, Rev. 0
[121] (1986-05-06) NRC Review, Addition of a Biocide to the TMI-2 Reactor Coolant System (re 05-05-1986)
[122] (1986-04-11) GPU Safety Evaluation, Replacement of Loaded Fuel Canister Head Gaskets
[123] (1986-05-15) GPU Safety Evaluation, Defueling SER, Rev. 10
[124] (1986-04-29) NRC Review, Replacement of Loaded Fuel Canister Head Gaskets (re 04-11-1986)
[125] (1986-01-17) GPU, Plan to Use Modified Debris Canisters for removing End Fittings, etc
[126] (1986-01-21) NRC Review, Plan to Use Modified Fuel Canisters to Remove End Fittings, etc (re 01-17-1986)
[127] (1987-04-02) GPU, Safety Evaluation, Canister Handling and Preparation for Shipment SER, Rev. 4 (Effective Pages Only)
[128] (1985-04-16) GPU Safety Evaluation, Justifying Non-Seismic Design of TMI-2 Post-Accident Systems, Rev. 0
[129] (1987-04-02) GPU, Safety Evaluation, Canister Handling and Preparation for Shipment SER, Rev. 4 (Effective Pages Only)
[130] (1986-06-20) NRC Safety Evaluation, Canister Handling and Preparation for Shipment Program (re 02-17, 06-11-1986)
[131] (1985-11-05) NRC Safety Evaluation, Defueling Canisters (re various letters)
[132] (1987-04-02) GPU, Safety Evaluation, Canister Handling and Preparation for Shipment SER, Rev. 4 (Effective Pages Only)
[133] (1987-05-26) GPU Safety Evaluation, Canister Handling and Preparation for Shipment, Rev. 5 (Effective Pages Only)

[134] (1986-06-20) NRC Safety Evaluation, Canister Handling and Preparation for Shipment Program (re 02-17, 06-11-1986)
[135] (1988-03-03) NRC Review, Canister Handling and Preparation for Shipment Program TER (re various letters)
[136] (1986-06-02) GPU, Testing of Core Region Defueling Techniques
[137] (1985-10-10) GPU Safety Evaluation, Early Defueling of the TMI-2 Reactor Vessel, Rev. 4
[138] (1986-01-22) GPU, Plan to Use Hydraulic Impact Chisel to Separate Fused Material
[139] (1986-06-04) NRC Safety Evaluation, Heavy Duty Tong Tool, Spade Bucket, Hydraulic Impact Chisel (re various letters)
[140] (1985-11-12) NRC Safety Evaluation, Early Defueling of the TMI-2 Reactor Vessel (10-10-1985)
[141] (1986-08-27) GPU, Plan to Use Plasma Arc Torch to Cut Upper End Fittings (re 08-18-1986)
[142] (1987-08-20) NRC Review, Use of Plasma Arc Torch to Cut Upper End Fittings (re various letters)
[143] (1987-12-28) GPU, Use of Core Bore Machine for Dismantling the Lower Core Support Assembly
[144] (1986-06-11) GPU Safety Evaluation, Core Stratification Sample Acquisition, Rev. 4
[145] (1988-01-08) NRC Review, Use of Core Bore Machine for Dismantling Lower Core Support Assembly (re 12-03, 12-28-1987)
[146] (1989-01-10) GPU Safety Evaluation, Sediment Transfer and Processing Operations, Rev. 4
[147] USNRC, "Age-Specific Radiation Dose Commitment Factors for a One-Year Chronic Intake," NUREG-0172, November 1977 [Available at nrc.gov]
[148] (1986-09-25) NRC Review, Sediment Transfer and Processing Operations (re 03-18-1985)
[149] (1986-03-14) GPU Safety Evaluation, Decontamination Using Ultrahigh Pressure Water Flush, Rev. 0
[150] (1986-03-28) GPU Safety Evaluation, Reactor Building Decontamination and Dose Reduction Activities for 1986, Rev. 0
[151] (1986-03-28) GPU Safety Evaluation, Reactor Building Decontamination and Dose Reduction Activities for 1986, Rev. 0
[152] (1986-07-16) NRC Safety Evaluation, Decontamination using Ultrahigh Pressure Water Flush (re 03-14-1986)
[153] (1985-10-24) GPU, Commencement of Preliminary Defueling Operations
[154] (1985-10-29) NRC Safety Evaluation, Commencement of TMI-2 Preliminary Defueling Operations (re 10-24-1985)
[155] (1985-09-27) GPU Hazards Analysis, Potential for Boron Dilution of Reactor Coolant System, Rev. 2
[156] (1985-10-10) GPU Safety Evaluation, Early Defueling of the TMI-2 Reactor Vessel, Rev. 4
[157] (1986-05-15) GPU Safety Evaluation, Defueling SER, Rev. 10
[158] (1986-05-20) GPU Safety Evaluation, Defueling SER, Rev. 10 (Correction)
[159] (1985-10-10) GPU Safety Evaluation, Early Defueling of the TMI-2 Reactor Vessel, Rev. 4
[160] (1986-05-15) GPU Safety Evaluation, Defueling SER, Rev. 10
[161] (1986-05-20) GPU Safety Evaluation, Defueling SER, Rev. 10 (Correction)
[162] (1985-11-12) NRC Safety Evaluation, Early Defueling of the TMI-2 Reactor Vessel (10-10-1985)
[163] (1986-08-16) GPU, Storage of Upper End Fittings in Shielded 55 Gal Drums
[164] (1986-09-10) NRC Review, Plans to Store Upper End Fittings (re various letters)
[165] (1986-09-25) NRC Safety Evaluation, Storage of Upper End Fittings in an Array of 55 Gallon Drums (re various letters)
[166] (1985-10-10) GPU Safety Evaluation, Early Defueling of the TMI-2 Reactor Vessel, Rev. 4

[167] ORNL-TM-8730, Estimated Amount of Kr-85 Available for Release from Intact Fuel Rods in TMI-2 (1983-10)
[168] (1993-11-12) GPU, Proposed POL Submittals-TMI Offsite Dose Calculation Manual [Note: This version is Revision 7, dated November 1993, provided as an example.]
[169] (1985-10-10) GPU Safety Evaluation, Early Defueling of the TMI-2 Reactor Vessel, Rev. 4
[170] (1985-05-24) BAW, TMI-2 Defueling Canisters Final Design Technical Report
[171] USNRC, "Aerosols Generated by Free Fall Spills of Powders and Solutions in Static Air," NUREG/CR-2139, December 1981 [Available at osti.gov]
[172] Hishima, J., "Potential Aerosol Generation Mechanisms from Damaged Shipping Packages," BNWL-SA-5742, July 1976 [Available at osti.gov]
[173] USNRC, "Age-Specific Radiation Dose Commitment Factors for a One-Year Chronic Intake," NUREG-0172, November 1977 [Available at nrc.gov, also see Errata]
[174] GEND-INF-019-VOL-2, ESTIMATED SOURCE TERMS DURING TMI-2 DEFUELING OPERATIONS PHASE II (1983-05)
[175] (1986-07-24) NRC Review, Core Region Bulk Defueling Operations Limited to the Core Region (re 05-15, 07-23-1986)
[176] (1985-11-12) NRC Safety Evaluation, Early Defueling of the TMI-2 Reactor Vessel (10-10-1985)
[177] (1988-01-18) GPU Safety Evaluation, Lower Core Support Assembly Defueling, Rev. 2
[178] (1986-05-15) GPU Safety Evaluation, Defueling SER, Rev. 10
[179] (1988-04-01) NRC Safety Evaluation, Lower Core Support Assembly Defueling (re 01-18-1988)
[180] (1988-06-06) GPU Safety Evaluation, Completion of Lower Core Support and Lower Head Defueling, Rev. 0
[181] (1986-05-15) GPU Safety Evaluation, Defueling SER, Rev. 10
[182] (1988-12-01) NRC Safety Evaluation, Lower Core Support and Lower Head Defueling (re 06-06-1988)
[183] (1988-09-15) GPU Safety Evaluation, Completion of Upper Support Core Assembly Defueling, Rev. 0
[184] (1986-05-15) GPU Safety Evaluation, Defueling SER, Rev. 10
[185] (1989-04-04) NRC Safety Evaluation, Completion of Upper Core Support Assembly Defueling (re 9-15-1988)
[186] NUREG-0591, Environmental Assessment, Use of EPICOR-II at TMI-2 (1979-08)
[187] (1981-03-11) GPU Technical Evaluation, SDS, Revised
[188] USNRC, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors (PWR-GALE-Code)," NUREG-0017, April 1976 [Available at nrc.gov]
[189] (1981-03-11) GPU Technical Evaluation, SDS, Revised
[190] *U.S. Code of Federal Regulations*, "Concentrations in Air and Water Above Natural Background," Appendix B, Part 20, "Standards for Protection Against Radiation," Chapter I, Title 10, "Energy," 1981
[191] NUREG-0796, Operation of the Submerged Demineralizer System at TMI-2 (1981-06)
[192] USNRC, "Three Mile Island Nuclear Station Unit 2 Environmental Technical Specifications," Appendix B to Operating License, February 8, 1978 [ADAMS Accession No. ML20076K096]
[193] NUREG-0683, Vol. 2, PEIS-Decontamination and Disposal of Radioactive Wastes Resulting from TMI-2 (1981-03)
[194] NUREG-0796, Operation of the Submerged Demineralizer System at TMI-2 (1981-06)
[195] NUREG-0796, Operation of the Submerged Demineralizer System at TMI-2 (1981-06)

[196] NUREG-0796, Operation of the Submerged Demineralizer System at TMI-2 (1981-06)
[197] (1981-03-11) GPU Technical Evaluation, SDS, Revised
[198] (1981-05-11) GPU System Description, SDS, Rev. 2
[199] (1982-10-13) GPU Safety Evaluation, SDS Liner Recombiner and Vacuum Outgassing System
[200] (1982-11-11) GPU Response to NRC (Discussions), SDS Liner Recombiner and Vacuum Outgassing System
[201] (1982-11-23) NRC Review, SDS Liner Recombiner and Vacuum Outgassing System
[202] (1981-03-11) GPU Technical Evaluation, SDS, Revised
[203] NUREG-0796, Operation of the Submerged Demineralizer System at TMI-2 (1981-06)
[204] USNRC, “Three Mile Island Accident of 1979 Knowledge Management Digest: Recovery and Cleanup,” NUREG/KM-0001, Supplement 1, June 2016 [Available at nrc.gov]

11 OTHER SAFETY EVALUATIONS

TABLE OF CONTENTS

Note: "NA" (not applicable) means the licensee and the NRC determined that the safety topic was not important for the activity.

11.1 Introduction

11.1.1 Background

This chapter includes the evaluations of the following safety considerations: (●) fire protection; (●) industrial safety; (●) instrument interference; (●) seismic hazard; (●) Unit 1 impact; and (●) vital equipment protection. These safety considerations were less frequent in the evaluations of most cleanup activities.

- ***Fire Protection***. The TMI-2 cleanup did not present any unique concerns for fire protection safety. Safety evaluations of cleanup activities evaluated the adequacy of fire protection measures, such as the availability of fire extinguishing equipment and systems, mechanisms to detect smoke and fire, and controls of combustible loading inside the containment building. The licensee was required to comply with fire protection requirements and standards unless an exemption was approved by the NRC. The NRC approved exemptions for some requirements of 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," Appendix R, [1] "Fire Protection Program for Nuclear Power Facilities," given that the facility was no longer in operation.

- ***Industrial Safety***. Industrial hazards that were inherent in the planned operations during the cleanup at TMI-2 generally included: (●) heat stress; (●) fall hazards; (●) electrical shock; (●) high-pressure water sprays; (●) noise; (●) eye hazards; (●) tripping hazards; (●) rotating equipment; (●) sharp equipment and objects; and (●) materials lifting and handling. Safety requirements to reduce the risk of these hazards included written procedures, personnel training, use of safety equipment or protective clothing, and engineered accident prevention measures.

Editor's Note: Refer to GEND-034, "Gross Decontamination Experiment Report," [2] for details of the licensee's industrial safety program during the 1982 period. Appendix D to that report described the safety issues affecting the decontamination experiment. Issues included: (●) heat stress; (●) fall protection; (●) belts, ladders, and scaffolding; (●) mobile work platform; (●) polar crane safety; (●) protective devices for decontamination; (●) confined space requirements; (●) equipment movement; and (●) fire protection. The training lesson plan for general containment building entry was also attached.

- ***Instrument Interference***. Instrument interference was generally a concern with activities that used a plasma arc cutting torch. A high-frequency generator was used to establish and stabilize the arc between the electrode and the work piece. The frequencies encountered had the capability to disrupt instruments inside the containment building, the most crucial being the criticality monitors.

- ***Seismic Hazard***. Equipment that was used or staged in the truck bay included storage stands associated with the transfer and offsite shipment of defueling canisters. This equipment was designed so that a design-basis seismic event would not cause the equipment to fail or collapse in such a way as to cause damage to Three Mile Island Unit 1 (TMI-1) safe-shutdown equipment or systems. Canister and cask handling activities in the truck bay area were evaluated to estimate the probability of the failure of equipment and structures used to perform the activity concurrent with the postulated seismic event. When the probability, consequences, or both were deemed acceptable, no seismic analysis was performed. The licensee was required to comply with seismic design requirements and standards throughout the plant unless the NRC approved an exemption. The NRC approved exemptions for some requirements of 10 CFR Part 50, Appendix A, [(3)] "General Design Criteria for Nuclear Power Plants," given that the facility was no longer in operation.

- ***Unit 1 Impact***. The Unit 1 and Unit 2 fuel handling buildings shared a common area in the truck bay. Activities that could potentially impact TMI-1 operations and safe-shutdown equipment included the loading of the defueling canisters into the shipping cask on a rail car in the truck bay. The NRC's Atomic Safety and Licensing Board imposed a condition on TMI-1 restart. This condition required that the effects of TMI-2 defueling canister movement on TMI-1 personnel in the fuel handling building be addressed by NRC-approved procedures, or work in the Unit 1 area of the fuel handling building be suspended during Unit 2 defueling canister movement. The concerns during normal operations and accident conditions that were considered included: (●) liquid release; (●) airborne release; and (●) direct radiation.

- ***Vital Equipment Protection***. The vital components were generally defined as: (●) necessary to protect the integrity of the reactor coolant system; (●) required to maintain and monitor the boron concentration in the reactor coolant system; (●) required to prevent unacceptable offsite releases; and (●) required to be operable by the recovery technical specifications. Cleanup activities that had the greatest potential to damage vital equipment generally included the handling of heavy loads that could result in accident load drops and the use of abrasive tools for decontamination inside the containment building. All activities that had the potential for load drops were evaluated under the guidance of NUREG-0612, [(4)] "Control of Heavy Loads at Nuclear Power Plants."

11.1.2 Chapter Contents

This chapter presents various other safety concerns of the postaccident TMI-2 and defueling operations. It describes the many studies performed to give the reader an understanding of the thinking of the analysts at the time, the expectations and the reality, the uncertainties in the data, and the measurement and mitigation methods. It also presents a high-level timeline of cleanup activities.

The evaluations presented in this chapter ensured that: (●) all activities that could result in plant damage and harm to workers, the public, and environment were addressed and consequences evaluated; (●) controls were maintained in accordance with the requirements of the plant's license, technical specifications, procedures, and applicable regulatory requirements; and (●) adequate contingencies were developed for normal and accident conditions.

Section 2 summarizes a key study that was used to support safety evaluations to protect vital equipment from load drops. The following sections present the safety evaluation for each applicable cleanup activity or system. Section 8 lists endnotes for references cited throughout this chapter.

11.2 Key Studies

11.2.1 Control of Heavy Loads at Nuclear Power Plants
(USNRC, NUREG-0612, July 1980)

This report [5, a] summarized work performed by the NRC in the resolution of Generic Technical Activity A-36, "Control of Heavy Loads Near Spent Fuel." This technical activity [b] was one of the subjects designated as "unresolved safety issues" pursuant to Section 210 of the Energy Reorganization Act of 1974. The report described the technical studies and evaluations performed by the NRC, the NRC's guidelines based on these studies, and the NRC's plans for implementing its technical guidelines.

Section 2 of the report discussed the potential for an accidental load drop that could impact nuclear fuel or safety-related equipment, with the potential for: (●) excessive offsite releases; (●) inadvertent criticality; (●) loss of water inventory in the reactor (or spent fuel pool); or (●) loss of safe-shutdown equipment. The licensee's safety evaluation report included an assessment of impacts due to a heavy load drop on required safe-shutdown equipment. Criterion IV of NUREG-0612 defined "required safe-shutdown functions" as follows: (●) maintain the reactor coolant pressure boundary; (●) reach and maintain subcriticality; (●) remove decay heat; and (●) maintain the integrity of components whose failures could result in excessive offsite releases. Criterion IV stated that damage to equipment, in redundant or dual safe-shutdown paths, based on calculations assuming the accidental drop of a postulated heavy load, would be limited so as not to result in the loss of required safe-shutdown functions.

The required safe-shutdown functions that applied to TMI-2 in the cooling mode and core configuration at the time during cleanup included: (●) capability to maintain subcriticality; (●) decay heat removal; and (●) capability to maintain the integrity of components whose failures could result in excessive offsite releases. The reactor coolant pressure boundary needed to be maintained only as part of maintaining the reactor coolant for decay heat removal and reactivity control in the reactor coolant system.

The safety evaluation included the identification of loads, targets, and load/target interactions, along with the evaluation of the worst credible consequence (refer to NUREG/KM Chapter 7 on

---

[a] Editor's Note: In October 2005, the NRC issued Regulatory Issue Summary (RIS) 2005-25, "Clarification of NRC Guidelines for Control of Heavy Loads," as a result of recommendations developed through Generic Issue 186, "Potential Risk and Consequence of Heavy Load Drops in Nuclear Power Plants," and findings from the NRC inspection program. This RIS reemphasized the need to follow NUREG-0612 guidance, which addressed good practices for crane operations and load movements. Attachment 1 to this RIS described the application of insights gained from operating experience and inspection to the guidelines of NUREG-0612. The attachment also clarified the guidelines where operating experience or inspection results indicated further explanation was necessary.

[b] Editor's Note: Generic Technical Activity A-36 is described in NUREG-0933, "Resolution of Generic Safety Issues" (available at nrc.gov).

load drop evaluations). The evaluation showed that Criterion IV for damage to equipment was met.

11.2.2 Safety Evaluation Justifying the Nonseismic Design of TMI-2 Postaccident Systems

(GPU Nuclear, April 16, 1985)

- ***Purpose and Scope***. The purpose of the licensee's safety evaluation [6] was to justify the nonseismic design of systems that were installed after the accident. The scope of the evaluation covered all operational postaccident systems and systems that planned to be operational to support fuel removal from the reactor vessel. The evaluation demonstrated that the public consequences of radionuclide releases that could occur because of system failures due to a design-basis earthquake were less than the limits of applicable NRC standards.

- ***Summary***. This report is summarized below.

  - *Analytical Approach*. The evaluation considered the consequences of postulated accidents due to a seismic event. The evaluation was conducted in the following three phases: (1) identification of source terms and driving forces, (2) specification of seismic-induced accident sequences, and (3) consequence analysis.

  - *Driving Forces*. The evaluation considered the following potential driving forces that could develop from a seismic event or could credibly occur coincidentally with the event: (●) air pressure differential of 1 pound per square inch between the containment building and the environment that corresponded to a low-pressure weather front; (●) combustibles ignition from shorting nonseismically qualified electrical components or other unspecified ignition source; (●) fire resulting from a pyrophoric reaction of metal fines associated with the core debris or fines settled on reactor vessel internals; and (●) steaming of the reactor coolant system caused by criticality from core reconfiguration, boron dilution, or a vessel draindown (caused by leakage through the in-core instrumentation guide tubes).

  - *Source Terms*. The evaluation considered the following potential radionuclide source terms that could develop from a seismic event: (●) core exposed to the containment atmosphere after reactor vessel draindown; (●) increased core radionuclide inventory due to criticality from a boron dilution and core configuration; (●) increased concentrations of a select number of isotopes in the reactor coolant system because of seismic-induced disturbances; (●) smearable contamination on containment building walls and tooling; (●) increased source term in defueling canisters due to a criticality; (●) existing fuel and fission product inventory in defueling canisters; (●) sumps in the reactor and auxiliary building; (●) submerged demineralizer system liners, filters, and input process lines (e.g., reactor coolant system, makeup and purification system); (●) makeup and purification demineralizer tanks, holdup tanks, and associated piping (latter two components nonseismic qualified) during cleanup elution and resin sluicing processes; (●) waste storage (containment building, auxiliary and fuel handling building ventilation filters, and respirator and laundry facility); (●) EPICOR II system tanks, filters, and process piping; (●) liquid, solid, and gas waste disposal systems; (●) defueling water cleanup system ion exchangers, filters, and process piping; (●) internals indexing fixture water processing system; (●) interim solid waste staging facility for the storage of dry wastes and dewatered resin liners; (●) solid waste storage building for the storage of low-level dry wastes; (●) solid waste staging facility for the temporary storage of dry wastes before shipping; and (●) processed water storage and recycle system including two 500,000-gallon processed water storage tanks.

- *Accident Sequences*. The safety evaluation considered over 32 potential seismically induced accident sequences, which were listed in Table 3.1 of the report. Seismically induced failure mechanisms considered in the accident sequence formulation included, but were not limited to, (●) load drop; (●) disruption of loose contamination; (●) piping failure; (●) tank leaks; (●) fire; (●) resin liner rupture; and (●) structure collapse.

- *Consequences*. The consequences of the potential accident sequences were estimated using the methods and dose conversion factors specified in Regulatory Guide 1.109, "Calculation of Annual Dose to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I" (7) ; NUREG-0172, "Age-Specific Radiation Dose Commitment Factors for a One-Year Chronic Intake," issued November 1977 (8) ; and Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors." [c] Table 3.1 of the report summarized and tabulated the consequences of potential accident sequences associated with a seismic event.

- *Assumptions*. The major assumptions used in the consequence calculation included: (●) 2-hour doses at the site boundary for potential airborne releases; (●) organ dose conversion factors in Regulatory Guide 1.109 and NUREG-0172; (●) insights into the significance of the calculated bone doses from NUREG/CR-3535, (9) "Age-Dependent Dose-Conversion Factors for Selected Bone-Seeking Radionuclides," issued May 1984, and International Commission on Radiological Protection Publication 26, (10) "Recommendations of the International Commission on Radiological Protection," issued 1977; (●) unfiltered releases to the environment; (●) airborne liquid aerosol release fraction of $1x10^{-3}$ for pressurized liquid sprays, $1x10^{-4}$ for nonpressurized liquid spills, and $3x10^{-5}$ for seismic-induced wave action on standing liquid pools; (●) airborne particulate release fraction of $1x10^{-3}$ for particulate material exposed to air; (●) airborne release fraction of $1x10^{-4}$ for particulates contained on resins and ruptured or crushed filters; (●) airborne release fraction of $1x10^{-3}$ for fires; (●) less radiological significance for particles larger than 10 micrometers; (●) fraction of $1x10^{-2}$ for particles in the 10-micrometer range of material in the fuel and knockout canisters (based on actual particle size characterization); (●) particle sizes less than 10 micrometers for all of the material collected in filter canisters; (●) 10 percent of the

[c] Editor's Note: Regulatory Guide (RG) 1.4, was withdrawn in December 2016. The guidance in RG 1.4 was updated and incorporated into RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," and RG 1.195, "Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors." The information in RG 1.183 provided guidance for new and existing light-water reactor plants that have adopted the alternative source term, and RG 1.195 provided guidance for those light-water reactor plants that have not adopted the alternative source term.

total core mass is particulate and resides on an exposed surface in the event of a vessel draindown; (●) total material release fractions of $1x10^{-5}$ and $1x10^{-4}$ for postulated accidents involving a direct airborne release from a fuel/knockout and filter canisters, respectively (based on the fraction that becomes airborne and the distribution of particulate sizes); and (●) a release fraction of $1x10^{-6}$ for an exposed core (based on the amount of exposed particulate core material, the distribution of particulate sizes, and the fraction that becomes airborne).

- ***Conclusion***. Offsite consequences were calculated for a spectrum of accident sequences that could result from a design-basis earthquake. The report stated that these calculations were based on conservative assumptions. The consequences were modeled in terms of a 2-hour dose at the site boundary, as specified in 10 CFR Part 100, "Reactor Site Criteria," (11) which set a limit 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 actual radionuclide inventory at the time of the evaluation, doses were calculated to the critical organ (the adolescent bone). Doses to the critical organ did not exceed the critical organ limitation for thyroid in 10 CFR Part 100. Insights from the literature indicated that the 10 CFR Part 100 organ dose limitation was an appropriate limit for the critical organ for postulated TMI-2 accidents.

More restrictive criteria for certain accident sequences were promulgated in Chapter 15, "Transient and Accident Analyses," (12) of the NRC Standard Review Plan. Specifically, the consequences of some accident sequences must be "well within" (less than 25 percent) or a "small fraction" (less than 10 percent) of 10 CFR Part 100 limits. The analysis indicated that there was no accident sequence that exceeded these more restrictive guidelines.

11.3 Data Collection Activities

11.3.1 Axial Power Shaping Rod Insertion Test

- ***Purpose***. To individually move the eight axial power shaping rod assemblies in the core to provide insight into the extent of core and upper plenum damage. This early insight allowed time to factor this information into plans for subsequent inspections for the head, upper plenum, and core removal.

- ***Evaluation: Fire Protection***. (13) The licensee's safety evaluation noted that for this test, power would be applied to the control drive mechanisms. Video cameras would be positioned to monitor the area over the mechanisms to detect smoke and fire. The test procedure required that these cameras be monitored during the test. This monitor ensured that a fire would be detected immediately. Existing TMI-2 procedures would be adequate to ensure that a fire would be promptly and safely extinguished.

- ***NRC Review***. [14] Editor's Note: The NRC's safety evaluation report did not specifically address this evaluation topic.

11.3.2 Camera Insertion through Reactor Vessel Leadscrew Opening

- ***Purpose***. To insert a camera into the reactor vessel through a control rod drive mechanism leadscrew opening to provide the first visual assessment of conditions inside the reactor vessel. This activity was known as "Quick Look."

- ***Evaluation***: ***Fire Protection***. [15] The licensee's safety evaluation stated that fire protection for the Quick Look program was provided in accordance with the requirements of the TMI-2 fire hazards analysis report and TMI-2 administrative procedure, "Control of Combustible Materials." The estimated increase in combustible loading in the containment building for Quick Look was 0.022 pounds per square foot (wood equivalent). Based on this small increase in combustible loading, additional fire protection was not required for the Quick Look activities. At the conclusion of Quick Look, all Quick Look materials that were not removed from the containment building would be factored into the TMI-2 fire hazards analysis.

- ***NRC Review***. [16] Editor's Note: The NRC's safety evaluation report did not specifically address this evaluation topic.

11.3.3 Reactor Vessel Underhead Characterization (Radiation Characterization) (NA)

11.3.4 Reactor Vessel Underhead Characterization (Core Sampling) (NA)

11.3.5 Core Stratification Sample Acquisition (NA)

11.3.6 Use of Metal Disintegration Machining System to Cut Reactor Vessel Lower Head Wall Samples

- ***Purpose***. To remove metallurgical samples from the inner surface of the lower reactor vessel head using the metal disintegration machining (MDM) system. An international research program used these samples to study core melt and reactor vessel interactions. This program was conducted after the reactor vessel was defueled.

- ***Evaluation: Fire Protection***. [17, 18] The licensee's safety evaluation stated that its safety evaluation report [19] for bulk defueling bounded fire concerns during sampling activities. Fire protection was provided in accordance with the licensee's fire protection program evaluation and the TMI-2 administrative procedure. Existing fire protection equipment would be available during the sampling process.

- ***Evaluation: Instrument Interference***. (20, 21) The licensee's safety evaluation stated that the plasma arc torch operation involved higher power levels than would be experienced during MDM cutting (i.e., the plasma arc used cutting power of 200 volts direct current (vdc), 900 amps, and MDM uses 21 vdc, 100 amps). Since no instrument interference was experienced during plasma arc cutting, the evaluation concluded that MDM cutting would not result in instrument interference.

- ***NRC Review: Fire Protection***. (22) The NRC's safety evaluation stated that issues related to pyrophoricity, fire protection, decay heat, and release of radioactivity fell within the bounds of previous NRC safety analysis reports (23, 24, 25) for defueling activities.

11.4 Pre-Defueling Preparations

11.4.1 Containment Building Decontamination and Dose Reduction Activities

Purpose. To conduct decontamination and dose reduction activities in the containment building at elevation levels 305 feet (entry level) and above. Planned activities included: (●) flushing containment building surfaces with deborated water; (●) scrubbing selected components of the polar crane; (●) conducting hands-on decontamination of vertical surfaces; (●) decontaminating the air coolers; (●) flushing the elevator pit; (●) cleaning floor drains; (●) shielding the seismic gap and penetration; (●) decontaminating and shielding hot spots; (●) decontaminating missile shielding; (●) shielding reactor vessel head surface structure; (●) removing concrete and paint; (●) decontaminating cable trays; (●) decontaminating equipment; (●) remote flushing the containment building basement; and (●) testing remote decontamination technology.

- ***Evaluation: Fire Protection***. (26, 27) The licensee's safety evaluation considered precautions to reduce the likelihood of fires and the requirements for ensuring the availability of fire detectors.

  - *Precautions*. To reduce the likelihood of a fire in the containment building during decontamination, the following precautions were taken: (●) Transient combustible material was kept to a minimum in the containment building. (●) All activities that increased the likelihood of a fire, such as welding, burning, or grinding, were reviewed and controlled in accordance with plant procedures. (●) The licensee reviewed all modifications before implementation. (●) The fire detection system was returned to operation. (●) All personnel were equipped with small flashlights for emergency lighting. (●) Both airlocks were available for ingress and egress with the No. 2 airlock being the normal path.

  - *Fire Detector Requirements*. Fire protection in the containment building was provided by fire detectors placed around the tops of the D-rings [d] and in the outlet ducting of the air coolers. Decontamination activities that could interfere with the normal function of the detectors included water spraying in the D-rings and isolation of containment building areas, along

[d] D-rings were the shield enclosures around the steam generator compartments; they were so named because of their shape.

with modifications in the containment building ventilation systems. Requirements involving these activities included the following: (●) During water spraying activities, the fire detectors would normally be disconnected and covered for protection. A fire watch was required when the detector system was disabled. These requirements would be implemented during water spraying activities in the D-rings. (●) Isolation of the containment building areas and modification of the ventilation system required an assessment of the adequacy of the current duct-mounted fire detectors. Additional general area detectors would be provided to meet fire detection requirements within isolated areas in the containment building.

- ***Evaluation: Industrial Safety***. [28] The licensee's safety evaluation stated that during the performance of decontamination and dose reduction activities, personnel health and safety hazards would be reduced ALARA. Certain industrial hazards inherent in the planned operations included: (●) heat stress; (●) fall hazards; (●) electrical shock; (●) high-pressure water sprays; (●) noise; (●) eye hazards; (●) tripping hazards; (●) rotating equipment; (●) sharp equipment and objects; (●) and materials lifting and handling. Safety requirements to reduce the risk of these hazards included the following:

  - *Procedures, Training, and Safety Equipment*. Written procedures, personnel training, and the use of safety equipment would be used to minimize the risk from these hazards.

  - *Heat Stress Prevention*. Heat stress problems were reduced with the installation and operation of the containment building chilled water system. During the summer, the chiller could maintain the ambient containment building temperature at 65 degrees Fahrenheit. This modification enhanced worker comfort and improved worker productivity. Heat stress protective gear could be used, if considered necessary. This included vortex cooling suits and ice vests. The licensee determined when heat protection gear was required. This decision was based on factors such as ambient temperature, the type of work to be done, work duration, and the type of radiation protective equipment used.

  - *Fall Prevention*. Fall hazards were controlled by the installation of handrails and the use of safety belts and lifelines. The protection of workers from fall hazards, as well as other common industrial hazards listed above, would be ensured by a safety review of work procedures by the onsite safety office. The safety office would determine special protective equipment to mitigate fall hazards.

  - *Protection from High-Pressure Water Spray*. Personnel received extensive training and instruction in the proper use of high-pressure sprays to prevent injury. In addition, the equipment was designed with features that minimized the potential for operator injury. Workers were also provided with personnel protective equipment.

Editor's Note: Refer to GEND-034, "Gross Decontamination Experiment Report," [29] for details of the licensee's industrial safety program during the 1982 period. Appendix D to that report described the safety issues affecting the decontamination experiment. Issues included: (●) heat stress; (●) fall protection; (●) belts, ladders, and scaffolding; (●) mobile work platform; (●) polar crane safety; (●) protective devices for decontamination; (●) confined space requirements;

(●) equipment movement; and (●) fire protection. The training guide for general containment building entry was also attached.

- ***Evaluation: Vital Equipment Protection***. (30) The licensee's safety evaluation stated that some decontamination activities such as high-pressure water flush or abrasive blasting had the potential to damage plant equipment. During decontamination activities, administrative controls, physical barriers, or both would protect vital components that were susceptible to damage. These structures, systems, and components would be identified in implementing procedures. The vital components were defined as those: (●) necessary to protect the integrity of the reactor coolant system; (●) required to maintain and monitor the boron concentration in the reactor coolant system; (●) required to prevent unacceptable offsite releases; and (●) required to be operable by the recovery technical specifications.

- ***NRC Review: Industrial Safety***. (31) The NRC's safety evaluation concluded that worker protection was acceptable, in terms of industrial safety for the proposed decontamination and dose reduction activities. The proposed techniques for removing contamination were similar to those previously used at TMI-2. Past experience in the containment building indicated that the need for protective clothing in conjunction with the use of high-pressure or high-temperature cleaning fluids (e.g., spray jets) and high-reach personnel lifting presented potential industrial hazards to operating personnel.

Worker heat stress was a recurring problem during periods of physical exertion in the containment building. The use of high-temperature flush water exacerbated this problem, and the licensee anticipated the use of ice vests and vortex cooling suits in conjunction with regulated stay times to alleviate this concern. Implementing procedures for these activities would be reviewed thoroughly to minimize safety hazards to workers. Timely completion of the proposed containment building chilled water-cooling system would enhance overall worker comfort and safety. In the interest of industrial safety, the NRC recommended that priority be given to completing this system before the seasonal temperature increase impacted the work environment in the containment building.

- ***NRC Review***. (32) Editor's Note: The NRC's safety evaluation report did not specifically address these evaluation topics.

11.4.2 Reactor Coolant System Refill (NA)

11.4.3 Reactor Vessel Head Removal Operations

11.4.3.1 Polar Crane Load Test

- ***Purpose***. To conduct load testing of the polar crane in preparation for reactor vessel head lift and removal. The polar crane load test required four major sequential activities: (1) relocation of the internals indexing fixture; (2) assembly of test load; (3) load test; and (4) disassembly of test load.

- ***Evaluation: Vital Equipment Protection***. [33] The licensee's safety evaluation considered impacts to required safe-shutdown equipment from a heavy load drop in accordance with the guidance from NUREG-0612. Based on calculations assuming the accidental drop of a postulated load, Criterion IV In NUREG-0612 stated that damage to equipment in redundant or dual safe-shutdown paths should be limited so as not to result in loss of required safe-shutdown functions. Criterion IV defined "safe-shutdown functions" as those required to: (●) maintain the reactor coolant pressure boundary; (●) reach and maintain subcriticality; (●) remove decay heat; and (●) maintain the integrity of components whose failures could result in excessive offsite releases. The safety evaluation included the identification of loads, targets, and load/target interactions, as well as the evaluation of the worst credible consequence (refer to NUREG/KM Chapter 7 on load drop evaluations for details of these evaluations). The evaluation showed that Criterion IV for damage to equipment was met.

  - *TMI-2 Safe-Shutdown Functions*. The required safe-shutdown functions that applied to the TMI-2 reactor in its cooling mode and core configuration at that time included: (●) capability to maintain subcriticality; (●) capability to maintain decay heat removal; and (●) capability to maintain the integrity of components whose failures could result in excessive offsite releases. The reactor coolant pressure boundary needed to be maintained only insofar as reactor coolant must be maintained in the reactor coolant system (RCS) for decay heat removal and reactivity control.

  - *Results*. Required safe-shutdown functions unique to TMI-2 during the load test included the following:

    – *Criticality (Reactor Core)*. Because of the configuration of TMI-2, the only credible mechanism by which criticality control could be compromised was the deboration of water in the RCS. The investigation of systems within the load impact areas that contained unborated water found that they would fail in such a way as to drain their contents onto the containment building floor and not into the RCS. For example, the boron concentration of the RCS could be reduced by a postulated load drop that could cause a gross leakage from the unborated secondary side into the primary side. In addition, damage to the steam generators severe enough to cause such leakage would result in damage to the outer surface of the steam generator allowing the unborated water to drain to the containment sump. Systems for injecting highly borated water into the RCS were available, and it was not feasible that one load drop would reduce the functional capability of these systems to such a point that boron injection could not function.

    – *Criticality (Steam Generator)*. The sequential series of low-probability events required for a potential criticality in the steam generators, each conditioned on the occurrence of all the prior low-probability events, included: (1) A missile shield must be dropped above one of the D-rings. (2) The missile shield must travel far enough into the D-rings to impact a reactor coolant pump or cold-leg piping. Note that there were massive structural beams crossing the D-ring above the reactor coolant pump elevation where the reactor coolant pumps were vertically supported. (3) The missile shield must impact the reactor coolant pump or other structure in such a way as to rupture the pump suction line at a point well below the secondary-side water level. (4) An amount of fuel sufficient to raise criticality concerns would have had to be transferred to the steam generator and the steam generator tubes during the TMI-2 accident. (5) Fuel must be in a high-density close pack configuration within the tubes that would allow criticality, if the borated water were drained from the tubes during the period of time that the fuel was surrounded by unborated secondary water.

– *Criticality (Containment Building Sump)*. The only potential for a criticality in the containment building sump would be from the drop of a heavy load onto systems that could provide a source of unborated water to the sump. To eliminate the potential for a criticality due to a boron dilution accident, water supply to these systems would be isolated while the load test was being performed. (Note: The operable fire protection system was nominally isolated.) An evaluation showed that a significant quantity of unborated water would be required to lower the sump water concentration from current values to a level below 1700 parts per million, which was specified as the reasonable point for sump reactivity problem avoidance.

  Regardless of the amount of unborated water delivered to the sump, two other low-probability conditions could cause sump criticality. The first involved an amount of fuel sufficient to create a critical mass being washed to the sump during the TMI-2 accident. Second, this fuel would have to be in a configuration that could induce criticality if a global boron dilution of the sump water occurred. A qualitative assessment concluded that this issue could be eliminated as a legitimate safety concern. This conclusion was based on the administrative controls that would be in place at the time of the load test, which would limit the amount of unborated water being delivered to the sump, and the low probability of simultaneous occurrence of the initial conditions for fuel deposition in the sump that could lead to a criticality problem.

– *Decay Heat Removal*. Decay heat removal capability was ensured by maintaining water in the reactor vessel. Analysis showed that the water could be drained to the bottom of the cold-leg nozzles with no adverse consequences, such as boiling. In addition, if damage to the RCS that caused leakage of reactor coolant were to occur, makeup capability would exist at least through one loop since a load drop in both D-rings at the same time was not credible by physical separation. Damage to makeup system penetrations would not occur since they were located on the north side of the building away from the load paths. The only way to drain the vessel below this level would be through damage to the in-core instrument tubes.

– *Pressure Boundary Integrity*. The reactor coolant pressure boundary needed to be maintained only insofar as reactor coolant must be maintained in the RCS for decay heat removal and reactivity control. Both evaluations are discussed above.

– *Damaged In-core Instrument Lines*. A special consideration was evaluated regarding the potential consequences of a load drop damaging the in-core instrument lines. The reactor vessel lower head was penetrated by in-core instrument lines. These lines ran from beneath the bottom head of the vessel through a tunnel in the containment building base mat to terminations at the seal table area. The routing of these lines ran parallel to a line drawn between the center of the reactor vessel and the center of the seal table. The width of the area occupied by these lines was just smaller than the diameter of the seal table. The seal area was not located within the load path. A portion of the in-core instrument lines was, however, physically located below the area of the load path for the reactor missile shield blocks. This portion of the lines was separated vertically from load impact surfaces by concrete and steel structures of such massive proportion as to render load penetration incredible.

The only remaining scenario in which a dropped load could damage an in-core instrument line, required this sequence: (1) The missile shield must be dropped into the refueling canal. (2) The shield block must orient and reconfigure itself to fit between the reactor vessel and primary shield wall (present dimensions preclude such an event). (3) The shield block must travel down to the elevation of the reactor vessel skirt, disintegrate into pieces small enough to fit through holes in the reactor vessel skirt, and travel horizontally far enough and with sufficient remaining energy to damage the stainless-steel in-core lines. This scenario was judged not credible based on the improbability of these three events, especially the second, which would violate the physical laws of nature.

– *Offsite Radiological Release*. The containment building pressure boundary prevented an offsite release. The containment building integrity, as required by the recovery technical specifications, was established during the load test. All containment penetrations that could be damaged by a load drop were isolated outside the containment building.

Editor's Note: Refer to NUREG/KM Chapter 7 on load drop evaluations for the complete evaluation of this activity.

- ***NRC Review: Vital Equipment Protection***. [34] The NRC's safety evaluation concluded that since the TMI-2 facility was already in a safe-shutdown condition, there were no concerns related to a potential drop impacting the capability to achieve safe shutdown.

Editor's Note: Refer to NUREG/KM Chapter 7 on load drop evaluations for the complete evaluation of this activity.

11.4.3.2 First-Pass Stud Detensioning for Head Removal (NA)

11.4.3.3 Reactor Vessel Head Removal Operations

- ***Purpose***. To prepare for reactor vessel head removal to perform the lift and transfer of the head and conduct the activities necessary to place and maintain the reactor coolant system (RCS) in a stable configuration for the next phase of the recovery effort.

- ***Evaluation: Vital Equipment Protection***. [35] The licensee's safety evaluation considered impacts to required safe-shutdown equipment due to a heavy load drop in accordance with the guidance from NUREG-0612. Based on calculations assuming the accidental drop of a postulated heavy load, Criterion IV in NUREG-0612 stated that damage to equipment in redundant or dual safe-shutdown paths would be limited so as not to result in loss of required safe-shutdown functions. Criterion IV defined "safe-shutdown functions" as those required to: (●) maintain the reactor coolant pressure boundary; (●) reach and maintain subcriticality; (●) remove decay heat; and (●) maintain the integrity of components whose failures could result in excessive offsite releases. The safety evaluation included the identification of loads, targets, and load/target interactions, as well as the evaluation of the worst credible consequence (refer to NUREG/KM Chapter 7 on load drop evaluations for details of these evaluations). The evaluation showed that Criterion IV for damage to equipment was met.

  - *TMI-2 Safe-Shutdown Functions*. The required safe-shutdown functions applicable to the TMI-2 reactor in its current cooling mode and core configuration included the capability to maintain: (●) subcriticality, (●) decay heat removal, and (●) the integrity of components whose failures could result in excessive offsite releases. The reactor coolant pressure boundary needed to be maintained only insofar as reactor coolant must be maintained in the RCS for decay heat removal and reactivity control.

  - *Results*. Required safe-shutdown functions unique to TMI-2 during reactor vessel head removal included the following:

    – *Subcriticality*. The licensee's safety evaluation addressed the capability to maintain subcriticality in the core in the event of load impact on the reactor. Since a physical redistribution of fuel due to a load drop was not expected to cause criticality, the only credible mechanism by which criticality control could be compromised was the boron dilution of the RCS water. Systems within load impact areas that contained unborated water were investigated and found to fail in such a way as to drain their contents onto the containment building floor and not into the RCS. Further, systems capable of injecting highly borated water into the RCS were available, and it was not feasible for one load drop to reduce the functional capability of these systems to such a point that boron injection could not occur.

    – *Decay Heat Removal*. Decay heat removal capability was ensured by maintaining water in the reactor vessel. Best estimate analysis showed that equilibrium reactor coolant temperature was acceptable with the RCS drained to the bottom of the vessel nozzles. If leakage of reactor coolant were to occur because of damage to the RCS, borated water makeup capability would exist through at least one coolant loop. Damage to makeup system penetrations would not occur since they were located on the north side of the building away from the load paths.

    – *Radiological Release*. The containment building boundary prevented offsite releases. The containment building integrity would be maintained during the actual head lift.

Editor's Note: Refer to NUREG/KM Chapter 7 on load drop evaluations for the complete evaluation of this activity.

- ***Evaluation: Fire Protection***. [36] The licensee's safety evaluation considered the possibility of fire during this activity. The estimated increase in the containment building combustible loading for head removal activities was 0.098 pounds per square foot (wood equivalent) of floor area. Based on this small increase in combustible loading, additional fire protection was not required for head removal activities. At the conclusion of head removal activities, all materials that were not removed from the containment building would be factored into the fire protection program evaluation.

- ***NRC Review: Vital Equipment Protection***. [37] Criterion IV of NUREG-0612 required a matrix table showing all heavy loads and potential areas of impact where safety-related equipment might be damaged. The licensee provided the matrix in its SER. NUREG-0612 also required the identification of the load and impact area combinations that could be eliminated because of separation and redundancy of safety-related equipment. For load/target combinations that could impact safety-related equipment, the licensee was required to state the basis for load drops that would not affect the ability to perform a safety-related function (e.g., reactor shutdown, core decay heat removal, and containment building integrity).

The required safe-shutdown functions that applied to TMI-2 at the time of the head lift activities included: (●) capability to maintain subcriticality, (●) capability to maintain decay heat removal, and (●) capability to maintain the integrity of components whose failure could result in excessive offsite release. The SER discussed the ability to maintain subcriticality and to minimize offsite release. Decay heat removal capability would be maintained because of the head lift height restrictions, the passive loss-to-ambient cooling mode (which depended only on water being retained in the RCS), and the various options available for introducing borated water into the RCS. The NRC concluded that the requirements of NUREG-0612 were satisfied.

Editor's Note: Refer to NUREG/KM Chapter 7 on load drop evaluations for the complete evaluation of this activity.

- ***NRC Review: Fire Protection***. [38] The NRC's safety evaluation considered fire protection during reactor vessel head removal activities.

    - *Combustible Loading*. The licensee's fire protection program evaluation stated that the average combustible loading in the containment building was 1.10 pounds per square foot (equivalent pounds of wood). The licensee stated in its head lift safety evaluation that the head removal activities would increase the average combustible loading by 0.098 pounds per square foot (wood equivalent). This loading amounted to a relatively small increase (less than 10 percent) of combustible material to the existing inventory.

- *Fire Protection*. For the head lifting program, the NRC evaluated the fire protection measures available to cope with any fires. Fire hose stations described in the recovery technical specifications were located in two places within the containment. With the failure of a fire hose station, the licensee was required to route an equivalent capacity fire hose to the unprotected area within an hour. Fire detection instruments were provided in the containment, as described in the technical specifications. With the failure of one of these instruments, 1-hour fire watches were required.

- *Conclusion*. The NRC concluded that the existing fire protection measures, including fire watches, combustible inventory control, and video camera surveillance, were adequate to cope with the relatively small increase in combustible loading (8.9 percent) in the containment building.

11.4.4 Heavy Load Handling inside Containment

- ***Purpose***. To ensure that handling of heavy loads in the containment building and spent fuel pool "A" within the fuel handling building was in accordance with the safety requirements of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," [39] issued July 1980.

- ***Evaluation: Vital Equipment Protection***. [40] The licensee's safety evaluation considered impacts on required safe-shutdown equipment from a heavy load drop in accordance with the guidance in NUREG-0612. Based on calculations assuming the accidental drop of a postulated heavy load, Criterion IV in NUREG-0612 stated that damage to equipment in redundant or dual safe-shutdown paths would be limited so as not to result in loss of required safe-shutdown functions. Criterion IV defined "safe-shutdown functions" as those required to: (●) maintain the reactor coolant pressure boundary; (●) reach and maintain subcriticality; (●) remove decay heat; and (●) maintain the integrity of components whose failures could result in excessive offsite releases. The safety evaluation included the identification of loads, targets, and load/target interactions, as well as the evaluation of the worst credible consequence (refer to the NUREG/KM Chapter 7 on load drop evaluations for details of these evaluations). The evaluation showed that Criterion IV for damage to equipment was met.

  - *TMI-2 Safe-Shutdown Functions*. The required safe-shutdown functions applicable to the TMI-2 reactor in its current cooling mode and core configuration included: (●) capability to maintain subcriticality; (●) capability to maintain decay heat removal; and (●) capability to maintain the integrity of components whose failures could result in excessive offsite releases. The reactor coolant pressure boundary needed to be maintained only insofar as reactor coolant must be maintained in the reactor coolant system (RCS) for decay heat removal and reactivity control.

  - *Results*. Required safe-shutdown functions unique to TMI-2 during heavy load handling inside the containment building included the following:

    – *Decay Heat Removal Control*. Reactor coolant would be maintained in the RCS above the reactor vessel nozzles for decay heat removal and reactivity control. Decay heat was removed by heat losses to ambient air inside the containment building. These heat losses were demonstrated to be adequate to remove all decay heat [41] produced by the core material in the reactor vessel. As such, no additional equipment was necessary to remove decay heat.

  – *Criticality Control*. Reactivity would continue to be controlled if the level of borated water in the RCS and spent fuel pool "A"/fuel transfer canal was maintained. Thus, the drop of a heavy load would affect reactivity control only if the load drop resulted in breaking in-core instrument tubes, which would drain the reactor vessel below the 314-foot elevation. However, for the load drops postulated in the safety evaluation report (SER), the in-core instrument tubes would not break because there were no in-core instrument tubes outside of the load handling exclusion areas.

  – *Component Integrity*. As previously stated, for the load drops postulated in the SER, the in-core instrument tubes would not break because there were no in-core instrument tubes outside of the load handling exclusion areas.

- *Conclusion*. The licensee concluded that safe shutdown would be maintained for load handling and load drop accidents postulated in the SER.

Editor's Note: Refer to NUREG/KM Chapter 7 on load drop evaluations for the complete evaluation of this activity.

- ***NRC Review: Vital Equipment Protection***. [42] The NRC reviewed the results of load drops postulated in the SER against Criterion IV of NUREG-0612. The activities addressed would not result in loads over essential safe-shutdown equipment. In the current mode, forced cooling of the reactor core was not required. A dropped load could potentially cause leakage through the in-core instrument tubes. The consequence of this accident and the licensee's ability and method to mitigate the consequence were evaluated and found acceptable in a previous NRC SER [43] for heavy load handling over the reactor vessel.

11.4.5 Heavy Load Handling over the Reactor Vessel

- ***Purpose***. To permit the handling of heavy loads (greater than 2400 pounds including rigging weight) in the vicinity of the reactor vessel. This included any load handling activity that could result in a heavy load drop onto or into the vessel either directly or indirectly (as the result of the collapse of structures or equipment installed over the vessel).

- ***Evaluation: Vital Equipment Protection***. [44] The licensee's safety evaluation considered impacts on required safe-shutdown equipment from a heavy load drop in accordance with the guidance in NUREG-0612. Based on calculations assuming the accidental drop of a postulated heavy load, Criterion IV of NUREG-0612 stated that damage to equipment in redundant or dual safe-shutdown paths should be limited so as not to result in loss of required safe-shutdown functions. Criterion IV defined "safe-shutdown functions" as those required to: (●) maintain the reactor coolant pressure boundary; (●) reach and maintain subcriticality; (●) remove decay heat; and (●) maintain the integrity of components whose failures could result in excessive offsite releases. The safety evaluation included the identification of loads, targets, and load/target interactions, as well as the evaluation of the worst credible consequence (refer to NUREG/KM Chapter 7 on load drop evaluations for details of these evaluations). The evaluation showed that Criterion IV for damage to equipment was met.

- *TMI-2 Safe-Shutdown Functions*. The required safe-shutdown functions that applied to the TMI-2 reactor in its current cooling mode and core configuration included: (●) capability to maintain subcriticality, (●) capability to maintain decay heat removal, and (●) capability to maintain the integrity of components whose failures could result in excessive offsite releases. The reactor coolant pressure boundary needed to be maintained only insofar as reactor coolant must be maintained in the reactor coolant system for decay heat removal and reactivity control.

- *Results*. Reactor coolant would be maintained in the reactor coolant system above the reactor vessel nozzles for decay heat removal and reactivity control. The safety evaluation report (SER) concluded that subcriticality would be maintained (refer to Section 5.1.2 of the SER). Decay heat was removed by heat losses to ambient air inside the containment building as determined in the SER for the removal of the reactor vessel head, which was previously demonstrated to adequately remove all decay heat produced by the core material in the reactor vessel, as long as the water level in the reactor vessel was at the 314-foot elevation at a minimum. The SER stated that the water level in the reactor vessel would be maintained above the 314-foot elevation (refer to Section 3.1.3 of the SER). Therefore, the ability to adequately remove the decay heat by the losses to ambient air mode of cooling would also be maintained. As such, no additional equipment was necessary to remove decay heat. The SER indicated that offsite releases were well within acceptable limits (refer to Section 5.1.1 of the SER).

- *Conclusion*. The SER concluded that safe shutdown would be maintained for load handling and load drop accidents postulated in the SER. In addition to the safe-shutdown functions, the reactor coolant system water level would be maintained to provide personnel shielding.

Editor's Note: Refer to NUREG/KM Chapter 7 on load drop evaluations for the complete evaluation of this activity.

- ***Evaluation: Vital Equipment Protection***. [45] The NRC reviewed the licensee's SER for heavy load handling specifically over the reactor vessel. The NRC's safety evaluation included the worst case potential accident resulting from a load drop onto the reactor vessel. In this scenario, the load drop would breach the in-core instrument tubes, which penetrated the bottom of the vessel. In addition, the NRC evaluation included systems that were available to prevent the uncovering of the core such as: (●) decay heat removal system in recirculation mode; (●) a portable recirculation system with the ability to circulate more than 20 gallons per minute, which would be available for plenum lift to mix borated water in the containment building sump and transfer water from the sump back to the reactor vessel; and (●) gravity flow from the borated water storage tank to the reactor vessel (temporary solution only).

Editor's Note: Refer to NUREG/KM Chapter 7 on load drop evaluations for the complete evaluation of this activity.

11.4.6 Plenum Assembly Removal Preparatory Activities

- ***Purpose***. To confirm the adequacy of plenum removal equipment and techniques, the preparatory activities included: (●) video inspection of potential interference areas; (●) video inspection of the core void space; (●) video inspection of the axial power shaping rod (APSR) assemblies; (●) measurement of the loss-of-coolant accident restraint boss gaps; (●) measurement of the elevations of the APSR assemblies; (●) cleaning of the plenum and potential interference areas; (●) separation of unsupported fuel assembly end fittings; and (●) movement of the APSR assemblies.

- ***Evaluation: Vital Equipment Protection***. [46] The licensee's safety evaluation considered the consequences of dropping an internals indexing fixture platform shield plate onto the defueling work platform and into the reactor vessel. The evaluation concluded that the resulting reactor vessel displacements would not cause stresses on attached piping to exceed the faulted condition stress limits given in American Society of Mechanical Engineers, *Boiler and Pressure Vessel Code*, Section III, "Rules for Construction of Nuclear Facility Components," 1974 Edition, [47] which would preclude failure of the attached piping.

Editor's Note: Refer to NUREG/KM Chapter 7 on load drop evaluations for the complete evaluation of this activity.

- ***NRC Review: Vital Equipment Protection***. [48, 49] The NRC's safety evaluation also considered the consequences of dropping an internals indexing fixture platform shield plate onto the defueling work platform and into the reactor vessel.

Editor's Note: Refer to NUREG/KM Chapter 7 on load drop evaluations for the complete evaluation of this activity.

11.4.7 Plenum Assembly Removal

- ***Purpose***. To remove the plenum assembly from the reactor vessel to gain access to the core region for defueling. The plenum was moved to the deep end of the fuel transfer canal, which contained reactor coolant for shielding.

- ***Evaluation: Vital Equipment Protection***. [50] The licensee's safety evaluation considered the consequences of dropping a heavy load on the defueling work platform, reactor vessel without the platform, and fuel transfer canal. The evaluation also considered a mechanical failure of the polar crane or its rigging that could result in a plenum assembly drop.

Editor's Note: Refer to NUREG/KM Chapter 7 on load drop evaluations for the complete evaluation of this activity.

- ***NRC Review: Vital Equipment Protection***. [51] The NRC's safety evaluation concluded that, based on the agency's safety evaluation of the licensee's safety evaluation report [52] for the removal of the reactor vessel head, the potential for a load drop accident during plenum assembly lift and transfer was extremely remote and that sufficient measures were available to mitigate the consequences of such an unlikely event.

Editor's Note: Refer to NUREG/KM Chapter 7 on load drop evaluations for the complete evaluation of this activity.

11.4.8 Makeup and Purification Demineralizer Resin Sampling (NA)

11.4.9 Makeup and Purification Demineralizer Cesium Elution (NA)

11.5 Defueling Tools and Systems (NA)

11.5.1 Internals Indexing Fixture Water Processing System (NA)

11.5.2 Defueling Water Cleanup (NA)

11.5.2.1 Defueling Water Cleanup System (NA)

11.5.2.2 Cross-Connect to Reactor Vessel Cleanup System (NA)

11.5.2.3 Temporary Reactor Vessel Filtration System (NA)

11.5.2.4 Filter-Aid Feed System and Use of Diatomaceous Earth as Feed Material (NA)

11.5.2.5 Use of Coagulants (NA)

11.5.2.6 Filter Canister Media Modification (NA)

11.5.2.7 Addition of a Biocide to the Reactor Coolant System (NA)

11.5.3 Defueling Canisters and Operations

11.5.3.1 Defueling Canisters: Filter, Knockout, and Fuel (NA)

11.5.3.2 Replacement of Loaded Fuel Canister Head Gaskets (NA)

11.5.3.3 Use of Debris Containers for Removing End Fittings (NA)

11.5.3.4 Fuel Canister Storage Racks (NA)

11.5.3.5 Canister Handling and Preparation for Shipment

- ***Purpose***. To transfer defueling canisters from spent fuel pool "A" (SFP-A) to the shipping cask for offsite shipment. Defueling canisters were transferred to locations within the containment building and fuel handling building (FHB) using a transfer shield. The transfer of canisters to the shipping cask used a different device called a "fuel transfer cask."

- ***Evaluation: Fire Hazards Analysis***. (53) The licensee's safety evaluation noted that the fire analysis for this area (Zone 2) of the FHB included the model room [e] (305-foot elevation, on ground level), 328-foot elevation, 347-foot elevation, and truck bay. The model room was technically a separate fire area from the remainder of the building, but since the door between the model room and the truck bay was normally open, the room was included in the general area.

  - *Characterization/Classification of Combustibles*. Zone 2 of the FHB contained: (●) predominantly ordinary combustibles; (●) qualified [f] fire-resistant cable; (●) some unqualified cables; (●) rubber (various types) hoses; and (●) lubricants contained in recovery components and systems. The most significant fire problem in Zone 2 was the chemistry laboratory trailer (sprinkler protected) in the model room.

  - *Location of Combustibles*. The present fire loading was distributed between the 305-foot elevation model room and the 347-foot elevation spent fuel pool floor. The truck bay and dock contained few combustibles.

  - *Fire Protection Features*. The model room and spent fuel pool had fire detection and hose standpipe systems. The rolling door connecting the model room to the truck bay could be shut, if necessary, to reduce combustibles. These and other fire protection features were shown on pre-plan sketches provided in the safety evaluation report (SER). Also, Three Mile Island Unit 1 (TMI-1) erected the environmental barrier using a 2-hour fire-rated design with seals, rated at 3 hours. The wall and seals were not subject to fire surveillance.

  - *Fire Loading*. The present fire loading for Zone 2 was $3.09 \times 10^{+8}$ British thermal units (Btu) or 13,800 Btu per square foot. This equated to about a 10-minute fire loading with a peak temperature of about 1300 degrees Fahrenheit (based on the time temperature curve in the American Society of Testing and Materials [g] Standard E119, (54) "Standard Test Methods for Fire Tests of Building Construction and Materials"). An administrative limit of 80,000 Btu per square foot was established before additional reviews for fixed suppression

---

[e] The model room was just off the FHB truck bay on the ground level. As the name implies, during construction, this was where scale plant models were kept.

[f] Cables qualified under the Institute of Electrical and Electronic Engineers Standard 383, "Standard for Qualifying Electric Cables and Splices for Nuclear Facilities."

[g] Editor's Note: Currently called ASTM International.

modifications or compensatory measures, such as firewatchers. This limit represented about a 1-hour fire load with a maximum temperature of 1700 degrees Fahrenheit.

- *Conclusion*. The licensee's safety evaluation of fire hazards concluded the following: (●) Fire loads in the truck bay were very low with the principal concentration in the model room, which could be isolated from the truck bay. (●) The installed detection system would provide notification of a fire to enable the fire brigade to extinguish it promptly. (●) The environmental barrier was of a 2-hour fire-rated design and provided separation from TMI-1, with the exception of the 347-foot elevation.

- ***Evaluation: Seismic Hazard***. (55) The licensee's safety evaluation stated that equipment located in the truck bay, including shipping cask loading equipment, was designed to prevent damage to TMI-1 safe-shutdown equipment or systems (failure or collapse) during a design-basis seismic event. An evaluation was performed for canister and cask handling activities in the truck bay area to estimate the probability of the failure of equipment and structures used to perform the activity concurrent with the postulated seismic event. When the probability or consequences were acceptable, no seismic analysis was performed.

The licensee's safety evaluation concluded: (●) The FHB crane was designed to withstand the design-basis seismic event, while retaining its design-rated load, per the TMI-1 final safety analysis report. (●) The truck bay with fuel shipping equipment installed would withstand the loads imparted to the truck bay floor for all load cases. (●) When the fuel transfer cask was stacked on the shipping cask loading collar and shipping cask, the seismic analysis was not included because of the low probability of a seismic event during this configuration. (●) The mini hot cell jib crane was seismically and structurally designed; however, the crane would not be seismically qualified when loaded because of the very low probability of the occurrence of a seismic event while the jib crane was moving a heavy load. (●) The cask righting system and cask unloading station were not designed to withstand a seismic event while raising or lowering the shipping cask or skid. This determination was made because of the very low probability of the occurrence of a seismic event during this operation. (●) The fuel transfer cask loading station was classified as nonseismic since the failure of the cask loading station during a seismic event would not create any safety concerns.

- ***Evaluation: Unit 1 Impact***. (56) The licensee's safety evaluation stated that although both plant units shared the FHB crane and the truck bay, their use by one unit or the other would be determined by operational considerations on a case-by-case basis. Because the Unit 1 and Unit 2 FHBs were joined by a common area in the truck bay, the activities described in the SER were evaluated for their possible radiological impact on TMI-1. The following concerns were considered: (●) liquid release; (●) airborne release; and (●) direct radiation.

- *Liquid Release*. The activities described in the SER did not present a credible potential for radioactive liquid release to TMI-1. Any transfer of liquid, such as by the dewatering system, was controlled and maintained within the Unit 2 FHB.

- *Airborne Release*. Since the activities described in the SER were not expected to generate significant quantities of airborne radioactivity, no increase in airborne radioactivity in TMI-1 was expected.

- *Direct Radiation*. During loading of the defueling canisters into the shipping cask, the canister sources could increase the gamma dose rates in the Unit 1 FHB. To minimize the impact on Unit 1 operations and personnel exposures, operations in Unit 2 would be carried out in such a fashion that the expected dose rate in Unit 1 would not exceed 2.5 millirem per hour from Unit 2 activities. The truck bay was normally considered part of Unit 1; however, no Unit 1 work would be performed in the truck bay area without approved procedures when Unit 2 fuel shipment activities were being performed. Consequently, for the purpose of this dose assessment, the Unit 1/Unit 2 boundary was considered to be at the interface between accessible areas of the Unit 1 FHB and the truck bay. Specific points of interest were the 347.5-foot elevation of the Unit 1 FHB, the open stairway on the Unit 1 side of the truck bay, and the environmental barrier. The environmental barrier was an unshielded structure on the north side of the truck bay that separated the Unit 1 and Unit 2 atmospheres below the 347.5-foot elevation.

    - *Source Term*. The canister source term used to calculate the dose rates at the Unit 1/Unit 2 boundaries was more conservative than that used to predict average dose rates to workers in Unit 2. Previous analyses developed best estimates of anticipated dose rates. Unit 1 dose rates were analyzed to determine the maximum credible dose rate. Therefore, the source term used to predict Unit 1 dose rates included conservative parameters and was not used to provide best estimate average dose rates. These parameters used the maximum amount of cobalt-60 expected. This amount was based on material assay of cobalt-59 present in in-core structural materials. In addition, a hot channel factor of 1.9 was used to account for areas of the core where higher specific activity of radioactive materials could occur. Other conservative parameters used in previous analyses, such as maximum loaded canister and no shielding credit for the canister or its internal structures, were retained in the model used to determine the maximum credible dose rate in Unit 1.

    - *Results: Dose Rates (SFP-A Activities)*. Activities carried out within the confines of SFP-A would not affect dose rates in Unit 1, because of the distance to Unit 1 and the shielding provided by SFP-A concrete walls.

    - *Results: Dose Rates (Canister Transfers)*. Canisters were transferred from the fuel transfer cask loading platform in the spent fuel pool to the shipping cask using the fuel transfer cask. The cask was then moved along the west side of the FHB to the truck bay where it was lowered onto the shipping cask loading collar. The maximum dose rate at locations in Unit 1 from a single canister loaded in the FTC was provided in the SER (refer to page 34 of Revision 4 of the SER) for various activities. Dose rates in Unit 1 ranged from 0.3 to 1.7 millirem per hour.

– *Results: Dose Rates (Shipping Cask)*. The shipping cask could contain up to seven canisters and was designed to meet U.S. Department of Transportation regulations for shipment on public highways. The cask was designed to ensure that a limit of 10 millirem per hour at 6.6 feet from the cask would not be exceeded. The SER [57] for the Model 125-B fuel canister shipping cask showed that the highest calculated dose rate from a fully loaded shipping cask was 6.3 millirem per hour at a distance of 6.6 feet. The total dose rate from a fully loaded shipping cask or from the FTC with a single canister and a shipping cask with up to six canisters was expected to be 2.5 millirem per hour or less at all accessible areas in Unit 1.

– *Conclusion*. The licensee concluded that the activities described in the SER could be performed without an unacceptable increase in personnel exposure in Unit 1. In addition, access to the FHB was gained via the FHB truck bay door. The aircraft missile shield for this door was controlled by Unit 1. Access to the FHB was frequently required during normal plant operations; therefore, Unit 2 use of this door for fuel shipping activities would not affect normal operation of Unit 1. The SER demonstrated that the activities described in the SER would not have an unacceptable impact on the safe operation of TMI-1.

- ***NRC Review***. [58, 59] The NRC's SER did not specifically address these topics.

11.5.3.6 Canister Dewatering System (NA)

11.5.3.7 Use of Nonborated Water in Canister Loading Decontamination System (NA)

11.5.4 Testing of Core Region Defueling Techniques (NA)

11.5.5 Fines/Debris Vacuum System (NA)

11.5.6 Hydraulic Shredder (NA)

11.5.7 Plasma Arc Torch

11.5.7.1 Use of Plasma Arc Torch to Cut Upper End Fittings

- ***Purpose***. Use of a plasma arc torch to cut the upper end fittings inside the reactor vessel to allow the fittings to be placed directly in fuel canisters for transfer from the reactor vessel to the fuel canal. This was the first use of the plasma arc torch inside the reactor vessel.

- ***Evaluation; Flammable Gas***. [60] The licensee's safety evaluation noted that the gases used with the plasma arc torch, either separately or in combination, were nitrogen, carbon dioxide, and argon. Each of these gases is nonflammable and nontoxic. Testing of the plasma arc torch showed that off-gas rates of 5 to 7 standard cubic feet per minute (scfm) were typical. An additional 1 to 2 scfm could be added for the cover gas. Therefore, maximum off-gas flow was estimated at less than 10 scfm. This gas would readily mix with the ventilation flow of 4000 scfm of the defueling platform off-gas system. Any steam that could be generated during operation would condense before breaking the water surface. The off-gas from the operation of the plasma arc would not present a safety concern to the operators.

- ***Evaluation: Industrial Safety***. [61] The licensee's safety evaluation noted that during review of the licensee's safety evaluation report, the NRC had requested additional information [62] concerning the hazards from the use of the plasma arc torch. In response to those questions, the licensee provided information on the occupational hazards from metal vaporization and compressed gas usage.

  - *Metal Vaporization Hazards*. The NRC requested additional information concerning the hazards associated with the vaporization of the materials in the central zone of the plasma arc cutting target. These hazards included the generation of heavy metals and gases.

    - *Heavy Metals*. The high temperatures developed by the plasma torch would vaporize some fissile material. However, noncondensable heavy metal or fissile material gas generation would not be expected because the energies available strongly favored the formation of condensable gases. The only noncondensable gas that could be released because of vaporization of fuel would be krypton. Recent drilling into the fuel rubble clearly showed that there was little or no krypton remaining. Most condensable gases, when exposed to the reactor coolant system (RCS) water, would quickly condense. The licensee concluded that noncondensable gases would not be formed, and if condensable gases were formed, they would condense in the reactor vessel water before the gas bubble broke the water surface.

      As documented above, heavy metal gases were not expected to be formed. The concern for heavy metal involved health and safety issues related to wastewater contamination from industrial processing operations, including cadmium, chromium, copper, nickel, lead, and zinc with a limiting potable water concentration of 1.0 parts per million total heavy metals. However, this issue was not relevant to the plasma arc cutting as the RCS water could hardly be considered a potable water supply.

    - *Gases*. Plasma arc cutting could also generate gases such as carbon monoxide and oxides of nitrogen. Carbon monoxide would be generated only when carbon dioxide was used as a shielding gas. As metal vapor was oxidized by carbon dioxide, carbon monoxide would be produced. Carbon monoxide would dissolve slightly in water and could be oxidized to some extent in the RCS water. The oxides of nitrogen would react with or be miscible with water. The RCS water would become a wet scrubber for these gases. Gases that escaped from the water surface would be captured by the off-gas system and diluted to insignificant concentrations in the containment building.

  - *Compressed Gas Hazard*. The NRC requested additional information concerning the quantities and locations of the stored gases and the safety implications of an accidental release of the gas into the containment building atmosphere. In addition, the NRC requested information on the hazards and protective measures to be employed related to oxygen displacement by the gases used for plasma arc cutting.

- *Accidental Release*. Compressed gases were routinely used inside the containment building for years without incident. These gases included nitrogen, carbon dioxide, oxygen, and acetylene; most recently, argon cylinders were used for dewatering of the fuel canisters in the reactor vessel. For plasma arc cutting, no more than 10 cylinders would be in the containment building. Storage areas for full and empty cylinders were located on the 305-foot elevation (entry level) of the containment building. Compressed gases were contained in 3AA cylinders that measured 9 inches in diameter by 51 inches in length and held a volume of about 2 cubic feet of gas at about 2200 pounds per square inch. All compressed gas cylinders would be inspected, tested, and maintained by the supplier according to U.S. Occupational Safety and Health Administration (OSHA) and U.S. Department of Transportation (DOT) regulations adopted from Compressed Gas Association standards. All hoses, fittings, valves, and regulators were used according to the Association's standards. Compressed gas cylinder leaks would most often occur in the regulator or hose during use. The user would quickly become aware of the leak through sounds of leaking gas or by noticing a reduction in the cylinder pressure. Quick actions by the user to close the cylinder and exit the area prevented safety consequences.

- *Oxygen Displacement*. Oxygen displacement was a concern during the use of gases. In all cases, sufficient ventilation was always provided in the containment building for dilution to ensure safe oxygen concentrations. For example, with normal containment building ventilation, the accidental release of the entire contents of one compressed gas cylinder would have a negligible effect on oxygen concentrations. One compressed gas cylinder would release about 270 cubic feet of gas at atmospheric pressure. While this would affect the oxygen concentration in the immediate area for a very short time, the excellent mixing of the containment building air by the operation of the fans would quickly mix and dilute this gas. The 270 cubic feet of gas released into a building of about 2 million cubic feet would have an insignificant effect on oxygen concentrations (the release would lower oxygen concentrations from 20.9 to 20.89 percent). About 100 compressed gas cylinders (27,000 cubic feet) would have to release their contents at one time for oxygen levels to drop to 19.5 percent. This analysis did not consider the operation of the containment building purge (18,000 to 38,000 cubic feet per minute (cfm)).

- *Conclusion*. The use of compressed gases for underwater plasma arc cutting inside the reactor vessel would be controlled by approved procedures that were in accordance with OSHA and DOT regulations. The licensee was confident that the use of compressed gases and any accidental release that could occur would be safely controlled, posing no safety hazard to defueling personnel or others inside the containment building.

o *Nitrogen Oxides*. In a supplemental safety evaluation [63] to the NRC, the licensee indicated that nitrogen would be used as the primary and secondary torch gas. During underwater plasma arc cutting operations with nitrogen, the relatively abundant elements of nitrogen and oxygen could combine to form oxides of different molecular composition. The principal oxides of concern were nitric oxide (NO) and nitrogen dioxide (NO_2). During cutting operations, NO would be formed initially, as its formation was more likely at the high temperatures produced by the plasma arc torch. NO was subsequently oxidized in the atmosphere to the more toxic and irritating compound NO_2.

- *Physical Properties*. NO gas is colorless, odorless, and only slightly soluble (about 60 parts per million (ppm) by weight at 25 degrees Celsius in water). The threshold limit value for NO as listed by the American Conference of Governmental Industrial Hygienists (AIGIH) was 30 milligrams per cubic meter or 25 ppm NO by volume. NO_2 gas is a light, yellowish-orange color at low concentrations and a reddish-brown color at high concentrations. NO_2 has a very pungent odor and a high oxidation rate and is very corrosive. The threshold limit value for NO_2 as listed by the AIGIH was 6 milligrams per cubic meter or 3 ppm NO_2 by volume. The oxides NO and NO_2 are commonly designated by the composite formula NO_x.

- *Data Measurements*. The licensee experimentally established the fractional conversion of nitrogen to NO_x from off-gas sampling measurements made during the course of testing the TMI-2 plasma arc torch. These sampling measurements were made during torch cutting operations in a small test tank (about 2200 gallons). The gas sampling results indicated one peak measurement of 7200 ppm; however, a simple arithmetic average of all peak measurements was less than 3000 ppm. Consequently, while the NO_x measurements varied, all gas sampling data indicated that less than 1 volume percent of the total nitrogen torch gas volume (4 cfm primary and 7 cfm secondary) was converted to NO_x during torch operation, and the majority of the data supported less than 0.3 volume percent [i.e., 0.003 cubic foot of NO_x per cubic foot of nitrogen (N^2)].

- *Flowpath*. During cutting in the reactor vessel, the off-gas would be collected above the reactor vessel and transferred to the “B” D-ring near the plant purge system exhaust suction point. The purge was routed to the station vent, where the exhaust was again diluted by the auxiliary building and fuel handling building ventilation system exhausts. Thus, the off-gas from the plasma arc cutting was diluted with large air volumes of the containment building or large airflows existing in plant ventilation systems before any exposure to personnel or the environment.

- *Concentration (Release Rate)*. The maximum NO production rate during cutting was about 0.033 cfm [(0.003 cubic foot of NO_x per cubic foot of N_2) x (11 cubic feet of N_2 per minute)]. Using this production rate, the maximum concentration for any dilution flow rate could be estimated. For example, at the release point from the defueling work platform off-gas system, which had a flow rate of about 4000 cfm, the concentration would be about 8 ppm of NO_x by volume [(0.033 cubic foot of NO_x per minute)/(4000 cubic feet of air per minute)]. The maximum concentration in the purge system, assuming direct intake from the off-gas system, and assuming a purge flow rate of 25,000 cfm, would be about 1 ppm. For a station vent flow rate of 100,000 cfm, the release point concentration would be 0.33 ppm by volume.

– *Concentration (Containment)*. The average concentration in the containment building could also be calculated for different time intervals. Assuming no purge operation, a containment building air volume of $2x10^{+6}$ cubic feet, and 2 minutes of cutting per hour, the average concentration at the end of a 24-hour period would be less than 1 ppm [(0.033 cubic foot per minute x 2 minutes per hour x 24 hours)/($2x10^{+6}$ cubic feet)]. Note that this calculation assumed uniform mixing in the containment building air volume. This calculation also assumed an optimistic cutting rate and an average production rate of 3000 ppm measured during test cutting.

– *Controls*. The safety evaluation report showed that the concentrations of byproduct gas in the containment building were expected to remain acceptably low, even during periods of no purge operation. Administrative controls would be required to ensure that personnel access to the "B" D-ring was prohibited during plasma arc cutting. The safety department would monitor the work area for byproduct gas to ensure that occupational exposure limits were not exceeded.

– *Release Permit*. In addition, since NO_2 was listed under the National Emission Standards for Hazardous Air Pollutants as promulgated by the U.S. Environmental Protection Agency under the Federal Clean Air Act, the licensee submitted an application requesting State exemption from plan approval and permitting requirements with respect to potential NO_2 emissions. The State granted the exemption.

- ***Evaluation: Instrument Interference***. [64] The licensee's safety evaluation stated that a high-frequency generator would be used to establish and stabilize the arc between the electrode and the work piece. The frequencies encountered had the capability to disrupt instruments inside the containment building, the most crucial being the criticality monitors. However, instrument interference was not expected to be a problem because of the shielding provided by the reactor vessel water and because the high-frequency generator was also shielded. In addition, the high-frequency generator was used only for very short periods of time to establish and stabilize the arc. If interference with the criticality monitors occurred, administrative controls would be developed to prohibit any fuel movements in the containment building during plasma arc torch operations. Any interference with other instruments would be handled in a similar manner.

- ***NRC Review: Industrial Hazard (Nickel Carbonyl)***. [65] The NRC's safety evaluation identified a potential concern with the generation of nickel carbonyl from plasma arc cutting and other forms of melting vaporization techniques. The licensee provided additional information for NRC review and evaluation (as described above).

- *Background*. In response to the NRC's letter requesting information, [66] the licensee's replied with a letter [67] that provided an evaluation of plasma arc torch tests performed by INEL and operating experience with plasma arc cutting at TMI-2. (Refer to the licensee's response letter for details.)

  In response to this additional information, the NRC stated in a subsequent letter [68] to the licensee that the licensee's response inadequately addressed the NRC's concerns on the formation of nickel carbonyl. The NRC asked that the licensee provide adequate analytical data and literature references describing the chemical processes that would affect the formation of nickel carbonyl to further support the licensee's position that this compound presented no safety hazard. As an alternative, the licensee could describe the theoretical maximum nickel carbonyl formation rates based on cutting rates and chemical composition of the alloys being cut and address the protective measures that would be used to protect the workers to ensure that toxic substances did not accumulate in the containment building.

  The licensee's second response letter [69] concluded that, based on its review of the chemical and physical properties of nickel carbonyl, the potential for creating a safety hazard was small. The licensee also summarized extensive studies by the International Nickel Company on this issue based on recently published work and provided a clear and definitive demonstration that nickel carbonyl was not a concern during cutting and welding operations at high temperatures. (Refer to the licensee's second response letter for details.)

  In addition, the licensee stated that it planned to use nitrogen for both the primary and secondary torch gas (carbon dioxide would not be used, and the use of argon was unlikely). The off-gas releases from the use of nitrogen were evaluated by the licensee and reported by separate letter [70] to the NRC. The safety evaluation concluded that off-gas releases (i.e., NO and NO_2) as a result of using nitrogen was not a safety concern. (Refer to the licensee's second response letter for details.)

- *Conclusion*. The NRC review determined that the licensee did not provide adequate information pertaining to the potential for generating nickel carbonyl and other toxic substances during plasma arc torch use and their concentration in confined spaces to make a conclusive determination that there was no safety hazard. However, the licensee's safety evaluation stated that during cutting in the reactor vessel, the off-gas would be collected above the reactor vessel and transferred to the "B" D-ring in the vicinity of the plant purge system exhaust suction point. The NRC found this to be an acceptable solution to the potential accumulation of toxic substances in confirmed spaces during plasma arc cutting inside the reactor vessel.

  The NRC concluded that the proposed operation of using a plasma arc torch to cut upper end fittings inside the reactor vessel presented no adverse impact on the public health and safety. This activity was supported based on the following conditions that were consistently listed in the licensee's submittals: (●) Only upper end fittings would be cut using the plasma arc torch. (●) The cutting station would be located within the 4-foot exclusion zone. (●) Nitrogen would be used as both primary and secondary torch gases. (●) Effluent from the defueling work platform gas system would be routed to the vicinity of the containment building purge system suction point. (●) The defueling work platform off-gas system and containment building purge system would be in operation during plasma arc cutting.

- ***NRC Review: Instrument Interference***. [(71)] In the NRC's approval of the licensee's request to use the plasma arc torch to cut upper end fittings in the reactor vessel, the NRC stated that in the event that plasma arc cutting interfered with nuclear instruments that monitored nuclear core conditions, no additional core alterations could take place during the cutting of upper end fittings in the reactor vessel.

11.5.7.2 Use of Plasma Arc Torch to Cut the Lower Core Support Assembly (NA)

11.5.7.3 Use of Plasma Arc Torch to Cut Baffle Plates and Core Support Shield (NA)

11.5.7.4 Use of Air as Secondary Gas for Plasma Arc Torch

- ***Purpose***. To replace nitrogen gas with air as the secondary gas to improve plasma arc torch performance by achieving longer and more efficient cuts.

- ***Evaluation: Industrial Safety***. [(72)] The licensee's safety evaluation noted that the potential for the generation of gases during plasma arc cutting had been previously evaluated [(73, 74)] by the licensee and reviewed [(75, 76)] by the NRC. The primary concern addressed in the safety evaluation report was the potential for generating nickel carbonyl and other toxic substances during plasma arc cutting. Previous use of the plasma arc torch had not created any adverse health effects. Since air contains less nitrogen than pure nitrogen, the use of air as the secondary gas for the automated cutting equipment system plasma arc torch was not expected to increase the potential for worker exposure to toxic substances. However, in accordance with an NRC requirement specified in its safety evaluation reports, the defueling work platform off-gas system and the containment building purge system would be operated whenever plasma arc cutting was performed to ensure the proper ventilation of any off-gases that might be generated. As an additional precaution, during the first cuts made with air as the secondary gas, samples would be taken from above the water surface to be analyzed for toxic substances. The evaluation concluded that this activity would not increase the probability of occurrence, the consequences of an accident, or malfunction of equipment important to safety which had been previously evaluated in the safety analysis report.

- ***NRC Review***. Editor's Note: The NRC's safety evaluation could not be located.

11.5.8 Use of Core Bore Machine for Dismantling Lower Core Support Assembly (NA)

11.5.9 Sediment Transfer and Processing Operations

- ***Purpose***. To collect sediment from tanks and sumps in the auxiliary and fuel handling buildings, and also from the containment building basement and sump, in order to transfer the sediment to the spent resin storage tanks and treat or process the sediment (for disposal).

- ***Evaluation: Intersystem Interactions***. [77] The licensee's safety evaluation noted that there were several tie-in points where new piping was connected to existing plant systems to incorporate the system hardware into the process. The sediment transfer and processing system interfaced with several temporary and permanent plant systems (refer to Section 4.1 of the safety evaluation report for details). Testing before system operation would determine the integrity of these barriers or the need to provide additional isolation. All process flow lines would be hydrostatically tested in accordance with American National Standards Institute Standard B31.1, "Power Piping," [78] to guard against line breaks, valve, pump, and flange leaks. Hydrostatic testing would be performed using the decontamination processed water system before any sediment handling operations.

- ***NRC Review***. [79] The NRC's safety evaluation report did not specifically address these topics.

11.5.10 Pressurizer Spray Line Defueling System (NA)

11.5.11 Decontamination Using Ultrahigh Pressure Water Flush

- ***Purpose***. To use ultrahigh pressure (UHP) water flush at 20,000 to 55,000 pounds per square inch (psi) to remove surface coatings and surface contamination inside the containment building.

- ***Evaluation: Vital Equipment Protection***. [80] The licensee's safety evaluation stated that a review would be performed before the decontamination of each area. This review would identify essential systems, structures, instruments, and other components in each area based on the following criteria: (●) necessary to protect the integrity of the reactor coolant system; (●) required to maintain and monitor boron concentration in the reactor coolant system; (●) required to prevent unacceptable offsite releases; and (●) required to be operable by the recovery technical specifications. Components that were identified as essential would not be decontaminated or would be protected during UHP water flush to avoid damage.

Concrete embeds (i.e., a fixed object embedded firmly and deeply in the surrounding concrete) for essential supports need not be avoided during UHP water flush decontamination activities, as embeds would not be damaged by the UHP waterjet. Additionally, analyses would be performed to determine the amount of concrete that could be removed from surfaces adjacent to embeds without impacting the load-bearing capacity of these embeds. If an embed could not tolerate any concrete removal, the embed would be identified and avoided in the appropriate work order. Furthermore, the removal of the small quantity of surface concrete adjacent to an embed would not significantly affect its capacity to resist the loads during the recovery.

Operators would be trained to avoid maintaining a constant jet at a single location on any surface. Additional procedural requirements for the use of the UHP water flush would be imposed as appropriate.

- ***Evaluation: Industrial Safety***. [81] The UHP water flush used water pressurized between 20,000 and 55,000 psi. The licensee's safety evaluation stated that water at this discharge pressure could create several personnel safety concerns.

- o *Hazards*. Personnel safety concerns included injury due to dust and other airborne particulates, exposure to discharge flow, and hose failure.

  - – *Airborne Hazard*. Operators using the UHP would wear anticontamination clothing and respirators. Experience with warm water hydro lasers showed that airborne mists would not impact the effectiveness of respirators. These measures would reduce the potential for injury due to dust and other airborne particulates.

  - – *Hose Failure Hazard*. The failure of a hose in the UHP water flush system would not create the potential for personnel injury as the pump discharge rate was not capable of maintaining pressure in the hose at its failure point. This failure would result in a sudden hose depressurization, which presented little danger to personnel.

  - – *Discharge Flow Hazard*. During performance of dose reduction and decontamination activities, personnel health and safety hazards would be reduced to as low a level of risk as was reasonably achievable. Certain hazards inherent in the operations being conducted could include: (●) falls; (●) high-pressure water sprays; (●) noise; (●) eye injury; (●) tripping; (●) rotating equipment; (●) electrical shock; (●) suspended equipment; (●) heat stress; and (●) sharp objects.

- o *Hazard Prevention*. Preventive measures and safety precautions that were taken to reduce worker hazards included the following:

  - – *Procedures/Training*. Written procedures, personnel training, and use of safety equipment were incorporated to minimize the risk associated with these hazards. Personnel received extensive training and instruction in the proper use of high-pressure sprays to prevent personnel injury.

  - – *Precautions*. The equipment was designed with features that minimized the potential for operator injury. The following additional safety precautions would be taken during UHP flush operations: (●) No body part would be permitted to approach within 1 meter in front of the discharge nozzle. (●) Personnel would be kept away from moving parts of the pump during operation. (●) Operation of the pumps when a hose was kinked or twisted would be avoided. (●) The pump would not be moved by pulling on the hose. (●) No work would be performed on the machine during operation. (●) Precautions would be taken to keep the pump dry. (●) Only approved fittings, lubricants, and pump parts would be used. (●) Nozzles would not be loosened or removed during pump operation. (●) Water filters would be changed as necessary. (●) Extra time would be taken to warm the pump when in cold areas. (●) Lubricant reservoirs would be filled as appropriate. (●) The nozzle would be immersed in water during operation. (●) Precautions would be taken to avoid electrical shocks, and connections were not to be made or broken unless electrical power was off. (●) Personnel protection devices, such as foot protection, would be used as required by the licensee's safety department.

- ***NRC Review: Vital Equipment Protection***. [82] The NRC's safety evaluation considered the potential effects on important safety components from prolonged impingement by the UHP water jet. The water jet ranged in size from 0.005 to 0.025 inch in diameter at pressures of 20,000 to 55,000 psi. A jet of this nature could remove galvanized or anodized surfaces from metals and could cause surface damage to soft metals such as copper or aluminum. However, coatings, such as paint, could be removed from soft metals without metal surface damage if care was taken to avoid allowing the jet to dwell for a long time on one spot. This could be accomplished by using a rapidly traversing or rotating nozzle. Harder metals, such as steels, alloys, and cast iron, would not be damaged by rotating or traversing jets. Prolonged impingement of the jet on a hard surface could cause some surface deterioration. Although there was only a small potential for damage from the jet during decontamination operations, the licensee would evaluate each area to be documented and ensure that vital components were either avoided or protected.

The operations would be administratively controlled to prevent the jet from dwelling for an extended time period at reactor coolant system piping systems where a failure induced by the UHP jet could cause draining of the reactor vessel below the level of the hot-leg nozzles, except for the in-core instrument piping. The areas around the in-core instrument piping would be avoided or protected by procedural controls or physical barriers. In addition, safety evaluations that were performed in support of an unrelated technical specification change [83] demonstrated that the reactor water level could be safely maintained in the event of an in-core instrument pipe break.

The NRC concluded that the licensee's proposed program did not present the potential for damage to components from water impingement that could result in any undue risk to the health and safety of the public.

11.6 Evaluations for Defueling Operations

11.6.1 Preliminary Defueling (NA)

11.6.2 Early Defueling

- ***Purpose***. To remove end fittings, structural materials, and related loose debris that would not involve removal of significant amounts of fuel material, to remove intact segments of fuel rods and other pieces of core debris, and to remove loose fuel fines (particles) by vacuuming operations.

- ***Evaluation: Fire Protection***. [84] Editor's Note: The licensee's safety evaluation for early defueling was practically identical to its subsequent safety evaluation report [85, 86] for bulk defueling; therefore, this section does not include the evaluation text. For details, please refer to the next section.

- ***Evaluation: Unit 1 Impact***. [87] Editor's Note: The licensee's safety evaluation for early defueling was practically identical to its subsequent safety evaluation report [88, 89] for bulk defueling; therefore, this section does not include the evaluation text. For details, please refer to the next section.

- ***NRC Review: Fire Protection***. [90] The NRC's safety evaluation stated that fire protection during early defueling activities would be provided in accordance with the TMI-2 fire protection program and associated procedures for control of combustible materials. Fire extinguishers and detection equipment would be available in the containment building to mitigate the consequences of a potential fire. The NRC did not anticipate that defueling operations would significantly increase the potential for a fire.

11.6.3 Storage of Upper End Fittings in an Array of 55-Gallon Drums (NA)

11.6.4 Defueling (Also Known as "Bulk" Defueling)

- ***Purpose***. To develop defueling tools and to conduct activities necessary to remove the remaining fuel and structural debris located in the original core volume. An additional activity was to address vacuuming of debris from the lower head of the vessel, a primary activity associated with lower head defueling.

- ***Evaluation: Fire Protection***. [91] The licensee's safety evaluation stated that fire protection during the defueling activities would be provided in accordance with the requirements of the licensee's fire protection program evaluation and the administrative procedure for the control of combustible materials. Existing fire detection or fire extinguishing equipment in the containment building was available for defueling. This would ensure that the potential for and consequences of a fire were minimized.

- ***Evaluation: Unit 1 Impact***. [92] The licensee's safety evaluation stated that the major impact of defueling on plant activities would be the effect of fuel movement in Unit 2 on operations in Unit 1. The NRC Atomic Safety and Licensing Board imposed a condition on the restart of TMI Unit 1: either the effects of TMI-2 fuel movement on TMI-1 personnel in the fuel handling building (FHB) would be addressed by NRC-approved procedures, or the work in the Unit 1 area of the FHB would be suspended during Unit 2 fuel movement.

- *Worst Case Accident (Canister Drop)*. Because of the environmental barrier that isolated the Unit 1 auxiliary building from the FHB areas of Unit 1 and Unit 2, the only Unit 1 area that potentially would be affected by Unit 2 fuel movement was the Unit 1 FHB area. The worst case defueling accident in the FHB within the scope of this safety evaluation was a defueling canister drop. An evaluation was performed to determine the radiological consequences from a canister drop in the FHB. The scope of this safety evaluation ended with the storage of the canisters in the racks; therefore, all canister movements in the FHB covered by the safety evaluation report would be made over the "A" spent fuel pool. Consequently, any postulated canister drops would be into the pool. Even though the design specifications of the canister allowed for canister leakage, the licensee did not expect that leakage would result from such a drop.

- *Results and Conclusion*. If any leakage were to occur, the leak would occur underwater; therefore, no airborne particulates would be released from a canister drop. Further, any debris that was released into the water would be shielded by the pool water so that the contribution to the area dose rate would be negligible. Ultimately, the debris would be cleaned up by the defueling water cleanup system or an alternate cleanup system. Releases of krypton-85 would be within acceptable limits as demonstrated in the safety evaluation report. The evaluation concluded that defueling operations in Unit 2 would not affect personnel in Unit 1.

- ***NRC Review: Fire Protection***. (93) The NRC's safety evaluation concluded that fire protection safety had been adequately addressed in previous NRC safety evaluations and that the agency's earlier conclusions were applicable to the proposed activities (refer to the NRC's safety evaluation report (94) for early defueling).

11.6.5 Use of Core Bore Machine for Bulk Defueling (NA)

11.6.6 Lower Core Support Assembly Defueling

- ***Purpose***. To dismantle and defuel the lower core support assembly (LCSA) and to partially defuel the lower reactor vessel head. Structural material removed from the LCSA included the: (●) lower grid rib assembly; (●) lower grid forging; (●) distribution plate; and (●) in-core guide support plate. The lowest elliptical flow distributor and the gusseted in-core guide tubes would not be removed under this proposed activity. The use of the core bore machine, plasma arc cutting equipment, cavitating water jet, robot manipulators, automatic cutting equipment system, and other previously approved tools and equipment was included in this activity and associated safety evaluations.

- ***Evaluation: Industrial Safety (Submerged Combustion)***. (95) The licensee's safety evaluation stated that the use of underwater burning devices (e.g., the plasma arc torch) created a heat source that was not considered in previous licensee safety evaluations. This additional heat source was not expected to create a combustion concern since the plasma arc torch would be operated underwater. Additionally, testing of thermic torch and plasma arc burning devices on alumina-filled zirconium tubes underwater did not produce any sustained ignition. [96] The licensee determined that it was reasonable not to postulate a combustion reaction of exposed fuel debris due to operation of the plasma arc torch.

- ***Evaluation: Fire Protection***. [97] The licensee's safety evaluation stated that fire hazards caused by the use of the plasma arc torch during LCSA defueling were bounded by the safety evaluation report (SER) [98] for bulk defueling.

- ***Evaluation: Instrument Interference***. [99] The licensee's safety evaluation stated that issues of instrument interference caused by the use of the plasma arc torch were bounded by its SER [100] for the use of the plasma arc torch.

- ***Evaluation: Unit 1 Impact***. [101] The licensee's safety evaluation stated that the major potential impact of LCSA defueling on plant activities was the effect of fuel movement in Unit 2 on operations in Unit 1. Based on the evaluation provided in the licensee's previous SER [102] for bulk defueling and the similarity of the activities considered in that report to those activities within the scope of this SER, the licensee concluded that the LCSA defueling operations in Unit 2 would not affect personnel in Unit 1.

- ***NRC Review: Industrial Safety***. [103] The NRC's safety evaluation concluded that the proposed activities could be accomplished without significant risk to the health and safety of the public, provided that they complied with the limits stated in the licensee's submittals and the limits in the NRC's SER. The following limitations related to industrial safety would apply: (●) Nitrogen would be used as both the primary and secondary plasma torch gas. (●) Effluent from the defueling work platform off-gas system would be routed to the vicinity of the containment building purge system suction point. (●) The defueling work platform off-gas system and containment building purge system would be operating whenever plasma arc cutting was performed.

11.6.7 Completion of Lower Core Support Assembly and Lower Head Defueling

- ***Purpose***. To remove the elliptical flow distributor and gusseted in-core guide tubes of the lower core support assembly (LCSA) and to defuel the reactor vessel lower head (RVLH).

- ***Evaluation: Industrial Safety (Submerged Combustion)***. [104] The licensee's safety evaluation stated that the use of the plasma arc torch created a heat source. This condition was evaluated in previous licensee and NRC safety evaluation reports [105, 106] for LCSA defueling. This additional heat source was not expected to create a combustion concern since the plasma arc torch operated underwater. Additionally, testing of thermic torch and plasma arc burning devices on alumina-filled zirconium tubes underwater did not produce any sustained ignition, based on results of an INEL plasma arc test. The licensee concluded that it was reasonable not to postulate a combustion reaction of exposed fuel debris due to operation of the plasma arc torch.

- ***Evaluation: Fire Protection***. (107) The licensee's safety evaluation stated that fire hazards caused by the use of the plasma arc torch during LCSA/RVLH defueling were bounded by its safety evaluation report (SER) (108) for bulk defueling.

- ***Evaluation: Instrument Interference***. (109) The licensee's safety evaluation noted that the operation of the plasma arc torch within the reactor vessel during previous uses of the torch did not result in any disruptions of instrumentation required by technical specifications.

- ***Evaluation: Unit 1 Impact***. (110) The licensee's safety evaluation stated the major potential impact of LCSA/RVLH defueling on plant activities was the effect of fuel movement in Unit 2 on operations in Unit 1. Based on the evaluation provided in the licensee's previous SER (111) for bulk defueling and the similarity of the activities considered in that report to those activities within the scope of this SER, the licensee concluded that the LCSA/RVLH defueling operations in Unit 2 would not affect personnel in Unit 1.

- ***NRC Review***. (112) The NRC's SER did not specifically address these topics.

11.6.8 Upper Core Support Assembly Defueling

- ***Purpose***. To cut and move the baffle plates and to defuel the upper core support assembly (UCSA). This evaluation addressed the following activities: (●) cutting the baffle plates for later removal; (●) removing bolts from the baffle plates; (●) removing the baffle plates; and (●) removing core debris from the baffle plates and core former plates.

- ***Evaluation: Industrial Safety (Submerged Combustion)***. (113) The licensee's safety evaluation stated that the use of the plasma arc torch created a heat source. This condition was evaluated in the licensee's previous safety evaluation report (SER) (114) for lower core support assembly defueling. This additional heat source was not expected to create a combustion concern since the plasma arc torch would be operated underwater. Additionally, testing at INEL of thermic torch and plasma arc burning devices on alumina-filled zirconium tubes underwater did not produce any sustained ignition. It was considered reasonable not to postulate a combustion reaction of exposed fuel debris due to operation of the plasma arc torch. Experience from previous defueling operations to date confirmed that submerged combustion was not a concern during plasma arc torch operation.

- ***Evaluation: Fire Protection***. (115) The licensee's safety evaluation stated that fire hazards caused by the use of the plasma arc torch during UCSA defueling were bounded by its safety SER (116) for bulk defueling.

- ***Evaluation: Unit 1 Impact***. (117) The licensee's safety evaluation stated that the major potential impact of UCSA defueling on plant activities was the effect of fuel movement in Unit 2 on operations in Unit 1. Based on the evaluation provided in the licensee's SER [118] for bulk defueling and the similarity of the activities considered in that report to those activities within the scope of this SER, the licensee concluded that the UCSA defueling operations in Unit 2 would not affect personnel in Unit 1.

- ***NRC Review***. [119] The NRC's SER did not specifically address these topics.

11.7 Evaluations for Waste Management

11.7.1 EPICOR II

- ***Purpose***. To decontaminate accident-generated, intermediate-level radioactive wastewater being held in tanks in the auxiliary building. Later, the system was used to polish effluents from the submerged demineralizer system (SDS) during the cleanup of highly radioactive water from the containment building sump, reactor coolant system, and reactor coolant drain tanks. Following the decommissioning of the SDS, EPICOR II was used to clean residual wastewater from decontaminating the structures and systems.

- ***Evaluation***. Editor's Note: The licensee's safety evaluation was not located.

- ***NRC Review: Other Topics***. Editor's Note: The NRC's formal review was an environmental assessment [120] that was issued October 1979 as NUREG-0591, "Environmental Assessment on the Use of EPICOR II at Three Mile Island Unit 2." An updated environmental assessment that applied to EPICOR II and all other cleanup activities was documented in the PEIS [121] issued March 1981. These reports addressed the following safety topics: (●) occupational exposure; (●) radiation protection/ALARA/shielding; and (●) radiological release. The NRC's safety evaluations did not specifically address any other topics. Refer to these reports for details of the NRC's evaluations.

11.7.2 Submerged Demineralizer System

11.7.2.1 Submerged Demineralizer System Operations

- ***Purpose***. To decontaminate the containment building sump water and reactor coolant system (RCS) water using the submerged demineralizer system (SDS), followed by effluent polishing with the EPICOR II system.

- ***Evaluation: Industrial Safety***. [122] The licensee's safety evaluation considered occupational and public safety hazards.

  - *Occupational Safety*. During the operation of the SDS, operating personnel would adhere to station requirements for occupational safety. Structural equipment and operating equipment used would meet the applicable requirements of the U.S. Occupational Safety and Health

Administration. Personnel protective equipment required for the operation of the SDS would be used in accordance with licensee procedures.

- *Public Safety*. From an industrial safety standpoint, operation of the SDS posed no risk to the general public because the lifting and handling activities described were performed within the TMI complex and because hazardous chemical species, flammable or explosive substances, heavy industrial processes, and concentrated manufacturing activities would not be involved in the installation or operations of the SDS.

- ***NRC Review***. [(123)] The NRC's safety evaluation did not specifically address these topics.

11.7.2.2 Submerged Demineralizer System Liner Recombiner and Vacuum Outgassing System (NA)

11.8 Other Safety Topics

11.8.1 Seismic Design Requirements of TMI-2 Postaccident Systems

- ***Purpose***. To demonstrate that the systems installed after the 1979 accident, or that were contemplated for later installation for the sole purpose of supporting recovery activities, were not required to meet seismic design requirements.

- ***Evaluation***. [(124)] Editor's Note: The licensee's analyses of postaccident systems and processes considered the consequences of postulated accidents due to a seismic event. The evaluation was conducted in three steps: (1) identification of source terms and driving forces; (2) specification of seismic-induced accident sequences; and (3) consequence analysis. Refer to Section 2 of this NUREG/KM chapter for a summary of this safety evaluation.

- ***NRC Review***. [(125)] The NRC's safety evaluation concluded that systems installed after the 1979 accident, or that were likely installed for later cleanup activities, would not be required to meet seismic design criteria referenced in Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants," [(126)] to 10 CFR Part 100. However, the NRC concluded that the applicability of the general design criteria in Appendix A [(127)] to 10 CFR Part 50 required assessment on a case-by-case basis. Individual determinations for seismic design had been made for the previously installed cleanup systems. The NRC concluded that any systems installed in the future would require this same individual assessment. The licensee demonstrated that, in general, the offsite dose consequences of potential worst case accidents would remain relatively small. The NRC also stated that this information was likely to be relevant to later cleanup system evaluations and would be referenced in any submittal requesting NRC approval.

11.8.2 Seismic Design Requirements for Penetrations

- ***Purpose***. To relax the constraint of 20 square feet of modified penetrations (or open pathways) between the containment building and the auxiliary and fuel handling building. A previous safety evaluation (see background section below) for an exemption to seismic design criteria for modified containment penetrations was based on a maximum 20 square feet of modified penetrations.

- ***Background (Original General Design Criterion 2 Exemption)***. (128) In 1984, the NRC approved a request for an exemption from 10 CFR Part 50, Appendix A, General Design Criterion (GDC) 2, "Design bases for protection against natural phenomena," regarding the design of containment penetrations after the removal of the reactor vessel head. The NRC evaluated the potential offsite dose consequences of a containment isolation valve failure when challenged by natural phenomena (e.g., earthquakes, tornados). The failure of the penetration itself did not present a potential hazard unless accompanied by a simultaneous event in the containment, which would cause the release of radioactive material. The NRC evaluated the potential offsite dose consequences of the failure of one or more penetrations coupled with a broad range of accidents in the containment. Calculations were performed to estimate the offsite dose consequences of various accident scenarios involving a breach of nonseismic containment penetrations. The scenarios were selected to be representative of the types and conditions that could occur at TMI-2 during defueling activities. The scenarios chosen were at the severe end of the spectrum. Four worst-case scenarios included: (●) fire in a radioactive materials storage area; (●) reactor coolant leak; (●) water processing or fuel canister drop; and (●) pyrophoric event. All of these scenarios assumed the concurrent failure of a containment penetration. The NRC developed and evaluated a representative source term for the offsite dose calculations.

The NRC evaluation considered the potential offsite dose consequences for the four worst case scenarios. The results of these scenarios show that the worst case offsite dose projections at the exclusion area boundary were within the exposure guidelines of 10 CFR Part 20. The NRC analysis found the effect of a penetration failure and simultaneous seismic event to be within 10 CFR Part 20 guidelines. Therefore, the NRC concluded that there was no undue risk to the health and safety of the public resulting from a seismically induced penetration failure and determined that the licensee's exemption request was justified. However, the NRC clarified in the subsequent basis document (129) that the (original) analysis was valid for up to 20 square feet of penetrations in the auxiliary and fuel handling building.

- ***Evaluation***. (130) The licensee requested that the original 20-square-foot constraint on modified penetrations be removed from the GDC exemption. The request submittal was brief and provided few details, except its reference to a licensee's previous submittal that documented the seismic design criteria safety evaluation (131) from a year earlier. Specific safety hazards were not mentioned in the request but were evaluated in detail in the prior evaluation. The details from the licensee's request letter are provided below. Section 2 in this NUREG/KM chapter summarizes details of the prior evaluation.

- *Analysis*. The submittal indicated that the prior analysis considered a broad spectrum of potential accident source terms and driving forces for radionuclide movement into the environment. The source terms were selected to bound recovery activities, including those that were ongoing at the time and those that were to occur at some later time. Each source term was coupled with conservative driving forces to bound the potential offsite consequences. The accident scenarios in the analysis included potential seismically induced accident releases associated with the following: (●) drained reactor pressure vessel; (●) wall and equipment contamination; (●) makeup and purification system cleanup; (●) EPICOR II processing; (●) submerged demineralizer system processing; (●) defueling water cleanup system processing; (●) sediment transfer and processing; (●) defueling canister movement; (●) radioactive trash storage areas; and (●) other radioactive waste storage areas (e.g., interim solid waste staging facility).

  In addition, this analysis assumed that the environmental releases for the noted seismically induced accidents were unfiltered and occurred through open pathways (e.g., a collapsed building, open fuel handling building truck bay door, or an open containment building equipment hatch). The cross-sectional area of these pathways was much greater than the 20-square-feet area limit established in the analysis.

- *Conclusion*. The primary conclusion of the licensee's previous analysis was that the failure of any TMI-2 recovery structure, system, or component, as a result of a seismic event, would not result in a radiological release in excess of a small fraction of the guideline values in 10 CFR Part 100. Therefore, the licensee concluded that the imposition of a numerical restriction on the cumulative area of modified containment penetrations was not warranted technically. The licensee also expressed concern that the limit could pose an undue restriction during later recovery activities, which could necessitate additional modification of existing penetrations.

- ***NRC Review***. [132] The NRC's previous analysis restricted the modified containment penetrations to the auxiliary and fuel handling building (including the annulus area) to a total area of 20 square feet. This restriction did not represent the limiting case in the NRC's analysis, since the previous analysis assumed double the amount of these penetrations and double the source terms leaving the containment building via these penetrations. The resultant worst case involved a dropped fuel canister coincident with the failed penetrations. The potential offsite dose consequences for the maximally exposed individual were less than 0.5 rem (i.e., 387 mrem) whole-body dose equivalent. With 40 square feet of modified penetrations, the licensee could reasonably affect temporary repairs and terminate the release within a few hours, as was previously assumed. With an unrestricted modification, this assumption would not be valid, and potential offsite doses could exceed a small fraction of 10 CFR Part 100 guidelines.

The NRC evaluated the potential risks associated with modifying up to 40 square feet of penetrations between the containment and the auxiliary and fuel handling building. The NRC determined that this did not involve a significant increase in the probability or consequences of an accident from the previously evaluated analysis, create the possibility of a new accident or involve a significant reduction in the margin of safety. This action did not authorize an increase in effluents from the facility and fell within the bounds of activities previously described in the NRC's PEIS.

The NRC's safety evaluation concluded that if a total of 40 square feet of the penetrations to the auxiliary and fuel handling building were modified, the offsite dose consequences of the modeled accident would remain a small portion of the limits in 10 CFR Part 100. The NRC concurred with expansion of the limit to 40 square feet. However, the agency stated that any additional modifications would be evaluated on a case-by-case basis. The evaluation would consider: (●) the nature of the modification; (●) the duration of the modification; (●) any restriction of activities while the modification was in place; and (●) the potential for offsite dose consequences to exceed a small portion of the 10 CFR Part 100 limits.

11.8.3 Seismic Design Requirements for Structures Inside Containment Building

- ***Purpose***. To provide the basis for the exemption from the seismic design requirements of 10 CFR Part 50, Appendix A, General Design Criterion 2, for the TMI-2 containment building floors and D-rings [h] and all systems within the containment building, excluding the reactor vessel.

- ***Evaluation***. [133] The licensee's safety evaluation considered structures, systems, and components that were designed to remain functional for the safe-shutdown earthquake, as defined in Appendix A [134] to 10 CFR Part 100. The structures, systems, and components considered were those necessary to ensure that the following criteria were met: (●) integrity of the reactor coolant pressure boundary; (●) capability to shut down the reactor and maintain it in a safe-shutdown condition; and (●) capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the 10 CFR Part 100 [135] guidelines. The evaluation of each criterion is as follows:

  - *Pressure Boundary Integrity*. Since this request for exemption excluded the reactor vessel, the safety function of the first criterion was determined to be maintained. The licensee's safety evaluation report [136] for the removal of the reactor vessel head provided information related to the first criterion.

  - *Safe-Shutdown Condition*. The second criterion was addressed by the analysis in the licensee's safety evaluation report [137] for bulk defueling. The discussion in Section 4.2 of the report on criticality control attested that the safe-shutdown condition of the reactor, which existed at the time of the evaluation, would be maintained throughout the recovery operations. Since the reactor vessel was capable of providing criticality control, Criterion 2 applicability was limited to the reactor vessel and was satisfied in that regard.

---

[h] D-rings were the shield enclosures around the steam generator compartments; they were so named because of their shape.

- *Consequence of an Accident*. Adherence to the third criterion was addressed by an NRC safety evaluation [(138)] that postulated a scenario in which a seismic event caused the failure of containment building penetration(s) and a leak in the reactor coolant system (RCS). The leak in the RCS was assumed to continue until the water drained to the level of the reactor vessel nozzles. The resultant offsite dose estimates for the postulated RCS leak scenario were less than 10 percent of 10 CFR Part 20 [(139)] limits and were well within the exposure guidelines of 10 CFR Part 100. This scenario comprised the worst case reactor coolant pressure boundary leak and was therefore the limiting scenario.

- *Conclusion*. The licensee determined that it was not necessary for systems within the containment building, excluding the reactor vessel, to perform their safety functions during an earthquake. In addition, General Design Criterion 2 applied only to the reactor vessel. The licensee committed to performing a safety review pursuant to section 10 CFR 50.59, "Changes, tests and experiments," [(140)] to demonstrate that no unreviewed safety question existed before the performance of any work that had the potential to degrade the seismic adequacy of the containment building floors and D-rings.

- ***NRC Review***. Editor's Note: The NRC's safety evaluation could not be located.

11.9 Endnotes

Reference citations are exact filenames of documents on the Digital Versatile Discs (DVDs) to NUREG/KM-0001, Supplement 1, [(141)] "Three Mile Island Accident of 1979 Knowledge Management Digest: Recovery and Cleanup," issued June 2016, DVD document filenames start with a full date (YYY-MM-DD) or end with a partial date (YYYY-DD). Citations for the references that are not on a DVD begin with the author organization or name(s), with the document title in quotation marks. Contents of the DVDs are available at the Idaho National Laboratory Web site (https://tmi2kml.inl.gov).

---

[1] *U.S. Code of Federal Regulations*, "Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979," Appendix R, "Domestic Licensing of Production and Utilization Facilities," Part 50, Chapter I, Title 10, "Energy"
[2] GEND-034, Gross Decontamination Experiment Report (1983-07)
[3] *U.S. Code of Federal Regulations*, "General Design Criteria for Nuclear Power Plants," Appendix A, "Domestic Licensing of Production and Utilization Facilities," Part 50, Chapter I, Title 10, "Energy"
[4] USNRC, "Control of Heavy Loads at Nuclear Power Plants," NUREG-0612, July 1980 [Available at nrc.gov]
[5] USNRC, "Control of Heavy Loads at Nuclear Power Plants," NUREG-0612, July 1980 [Available at nrc.gov]
[6] (1985-04-16) GPU Safety Evaluation, Justifying Non-Seismic Design of TMI-2 Post-Accident Systems, Rev. 0

[7] USNRC, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Regulatory Guide 1.109, Rev. 1, October 1977 [Current and past regulatory guides are available at nrc.gov]

[8] USNRC, "Age-Specific Radiation Dose Commitment Factors for a One-Year Chronic Intake," NUREG-0172, November 1977 [Available at nrc.gov]

[9] USNRC, "Age-Dependent Dose-Conversion Factors for Selected Bone-Seeking Radionuclides," NUREG/CR-3535, May 1984 [Available at ntis.gov]

[10] International Commission on Radiological Protection, Publication 26, "Recommendations of the ICRP," 1977, Ottawa, Ontario, Canada

[11] *U.S. Code of Federal Regulations*, "Seismic and Geologic Siting Criteria for Nuclear Power Plants," Appendix A, Part 100, "Reactor Site Criteria," Chapter I, Title 10, "Energy"

[12] USNRC, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," NUREG-0800 (Formerly NUREG-75/087) [Available at nrc.gov]

[13] (1982-06-15) GPU Safety Evaluation, Axial Power Shaping Rod Insertion Test

[14] (1982-05-17) Order Amendment

[15] (1982-07-06) GPU Safety Evaluation, Insertion Camera Through Reactor Vessel Leadscrew Opening, Rev. 2

[16] (1982-07-13) NRC Review, Control Rod Drive Mechanism Quick Look Camera Inspection (re 07-06-1982) (2)

[17] (1989-08-18) GPU Safety Evaluation, Remove Metallurgical Samples from Reactor Vessel SER

[18] (1989-10-20) GPU Safety Evaluation, Remove Metallurgical Samples from Vessel, Rev. 1 (re 08-18-1989) (effective pages)

[19] (1986-05-15) GPU Safety Evaluation, Defueling SER, Rev. 10

[20] (1989-08-18) GPU Safety Evaluation, Remove Metallurgical Samples from Reactor Vessel SER

[21] (1989-10-20) GPU Safety Evaluation, Remove Metallurgical Samples from Vessel, Rev. 1 (re 08-18-1989) (effective pages)

[22] (1989-11-28) NRC Safety Evaluation, Reactor Vessel Lower Head Metallurgical Sampling (re 08-18-1989)

[23] (1985-10-29) NRC Safety Evaluation, Commencement of TMI-2 Preliminary Defueling Operations (re 10-24-1985)

[24] (1985-11-12) NRC Safety Evaluation, Early Defueling of the TMI-2 Reactor Vessel (10-10-1985)

[25] (1986-07-24) NRC Review, Core Region Bulk Defueling Operations Limited to the Core Region (re 05-15, 07-23-1986)

[26] (1985-02-08) GPU Safety Evaluation, Reactor Building Decontamination and Dose Reduction Activities for 1985, Rev. 1

[27] (1986-03-28) GPU Safety Evaluation, Reactor Building Decontamination and Dose Reduction Activities for 1986, Rev. 0

[28] (1986-03-28) GPU Safety Evaluation, Reactor Building Decontamination and Dose Reduction Activities for 1986, Rev. 0

[29] GEND-034, Gross Decontamination Experiment Report (1983-07)

[30] (1986-03-28) GPU Safety Evaluation, Reactor Building Decontamination and Dose Reduction Activities for 1986, Rev. 0

[31] (1984-02-03) NRC Safety Evaluation, Containment Decontamination and Dose Reduction (re 09-29, 01-23-1983)

[32] (1984-02-03) NRC Safety Evaluation, Containment Decontamination and Dose Reduction (re 09-29, 01-23-1983)

[33] (1983-02-18) GPU Safety Evaluation, Polar Crane Load Test

[34] (1983-11-18) NRC Safety Evaluation, Reactor Building Polar Crane Load Test

[35] (1984-03-09) GPU Safety Evaluation, Head Removal, Rev. 5

[36] (1984-03-09) GPU Safety Evaluation, Head Removal, Rev. 5

[37] (1984-07-17) NRC Safety Evaluation, Reactor Pressure Vessel Head Lift (re -3-09-1984)

[38] (1984-07-17) NRC Safety Evaluation, Reactor Pressure Vessel Head Lift (re -3-09-1984)

[39] USNRC, "Control of Heavy Loads at Nuclear Power Plants," NUREG-0612, July 1980 [Available at nrc.gov]

[40] (1986-06-02) GPU Safety Evaluation, Heavy Load Handling Inside Containment, Rev. 3

[41] (1983-03-15) GPU Response to NRC, Polar Crane SER

[42] (1986-06-02) GPU Safety Evaluation, Heavy Load Handling Inside Containment, Rev. 3

[43] (1985-05-02) NRC Safety Evaluation, Heavy Load Handling Over the TMI-2 Reactor Vessel (re 04-19-1985)

[44] (1985-04-19) GPU Safety Evaluation, Heavy Load Handling Over Reactor Vessel SER

[45] (1985-04-19) GPU Safety Evaluation, Heavy Load Handling Over Reactor Vessel SER

[46] (1984-06-18) GPU Safety Evaluation, Preparatory Activities for Plenum Assembly Removal

[47] American Society of Mechanical Engineers, *Boiler and Pressure Vessel Code*, 1974 Edition, Section III, "Rules for Construction of Nuclear Facility Components," New York, NY.

[48] (1984-07-31) NRC Review, Preparatory Activities for Plenum Assembly Removal SER (re 06-18-1984)

[49] (1984-09-14) NRC Safety Evaluation, Plenum Assembly Removal (re various letters)

[50] (1985-05-01) GPU Safety Evaluation, Plenum Removal and Transfer, Rev. 3

[51] (1985-05-07) NRC Safety Evaluation, Lifting, Transfer, and Storage of Plenum Assembly (re various letters)

[52] (1984-03-09) GPU Safety Evaluation, Head Removal, Rev. 5

[53] (1987-04-02) GPU, Safety Evaluation, Canister Handling and Preparation for Shipment SER, Rev. 4 (Effective Pages Only)

[54] American Society of Testing and Materials, E119, "Standard Test Methods for Fire Tests of Building Construction and Materials," Philadelphia, PA [Revision not provided]

[55] (1987-04-02) GPU, Safety Evaluation, Canister Handling and Preparation for Shipment SER, Rev. 4 (Effective Pages Only)

[56] (1987-04-02) GPU, Safety Evaluation, Canister Handling and Preparation for Shipment SER, Rev. 4 (Effective Pages Only)

[57] Pacific Nuclear Systems, Inc., "Safety Analysis Report for the NuPac 125-B Fuel Shipping Cask," Rev. 0, June 1985 [ADAMS Accession No. ML20132E700]

[58] (1986-06-20) NRC Safety Evaluation, Canister Handling and Preparation for Shipment Program (re 02-17, 06-11-1986)

[59] (1988-03-03) NRC Review, Canister Handling and Preparation for Shipment Program TER (re various letters)

[60] (1986-08-27) GPU, Plan to Use Plasma Arc Torch to Cut Upper End Fittings (re 08-18-1986)

[61] (1986-08-27) GPU, Plan to Use Plasma Arc Torch to Cut Upper End Fittings (re 08-18-1986)

[62] (1986-12-02) NRC Review, Use of Plasma Arc Torch to Cut Upper End Fittings (re 08-27-1986)

[63] (1987-06-25) GPU, Off-Gas Releases Generated During Plasma Arc Cutting (re 05-07-1987)

[64] (1986-08-27) GPU, Plan to Use Plasma Arc Torch to Cut Upper End Fittings (re 08-18-1986)

[65] (1987-08-20) NRC Review, Use of Plasma Arc Torch to Cut Upper End Fittings (re various letters)

[66] (1986-12-02) NRC Review, Use of Plasma Arc Torch to Cut Upper End Fittings (re 08-27-1986)
[67] (1987-01-20) GPU Response to NRC, Plasma Arc Cutting
[68] (1987-03-20) NRC Review, Use of Plasma Arc Cutting of Fuel and Fittings (re 12-02-1986, 01-20-1987)
[69] (1987-05-07) GPU Response to NRC, Plasma Arc Cutting (re 03-20-1987, 08-27-1986)
[70] (1987-06-25) GPU, Off-Gas Releases Generated During Plasma Arc Cutting (re 05-07-1987)
[71] (1987-08-20) NRC Review, Use of Plasma Arc Torch to Cut Upper End Fittings (re various letters)
[72] (1989-02-08) GPU Safety Evaluation, Use of Air as Secondary Gas for Plasma Arc Torch
[73] (1987-05-07) GPU Response to NRC, Plasma Arc Cutting (re 03-20-1987, 08-27-1986)
[74] (1987-01-20) GPU Response to NRC, Plasma Arc Cutting
[75] (1988-04-01) NRC Safety Evaluation, Lower Core Support Assembly Defueling (re 01-18-1988)
[76] (1987-08-20) NRC Review, Use of Plasma Arc Torch to Cut Upper End Fittings (re various letters)
[77] (1989-01-10) GPU Safety Evaluation, Sediment Transfer and Processing Operations, Rev. 4
[78] American National Standards Institute, B31.1, “Power Piping,” New York, NY [Revision not provided]
[79] (1986-09-25) NRC Review, Sediment Transfer and Processing Operations (re 03-18-1985)
[80] (1986-03-14) GPU Safety Evaluation, Decontamination Using Ultrahigh Pressure Water Flush, Rev. 0
[81] (1986-03-14) GPU Safety Evaluation, Decontamination Using Ultrahigh Pressure Water Flush, Rev. 0
[82] (1986-07-16) NRC Safety Evaluation, Decontamination using Ultrahigh Pressure Water Flush (re 03-14-1986)
[83] (1985-08-08) Order Amendment (also exemption from 10CFR50 AppA)
[84] (1985-10-10) GPU Safety Evaluation, Early Defueling of the TMI-2 Reactor Vessel, Rev. 4
[85] (1986-05-15) GPU Safety Evaluation, Defueling SER, Rev. 10
[86] (1986-05-20) GPU Safety Evaluation, Defueling SER, Rev. 10 (Correction)
[87] (1985-10-10) GPU Safety Evaluation, Early Defueling of the TMI-2 Reactor Vessel, Rev. 4
[88] (1986-05-15) GPU Safety Evaluation, Defueling SER, Rev. 10
[89] (1986-05-20) GPU Safety Evaluation, Defueling SER, Rev. 10 (Correction)
[90] (1985-11-12) NRC Safety Evaluation, Early Defueling of the TMI-2 Reactor Vessel (10-10-1985)
[91] (1985-10-10) GPU Safety Evaluation, Early Defueling of the TMI-2 Reactor Vessel, Rev. 4
[92] (1985-10-10) GPU Safety Evaluation, Early Defueling of the TMI-2 Reactor Vessel, Rev. 4
[93] (1986-07-24) NRC Review, Core Region Bulk Defueling Operations Limited to the Core Region (re 05-15, 07-23-1986)
[94] (1985-11-12) NRC Safety Evaluation, Early Defueling of the TMI-2 Reactor Vessel (10-10-1985)
[95] (1988-01-18) GPU Safety Evaluation, Lower Core Support Assembly Defueling, Rev. 2
[96] (1986-08-27) GPU, Plan to Use Plasma Arc Torch to Cut Upper End Fittings (re 08-18-1986)
[97] (1988-01-18) GPU Safety Evaluation, Lower Core Support Assembly Defueling, Rev. 2
[98] (1986-05-15) GPU Safety Evaluation, Defueling SER, Rev. 10
[99] (1988-01-18) GPU Safety Evaluation, Lower Core Support Assembly Defueling, Rev. 2
[100] (1986-08-27) GPU, Plan to Use Plasma Arc Torch to Cut Upper End Fittings (re 08-18-1986)
[101] (1988-01-18) GPU Safety Evaluation, Lower Core Support Assembly Defueling, Rev. 2
[102] (1986-05-15) GPU Safety Evaluation, Defueling SER, Rev. 10

[103] (1988-04-01) NRC Safety Evaluation, Lower Core Support Assembly Defueling (re 01-18-1988)

[104] (1988-06-06) GPU Safety Evaluation, Completion of Lower Core Support and Lower Head Defueling, Rev. 0

[105] (1988-01-18) GPU Safety Evaluation, Lower Core Support Assembly Defueling, Rev. 2

[106] (1988-04-01) NRC Safety Evaluation, Lower Core Support Assembly Defueling (re 01-18-1988)

[107] (1988-06-06) GPU Safety Evaluation, Completion of Lower Core Support and Lower Head Defueling, Rev. 0

[108] (1986-05-15) GPU Safety Evaluation, Defueling SER, Rev. 10

[109] (1988-06-06) GPU Safety Evaluation, Completion of Lower Core Support and Lower Head Defueling, Rev. 0

[110] (1988-06-06) GPU Safety Evaluation, Completion of Lower Core Support and Lower Head Defueling, Rev. 0

[111] (1986-05-15) GPU Safety Evaluation, Defueling SER, Rev. 10

[112] (1988-12-01) NRC Safety Evaluation, Lower Core Support and Lower Head Defueling (re 06-06-1988)

[113] (1988-09-15) GPU Safety Evaluation, Completion of Upper Support Core Assembly Defueling, Rev. 0

[114] (1988-01-18) GPU Safety Evaluation, Lower Core Support Assembly Defueling, Rev. 2

[115] (1988-09-15) GPU Safety Evaluation, Completion of Upper Support Core Assembly Defueling, Rev. 0

[116] (1986-05-15) GPU Safety Evaluation, Defueling SER, Rev. 10

[117] (1988-09-15) GPU Safety Evaluation, Completion of Upper Support Core Assembly Defueling, Rev. 0

[118] (1986-05-15) GPU Safety Evaluation, Defueling SER, Rev. 10

[119] (1989-04-04) NRC Safety Evaluation, Completion of Upper Core Support Assembly Defueling (re 9-15-1988)

[120] NUREG-0591, Environmental Assessment, Use of EPICOR-II at TMI-2 (1979-08)

[121] NUREG-0683, Vol. 1, PEIS-Decontamination and Disposal of Radioactive Wastes Resulting from TMI-2 (1981-03)

[122] (1981-03-11) GPU Technical Evaluation, SDS, Revised

[123] NUREG-0796, Operation of the Submerged Demineralizer System at TMI-2 (1981-06)

[124] (1985-04-16) GPU Safety Evaluation, Justifying Non-Seismic Design of TMI-2 Post-Accident Systems, Rev. 0

[125] (1986-01-16) NRC Review, Seismic Design Requirements to Post Accident Systems (re 04-16, 07-29, 09-27-1985)

[126] *U.S. Code of Federal Regulations*, “Seismic and Geologic Siting Criteria for Nuclear Power Plants,” Appendix A, Part 100, “Reactor Site Criteria,” Chapter I, Title 10, “Energy”

[127] *U.S. Code of Federal Regulations*, “General Design Criteria for Nuclear Power Plants,” Appendix A, Part 50, “Domestic Licensing of Production and Utilization Facilities,” Chapter I, Title 10, “Energy”

[128] (1984-07-17) NRC Exempt, 10CFR50, App. A, Regarding Design of Containment Penetrations After Reactor Head Removal

[129] (1984-11-05) NRC Exempt, (07-17-1984) Clarification

[130] (1986-06-30) GPU, Seismic Design Criteria for Modified Containment Penetration

[131] (1985-04-16) GPU Safety Evaluation, Justifying Non-Seismic Design of TMI-2 Post-Accident Systems, Rev. 0

[132] (1987-04-03) NRC Safety Evaluation, Seismic Design Criteria for Modified Containment Penetrations (re various letters)

[133] (1988-09-27) GPU Request Exemption, Seismic Design Requirements Inside Reactor Building

[134] *U.S. Code of Federal Regulations*, "Seismic and Geologic Siting Criteria for Nuclear Power Plants," Appendix A, Part 100, "Reactor Site Criteria," Chapter I, Title 10, "Energy"

[135] *U.S. Code of Federal Regulations*, "Reactor Site Criteria," Part 100, Chapter I, Title 10, "Energy"

[136] (1984-03-09) GPU Safety Evaluation, Head Removal, Rev. 5

[137] (1986-05-15) GPU Safety Evaluation, Defueling SER, Rev. 10

[138] (1984-11-05) NRC Exempt, (07-17-1984) Clarification

[139] *U.S. Code of Federal Regulations*, "Standards for Protection gainst Radiation," Part 20, Chapter I, Title 10, "Energy"

[140] *U.S. Code of Federal Regulations*, "Changes, tests, and experiments," Section 50.59, Part 50, "Domestic Licensing of Production and Utilization Facilities," Chapter I, Title 10, "Energy"

[141] USNRC, "Three Mile Island Accident of 1979 Knowledge Management Digest: Recovery and Cleanup," NUREG/KM-0001, Supplement 1, June 2016 [Available at nrc.gov]

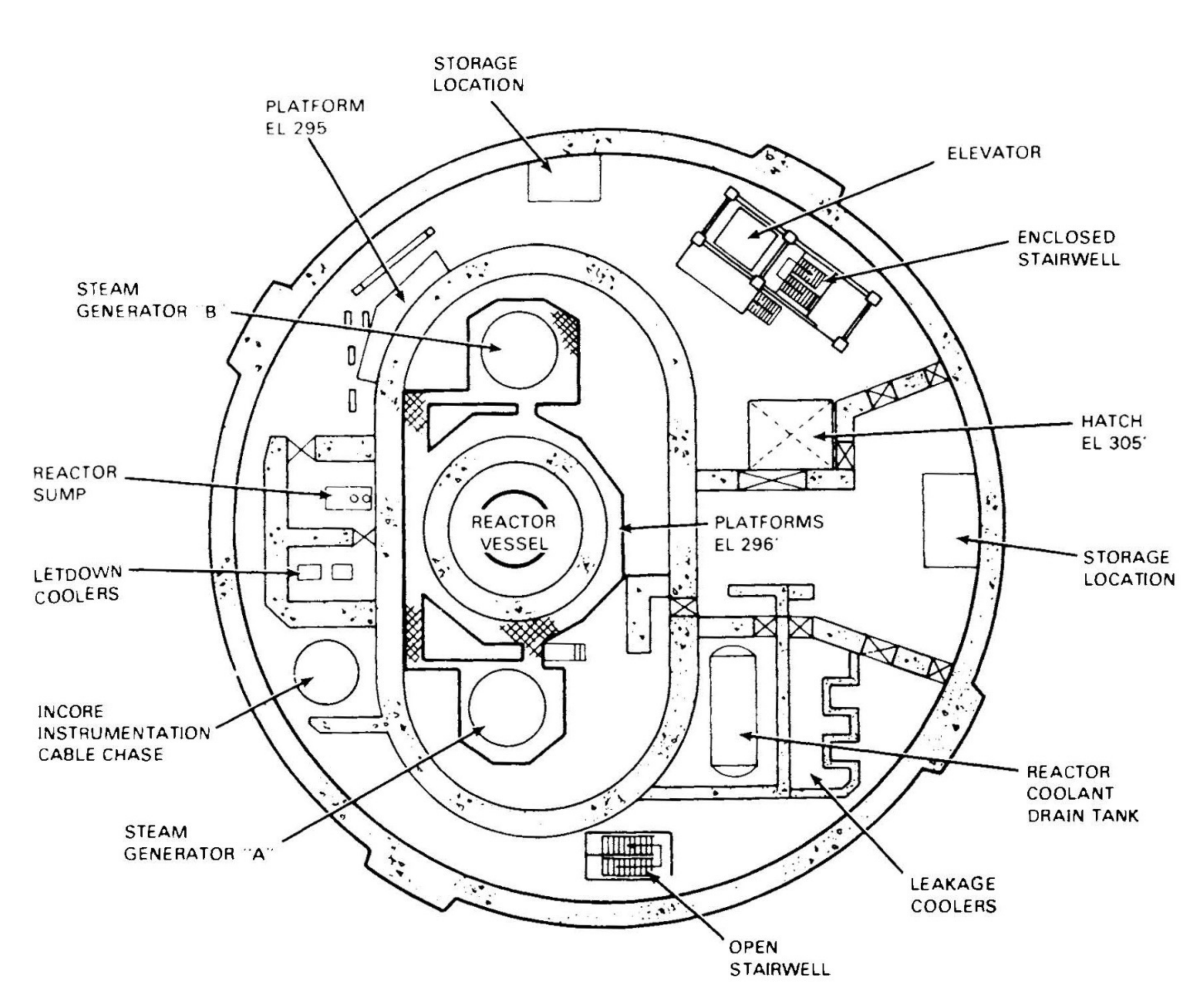
APPENDIX DRAWINGS

TABLE OF CONTENTS

Note: Unless otherwise noted, cited sources are available on the Digital Versatile Discs (DVDs) to NUREG/KM 0001, Supplement 1, “Three Mile Island Accident of 1979 Knowledge Management Digest: Recovery and Cleanup,” issued June 2016. Contents of the DVDs are available at the Idaho National Laboratory Web site (https://tmi2kml.inl.gov).

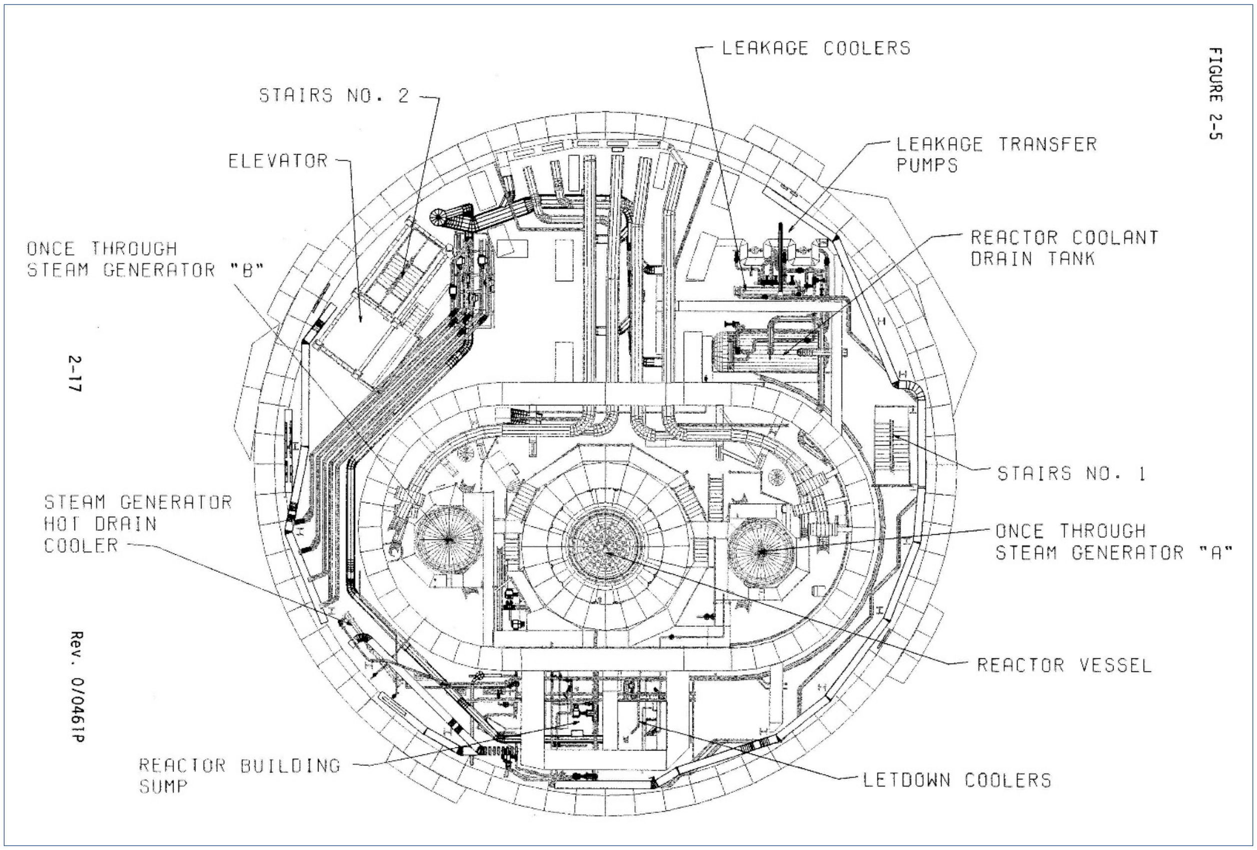
A.1 Containment Building

A.1.1 Containment Building 282-Foot Elevation (Basement Floor)



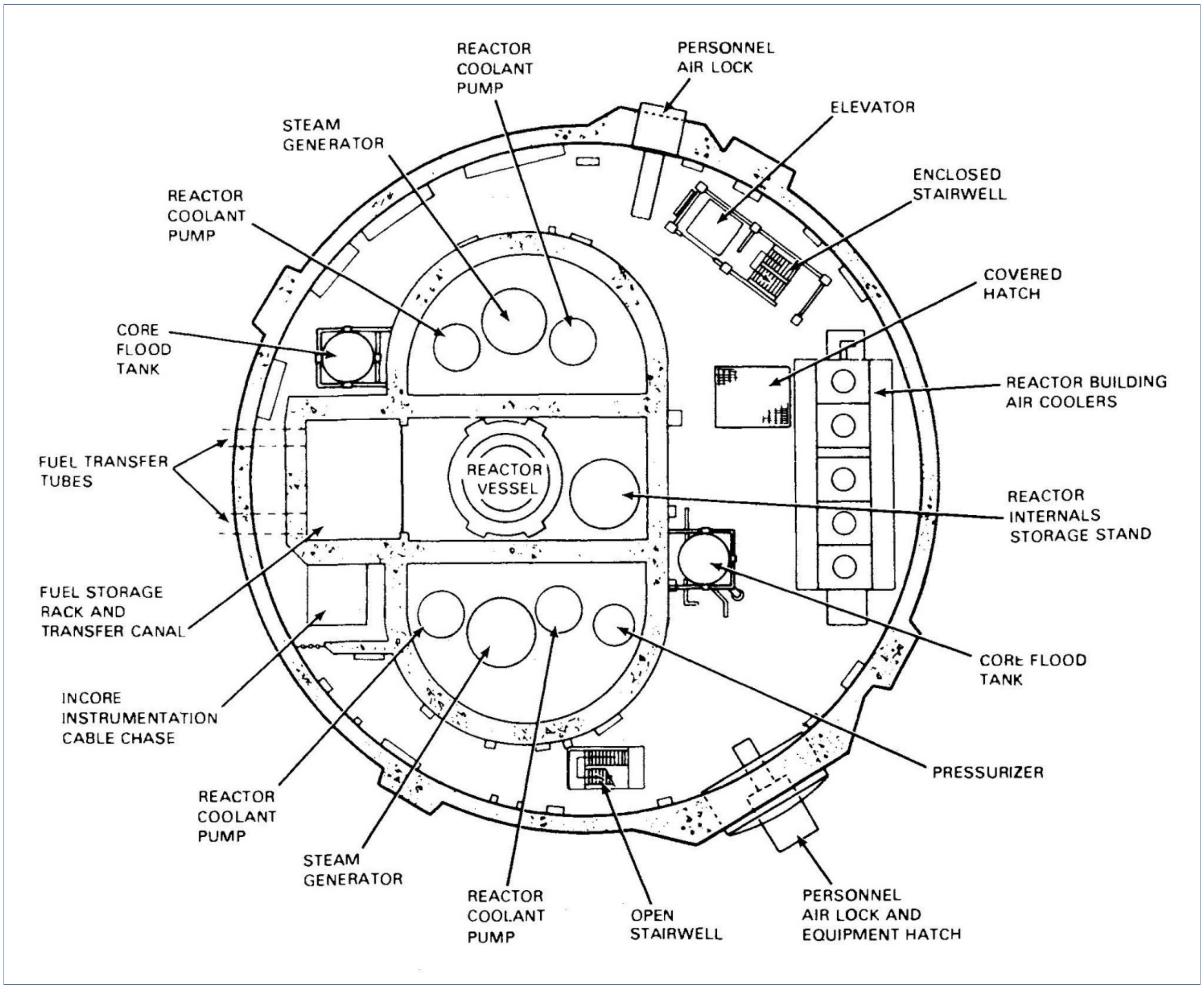
Source: NUREG-0683, SUPP. 1, PEIS, Occupational Radiation Dose (1984-10)

A.1.2 Containment Building Basement Floor (Detailed)



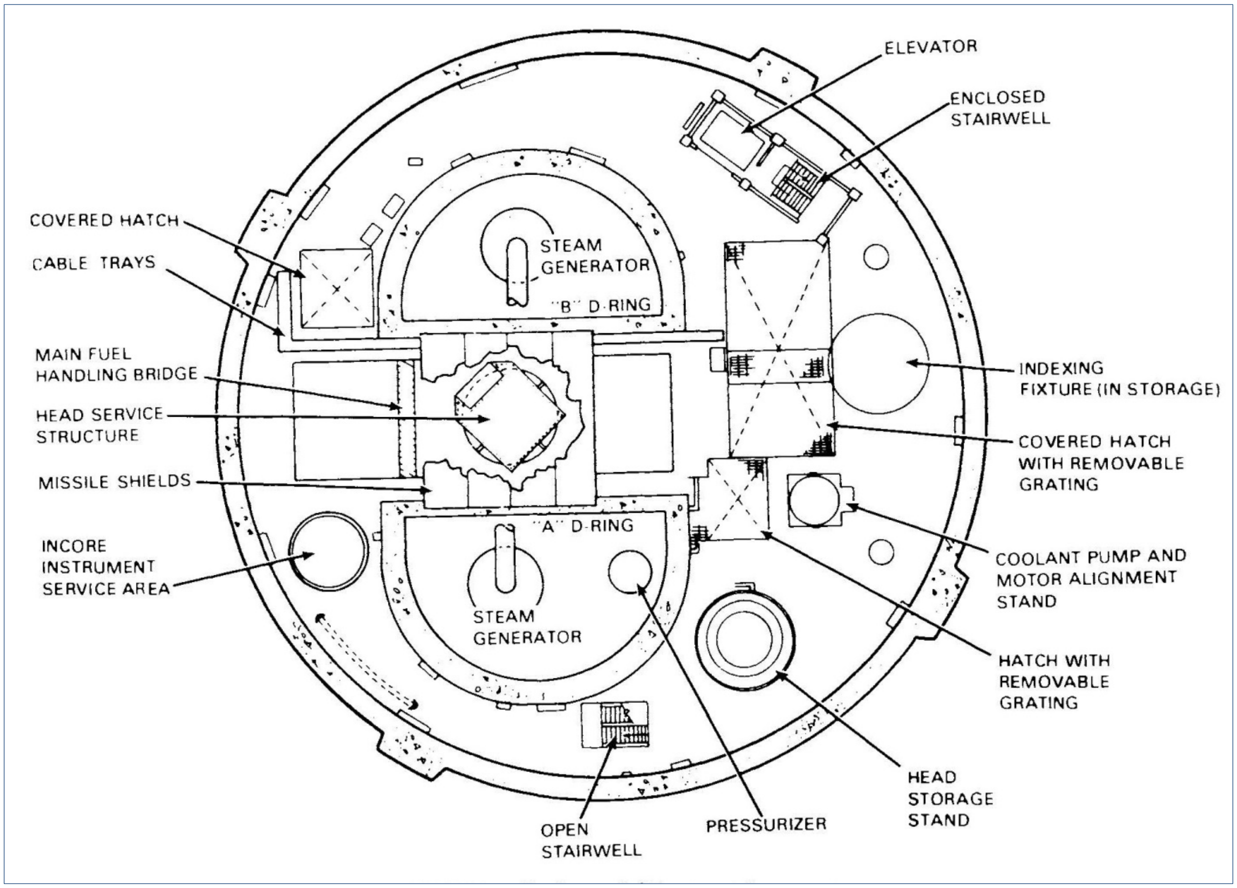
Source: (1990-02-22) GPU, Defueling Completion Report, Final Submittal

A.1.3 Containment Building 305-Foot Elevation (First Floor, Entry Level)



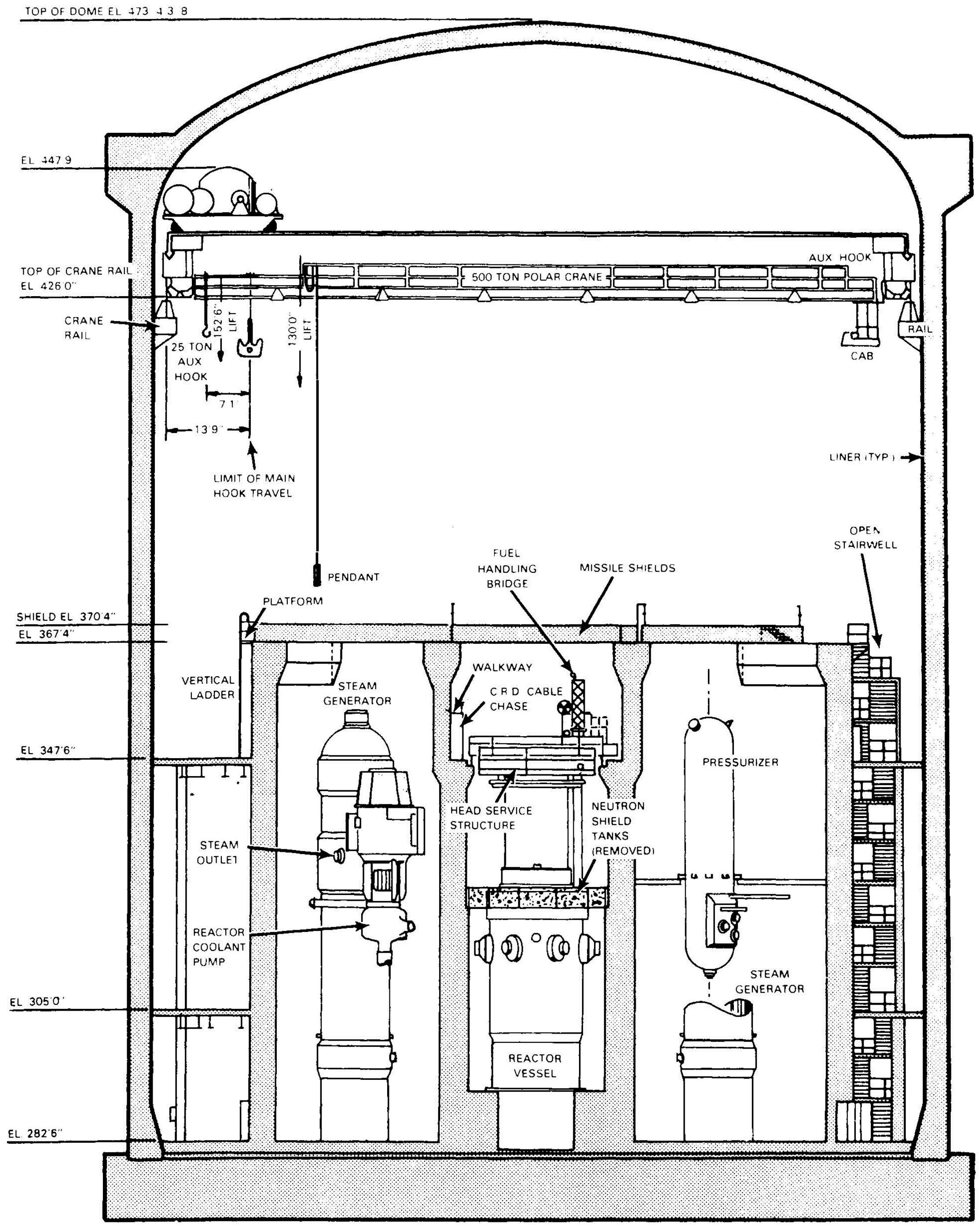
Source: NUREG-0683, SUPP. 1, PEIS, Occupational Radiation Dose (1984-10)

A.1.4 Containment Building 347-Foot Elevation (Second Floor)



Source: NUREG-0683, SUPP. 1, PEIS, Occupational Radiation Dose (1984-10)

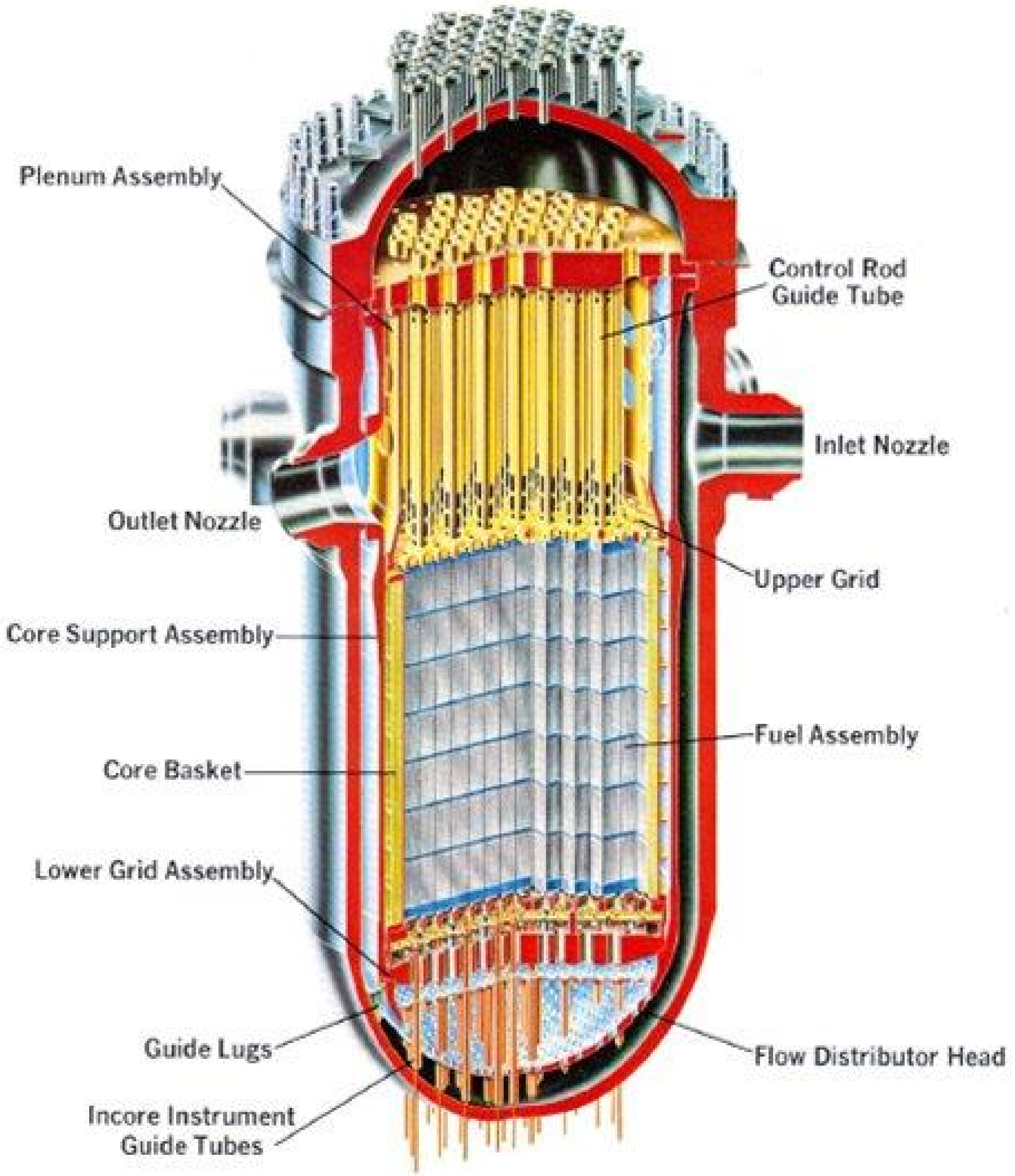
A.1.5 Containment Building (Cross Section)



Source: NUREG-0683, SUPP. 1, PEIS, Occupational Radiation Dose (1984-10)

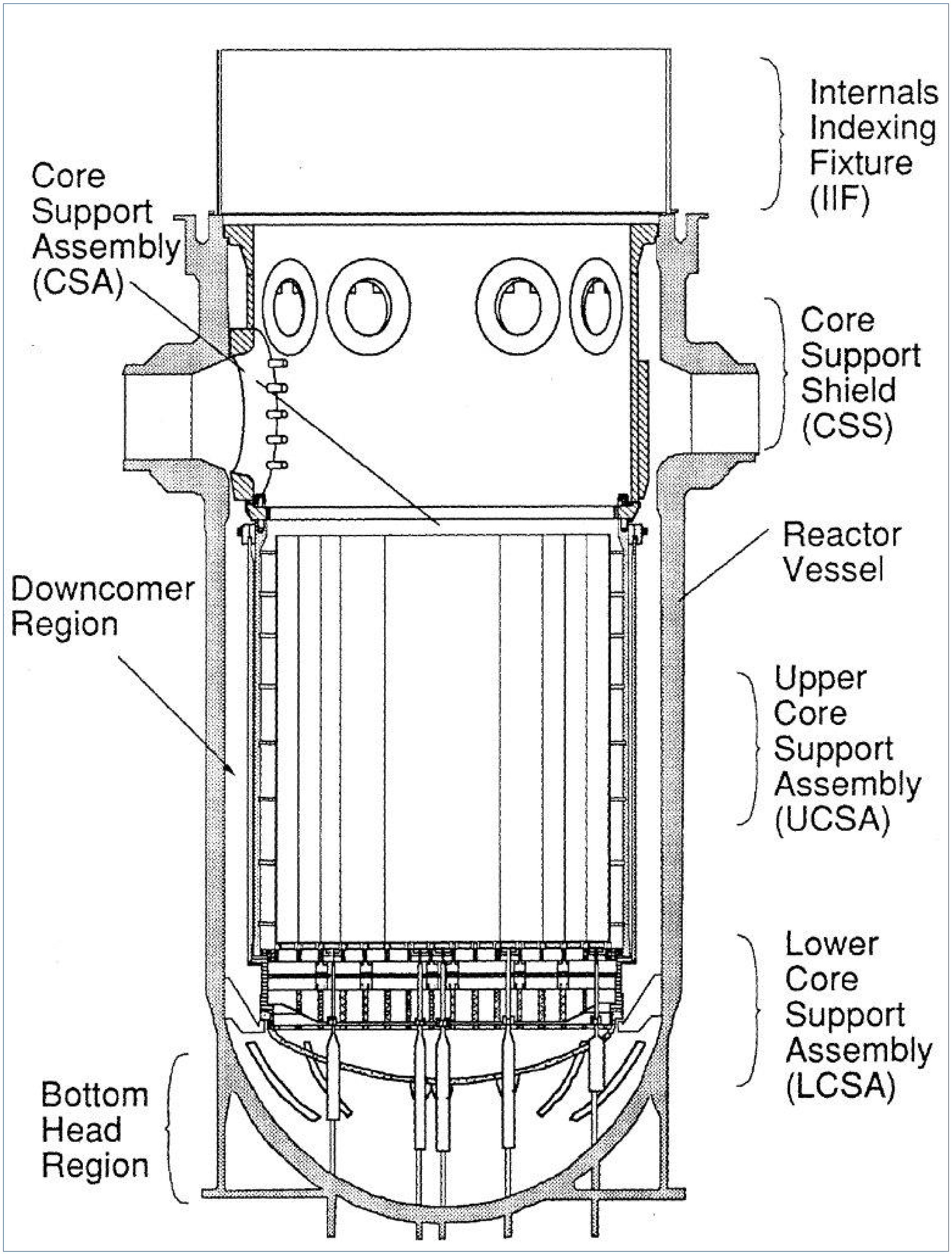
A.2 Reactor Vessel Internals

A.2.1 Reactor Vessel (Cross Section, With Fuel)



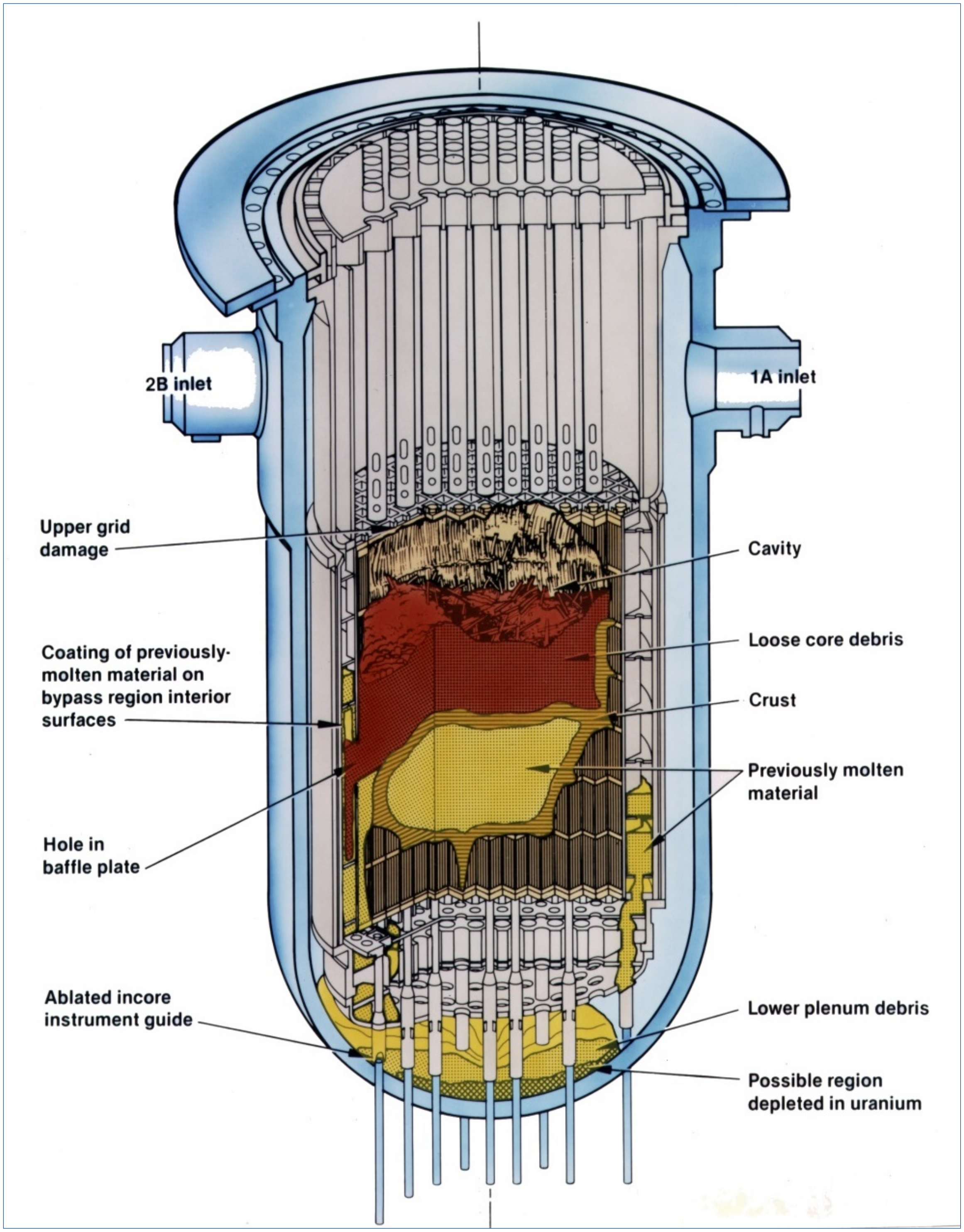
Source: Babcock and Wilcox training aid from NUREG/KM-0001, Three Mile Island Accident of 1979 Knowledge Management Digest, Supplement 1, Recovery and Cleanup.

A.2.2 Reactor Vessel (Cross Section, Without Fuel)



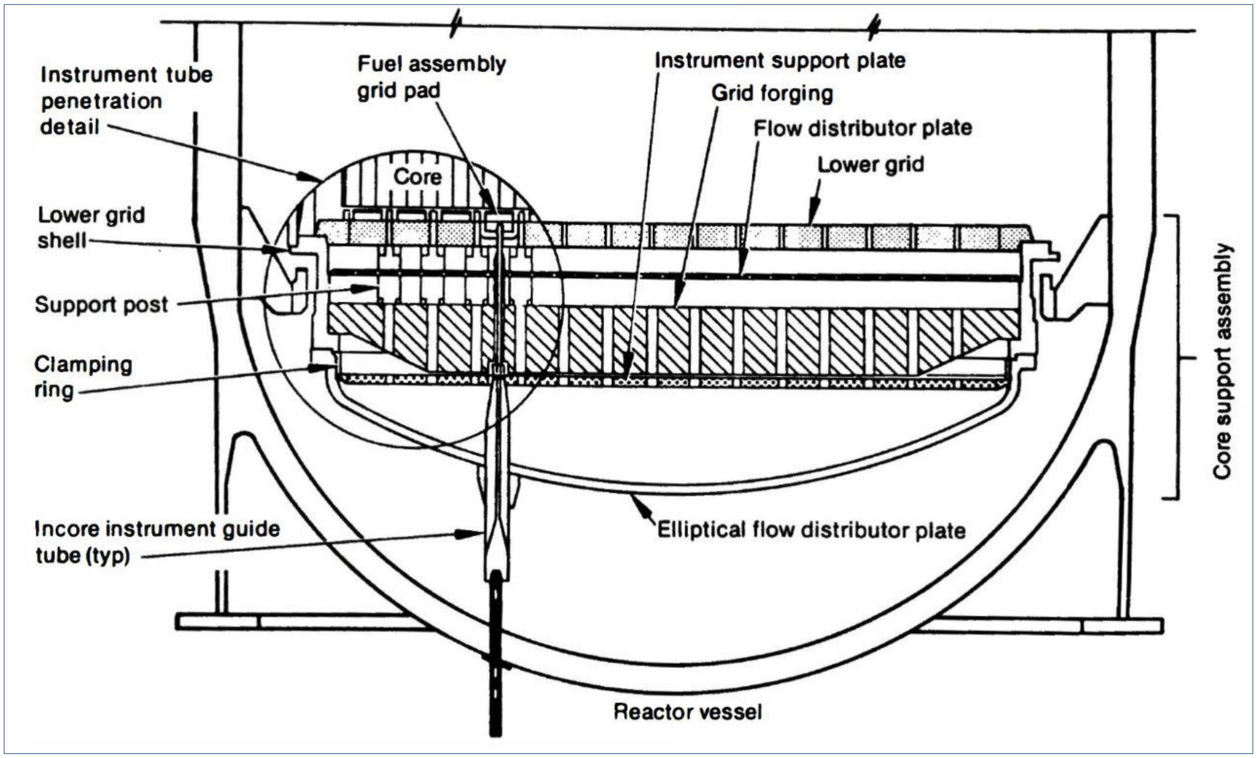
Source: (1990-02-22) GPU, Defueling Completion Report, Final Submittal

A.2.3 Reactor Vessel Showing Core Damage End State



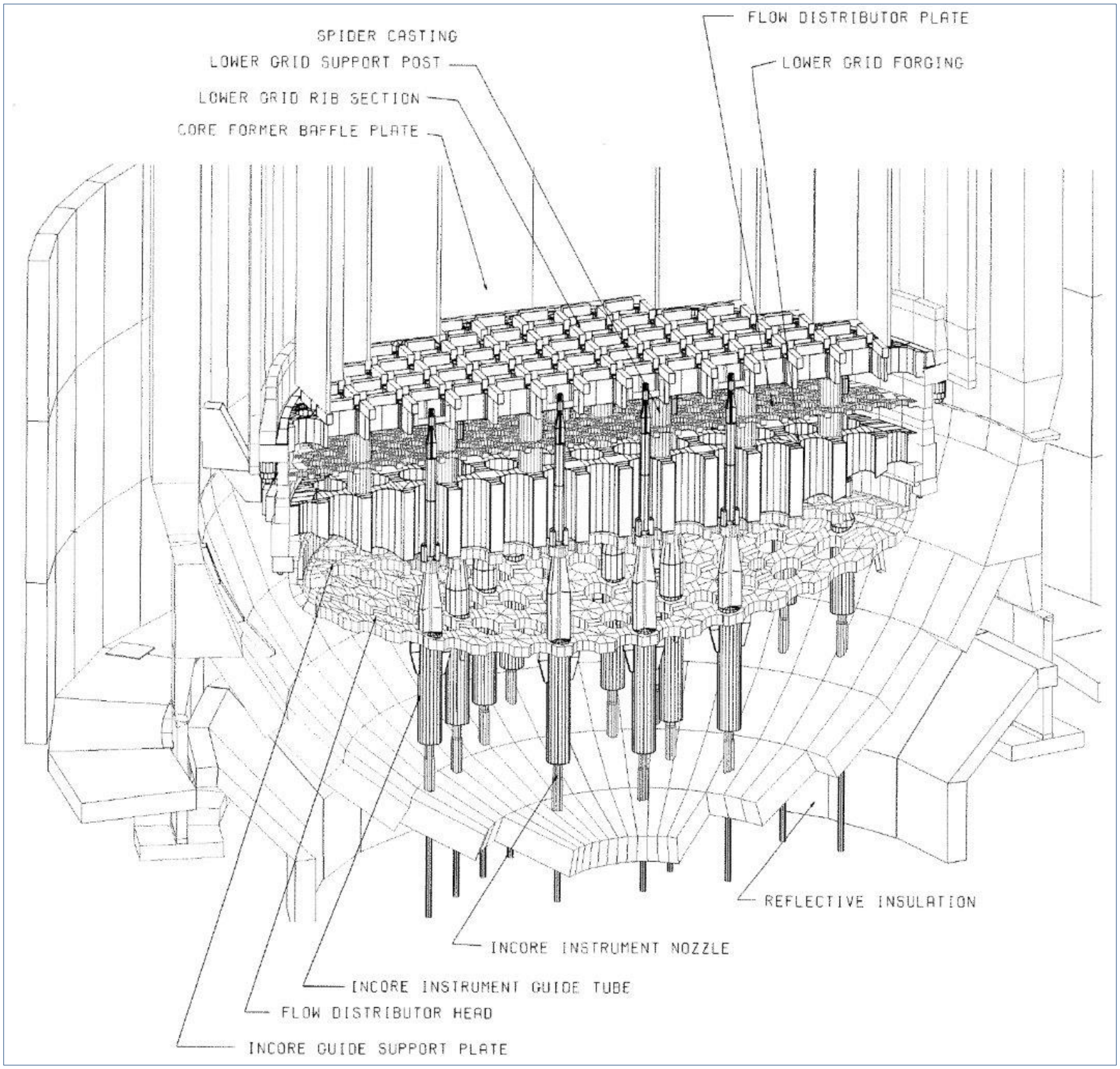
Source: EGG-TMI-7489, TMI-2 Accident Scenario Update (1986-12)

A.2.4 Reactor Vessel Lower Core Support Assembly (Cross Section)



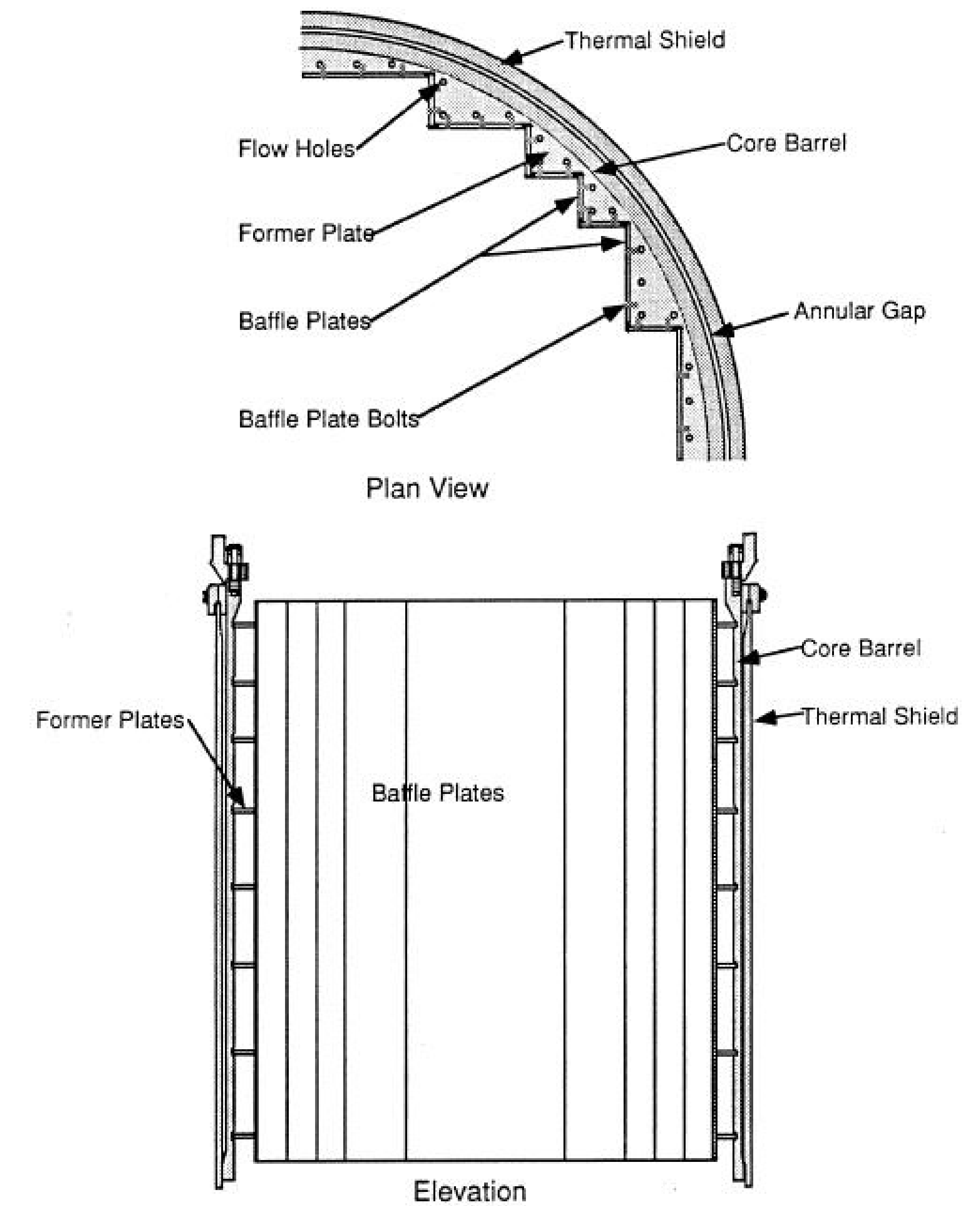
Source: EGG-TMI-7429, TMI-2 Lower Plenum Video Data Summary (1987-07)

A.2.5 Reactor Vessel Lower Core Support Assembly (Isometric Cross Section)



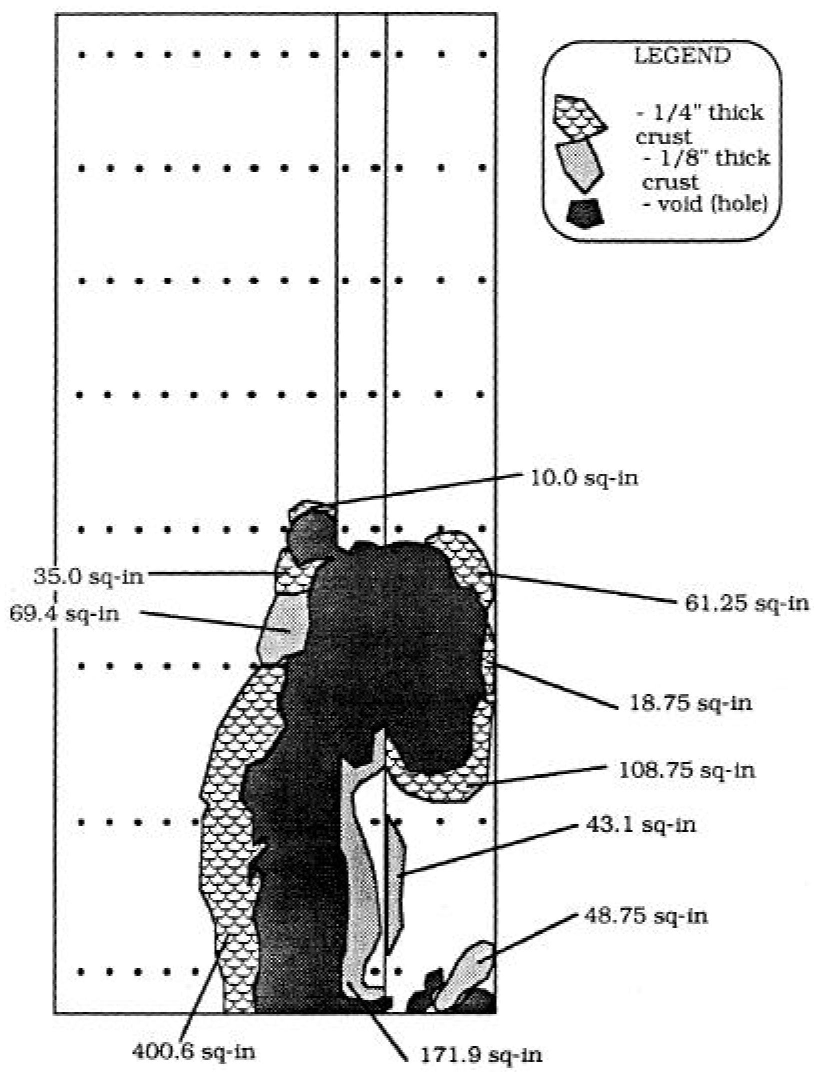
Source: (1990-02-22) GPU, Defueling Completion Report, Final Submittal

A.2.6 Reactor Vessel Upper Core Support Assembly



Source: (1990-02-22) GPU, Defueling Completion Report, Final Submittal

A.2.7 Reactor Vessel Baffle Plate Damage

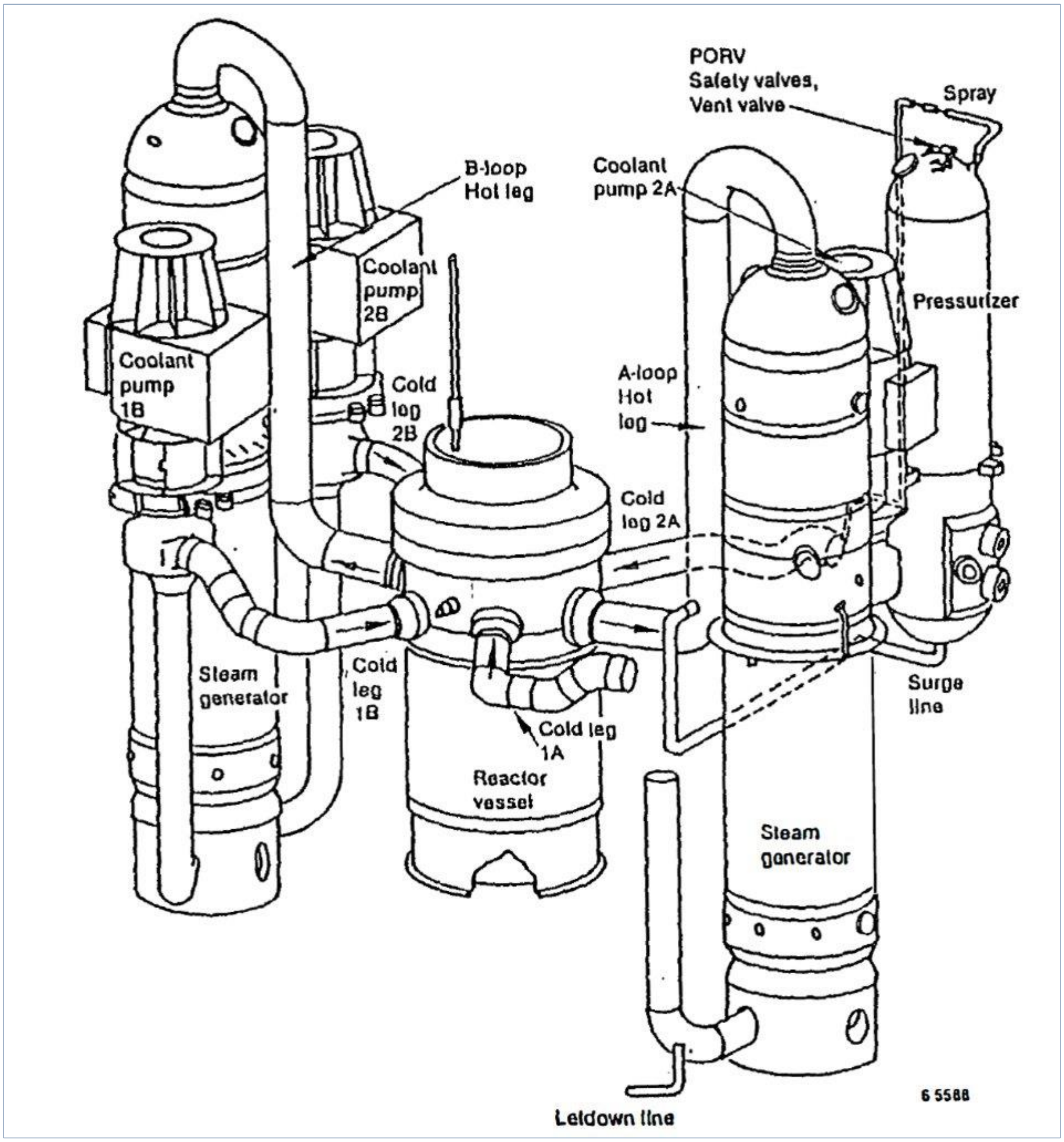


Melt Damage Near R-6 Grid Location

Source: (1990-02-22) GPU, Defueling Completion Report, Final Submittal

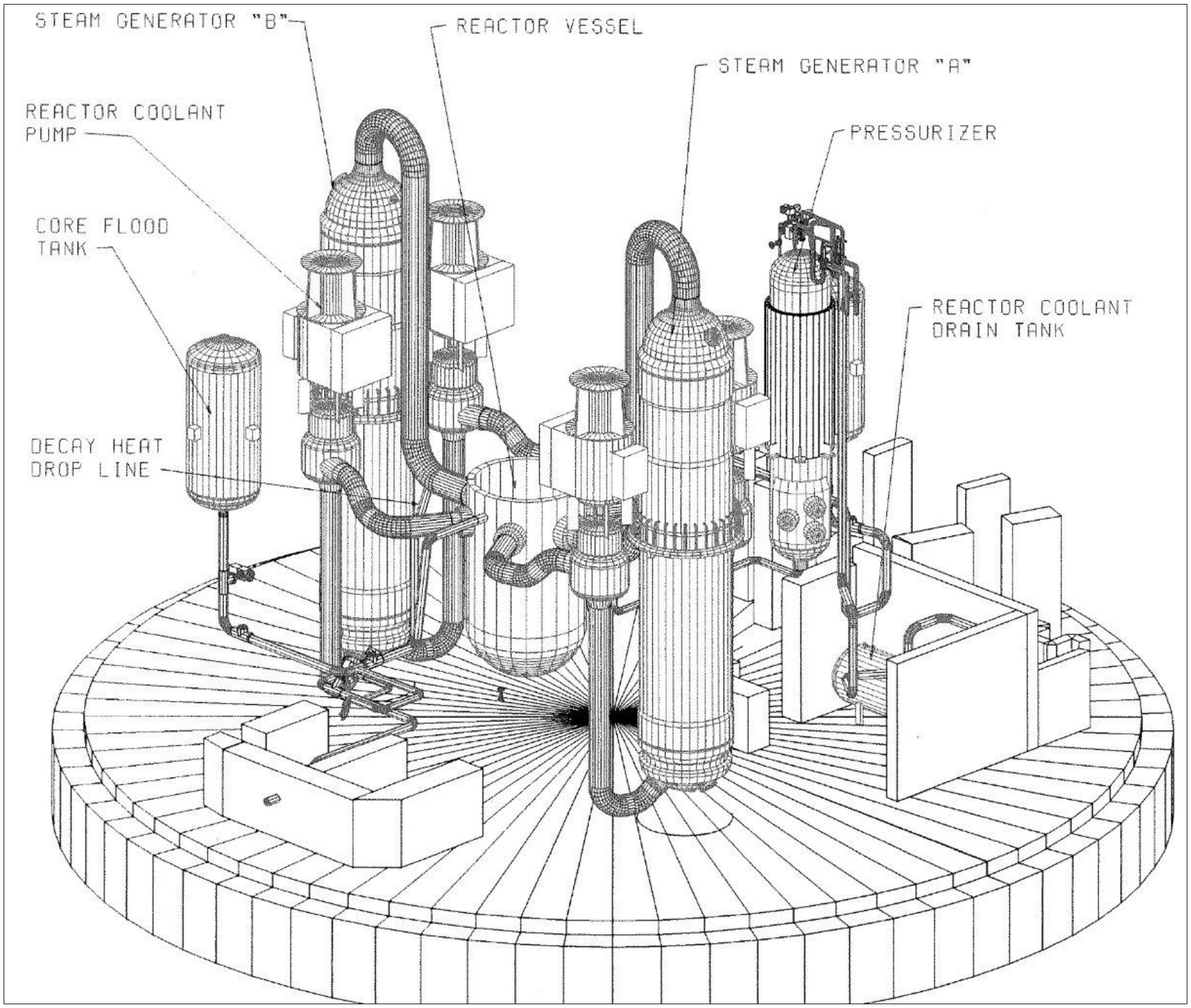
A.3 Reactor Coolant System

A.3.1 Reactor Coolant System (Isometric)



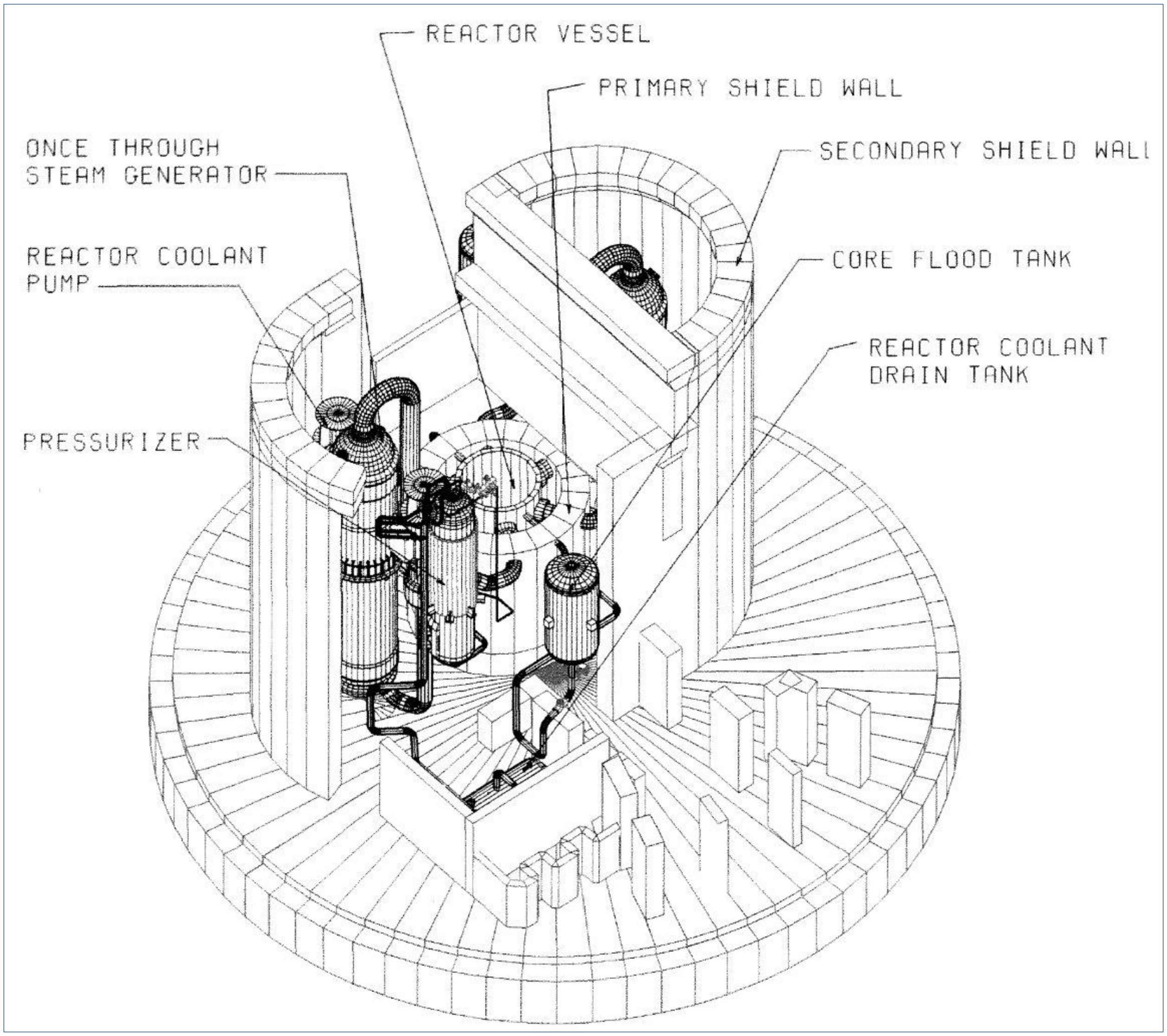
Source: NUREGCR-6197, TMI-2 Vessel Investigation Project Integration Report (1994-03)

A.3.2 Reactor Coolant System (Isometric View from Floor)



Source: (1990-02-22) GPU, Defueling Completion Report, Final Submittal

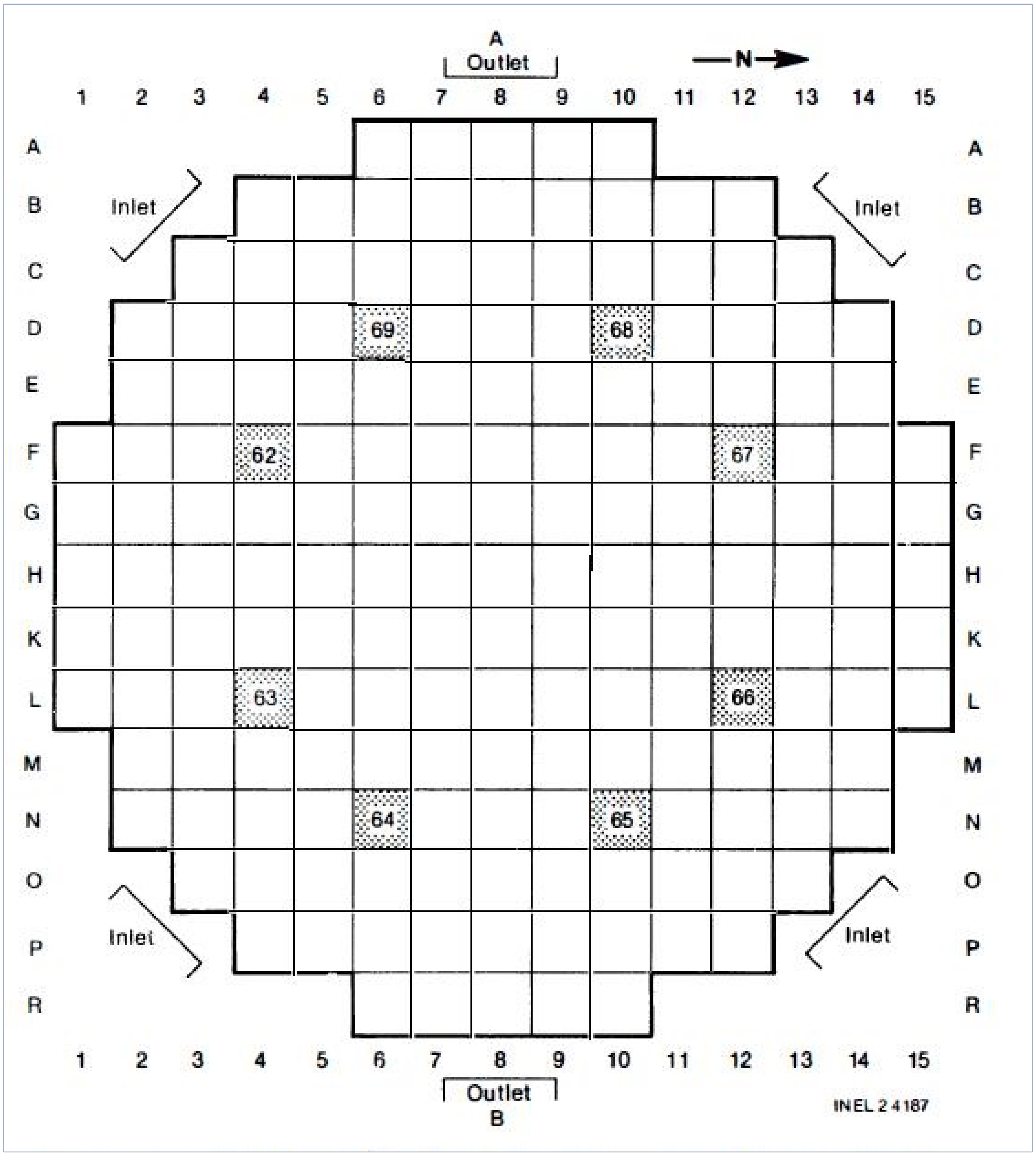
A.3.3 Reactor Coolant System (Isometric View Showing D-Rings)



Source: (1990-02-22) GPU, Defueling Completion Report, Final Submittal

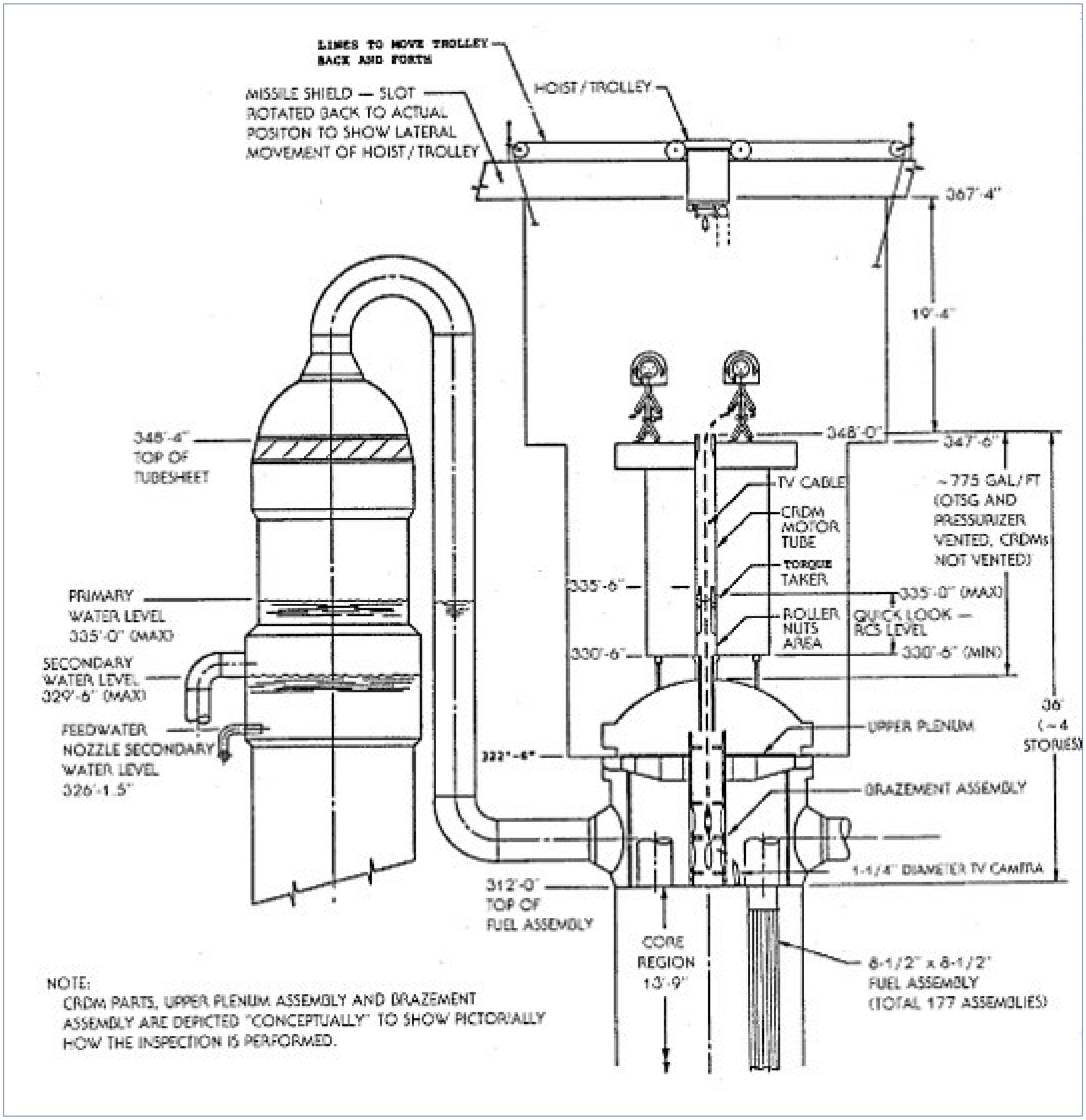
A.4 Data Collection Activities

A.4.1 Axial Power Shaping Rod Insertion Test (Rod Locations)



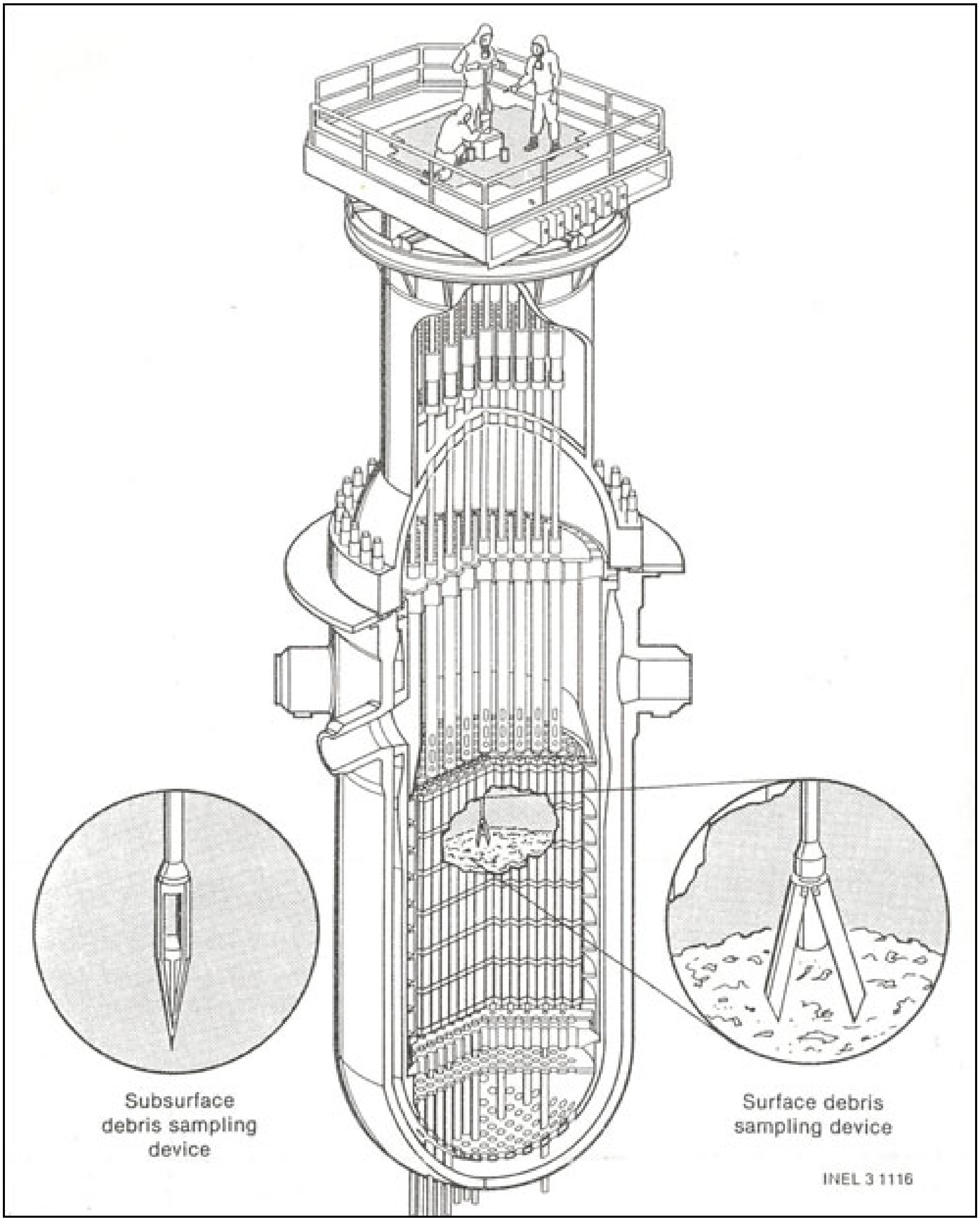
Source: GEND-INF-038, Assessment of the TMI-2 Axial Power Shaping Rod Dynamic Test Results (1983-04)

A.4.2 Camera Insertion Through Reactor Vessel Leadscrew Opening (Quick Look)



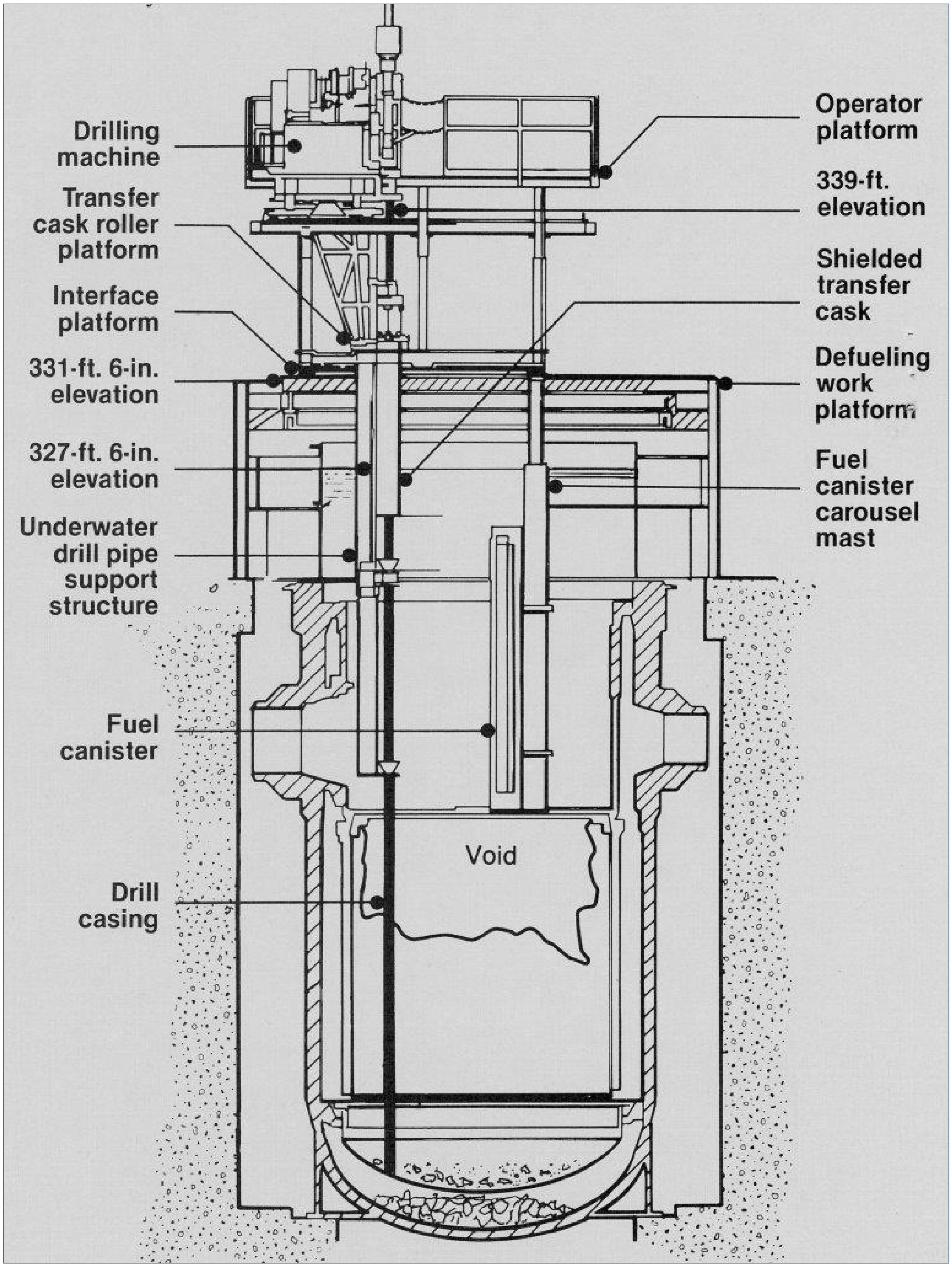
Source: Cole, Normal M., "Preparation for the Cleanup: Assessing the Damage," American Nuclear Society Executive Conferrence, "TMI-2: A Learning Experience," October 13-16, 1985, Hershey, PA

A.4.3 Reactor Vessel Underhead Characterization (Core Sampling)



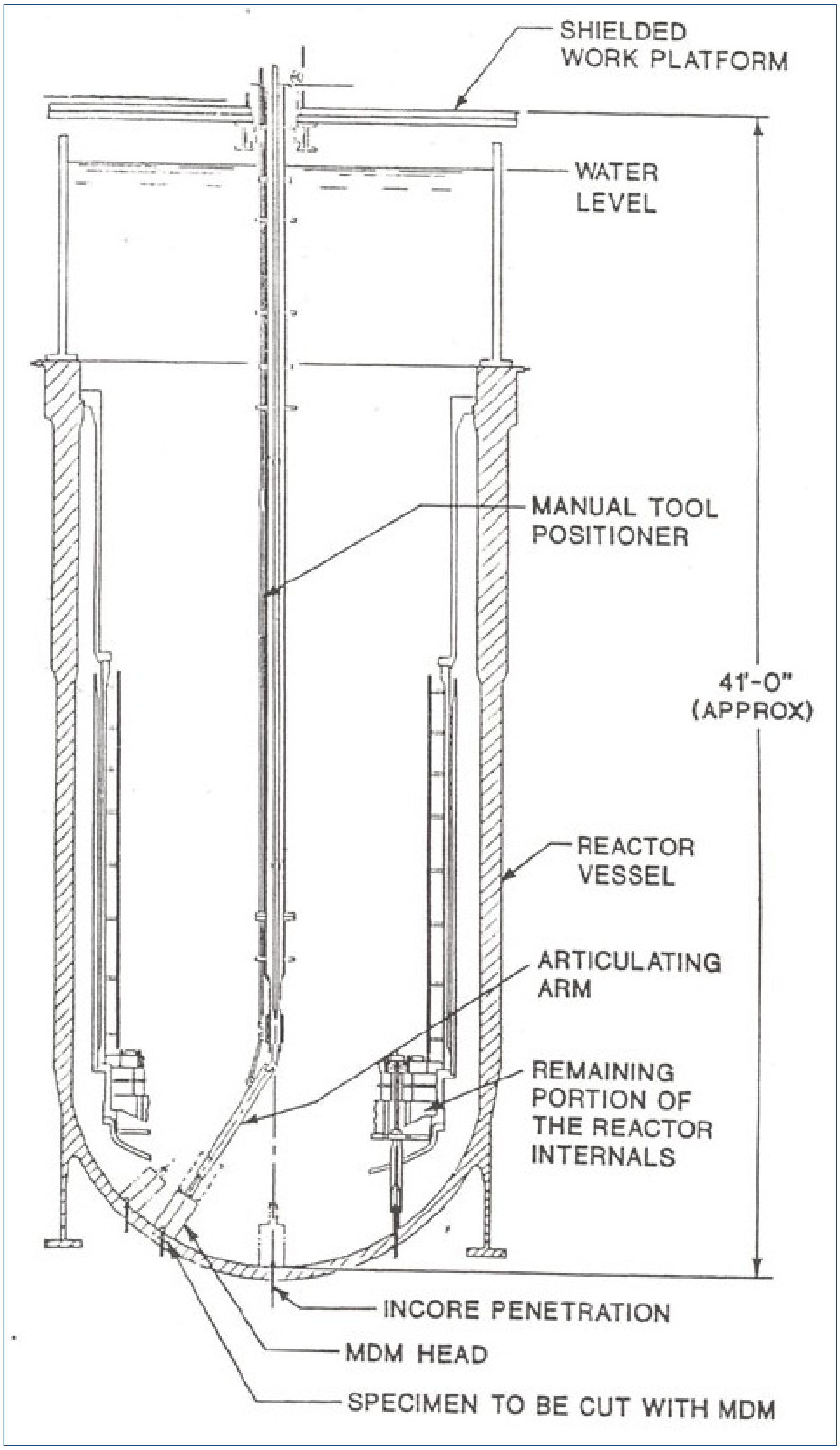
Source: GEND-057, TMI-2 Fission Product Inventory Program FY-85 Status Report (1986-11)

A.4.4 Core Stratification Sample Acquisition (Core Bore Samples)



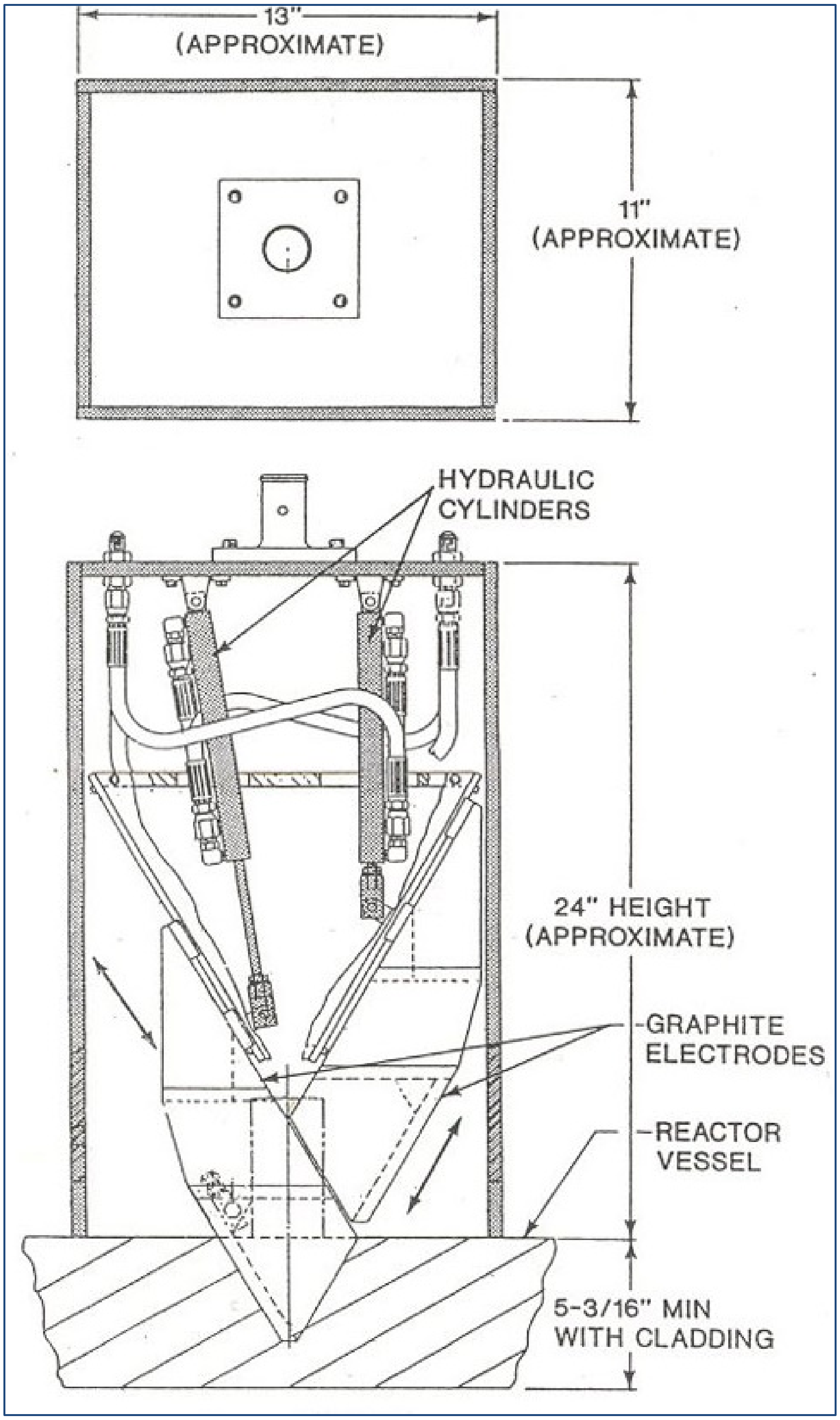
Source: EGG-TMI-7385-R1, TMI-2 Core Bore Acquisition Summary Report (1987-02)

A.4.5 Reactor Vessel Lower Head Wall Samples (MDM Cutting System)



MPR Associates, Inc., "Phase 4 Status Report, Removal of Test Specimens from the TMI-2 Reactor Vessel Bottom Head Project Summary," MPR-1195, July 27, 1990

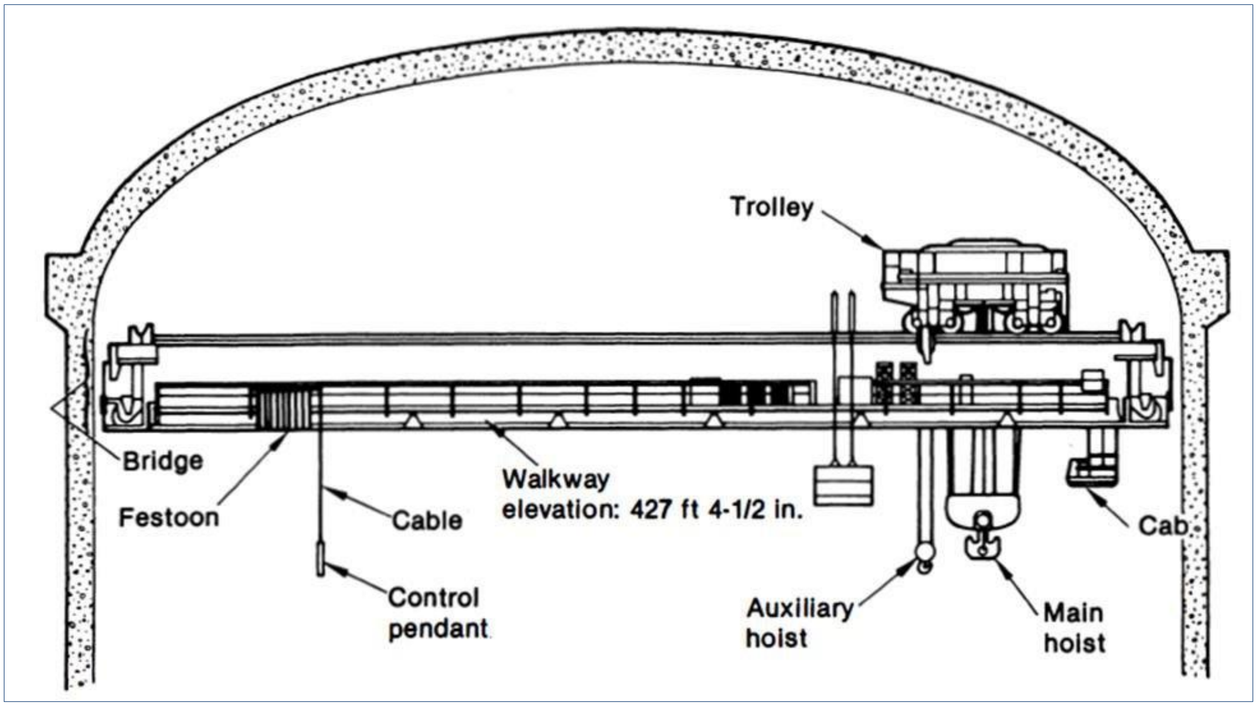
A.4.6 Reactor Vessel Lower Head Wall Samples (MDM Cutting Head)



Source: MPR Associates, Inc., "Phase 4 Status Report, Removal of Test Specimens from the TMI-2 Reactor Vessel Bottom Head Project Summary," MPR-1195, July 27, 1990

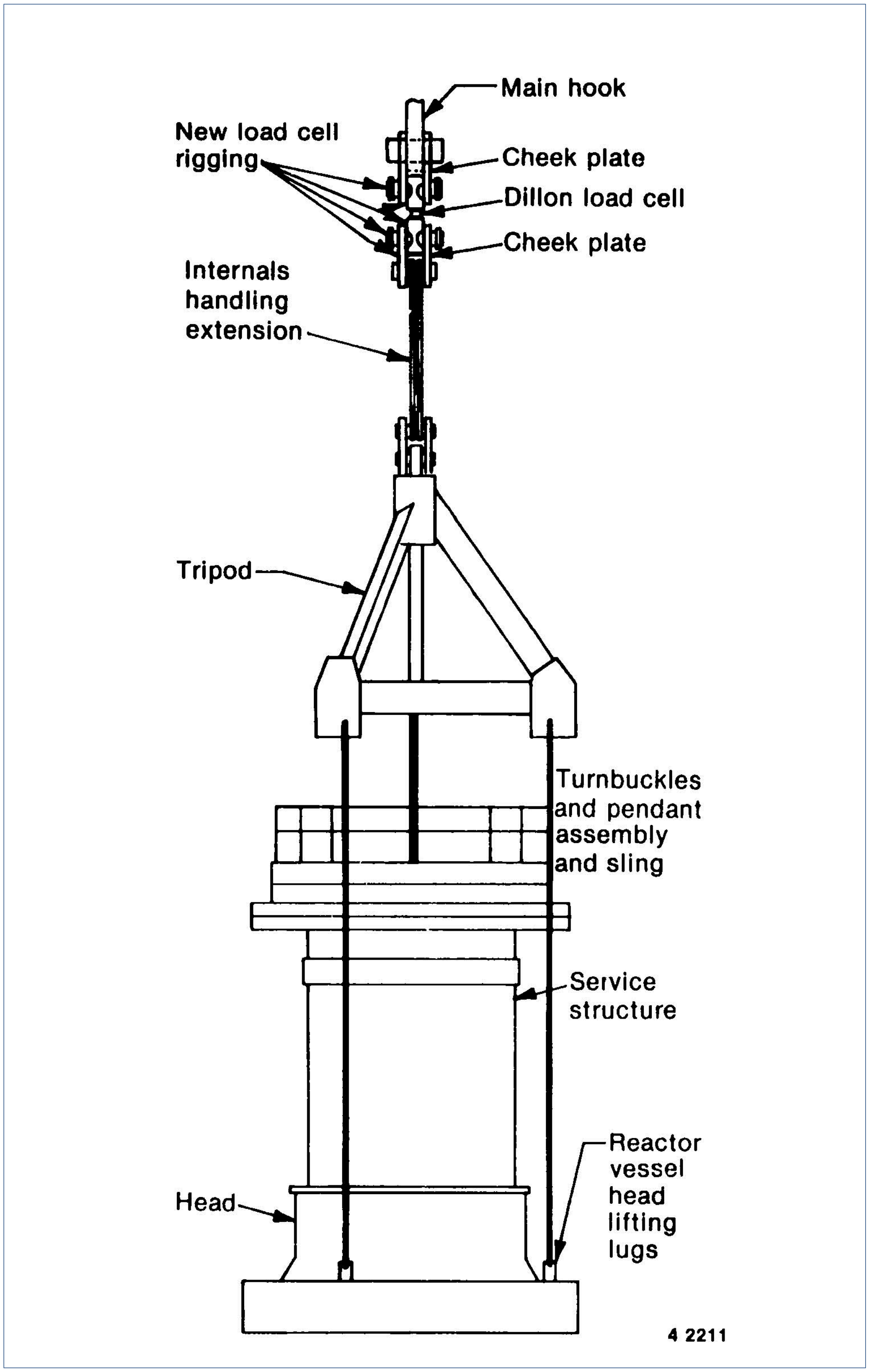
A.5 Pre-Defueling Preparations

A.5.1 Containment Building Polar Crane



Source: GEND-039, TMI-2 Technical Information and Examination Program Annual Report 1983 (1984-04)

A.5.2 Reactor Vessel Head Removal (Rigging Arrangement)



Source: GEND-044, TMI-2 Reactor Vessel Head Removal (1985-09)

A.5.3 Reactor Vessel Head Removal (Containment Building Arrangement)

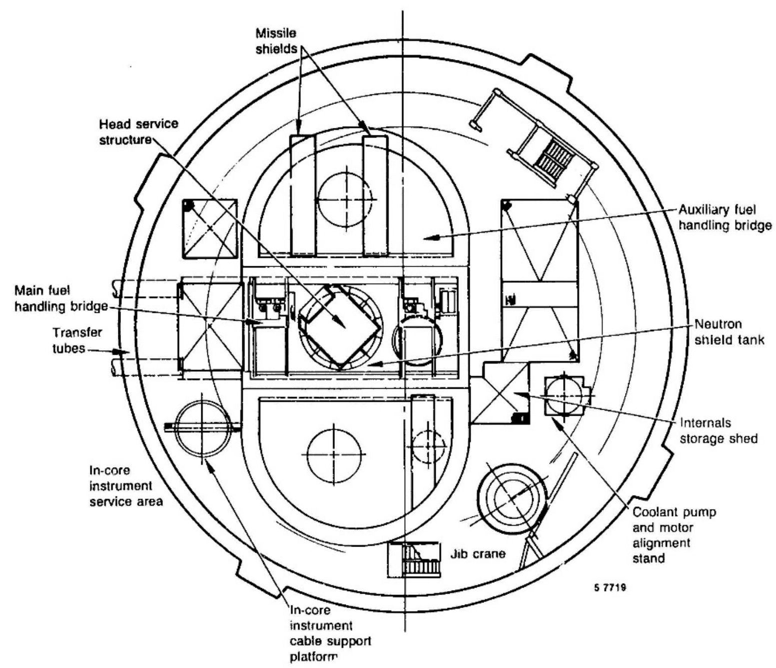


Figure A-1. Reactor building general arrangement before head removal.

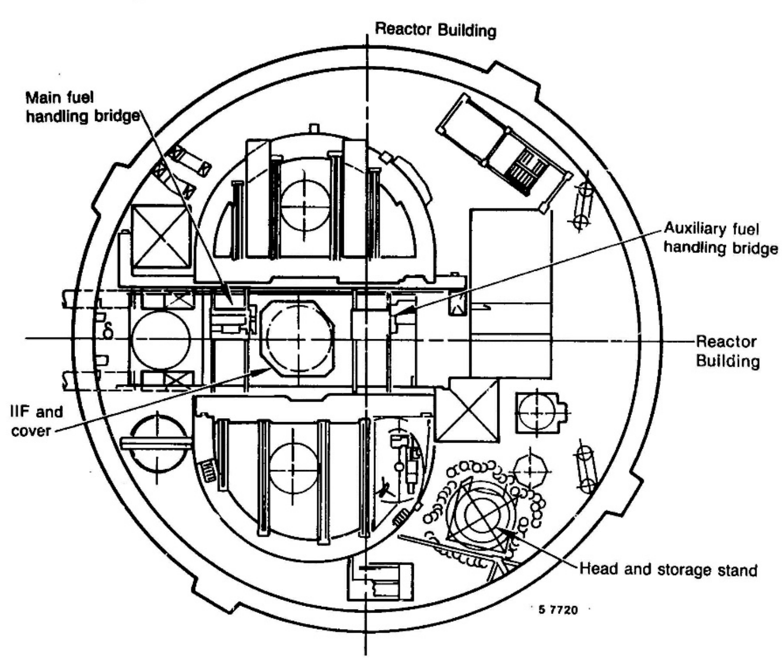
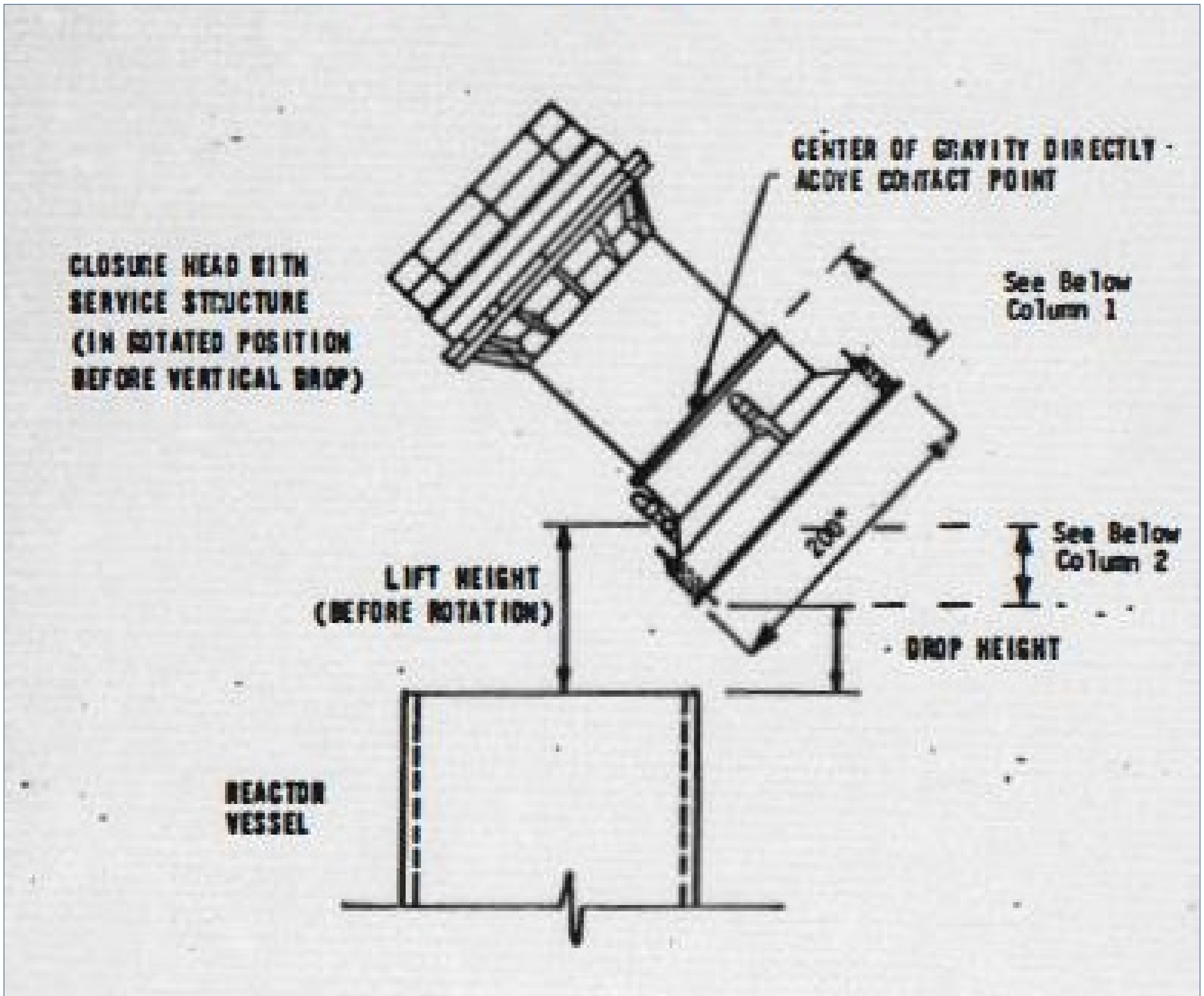


Figure A-2. Reactor building general arrangement after head removal.

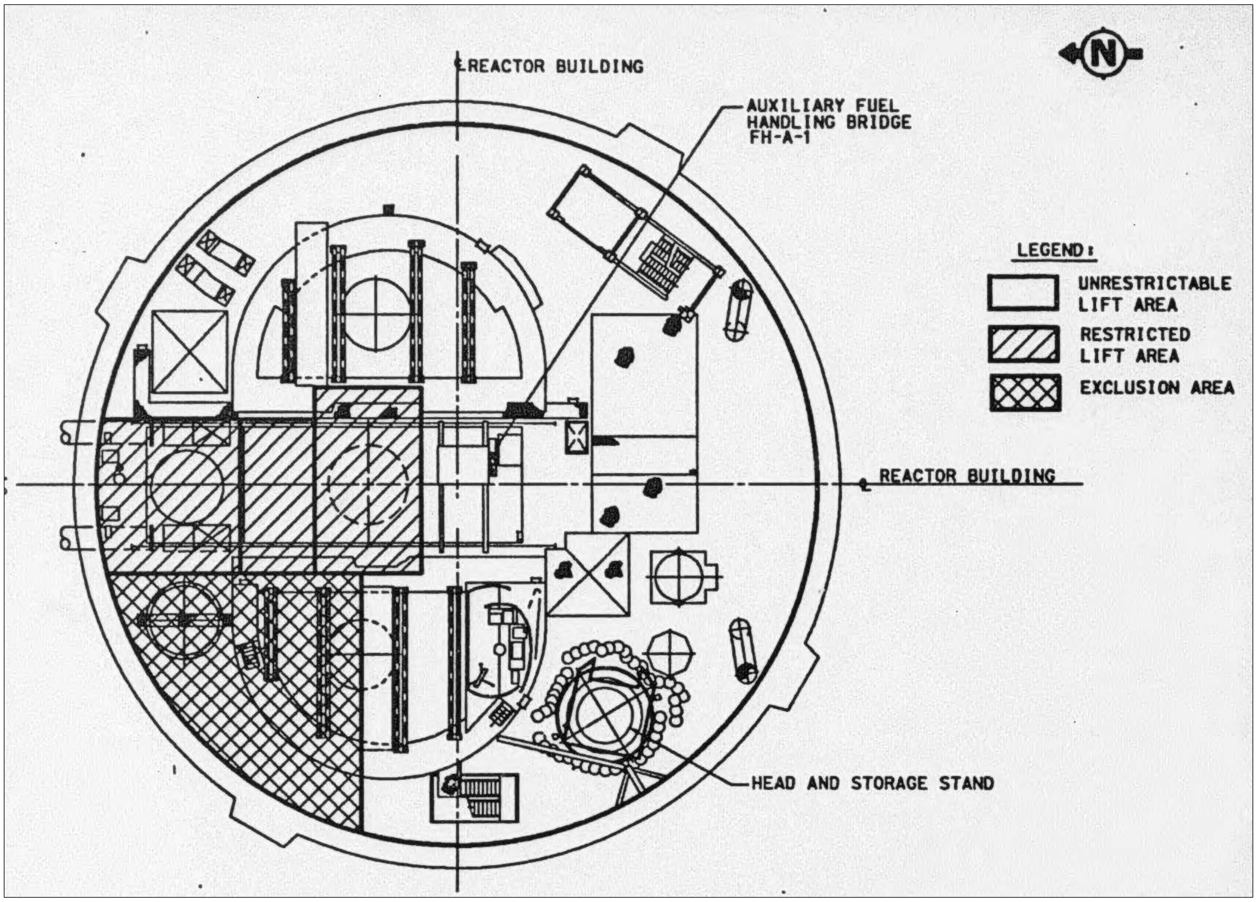
Source: GEND-044, TMI-2 Reactor Vessel Head Removal (1985-09)

A.5.4 Reactor Vessel Head Removal (Worst-Case Load Drop Configuration)



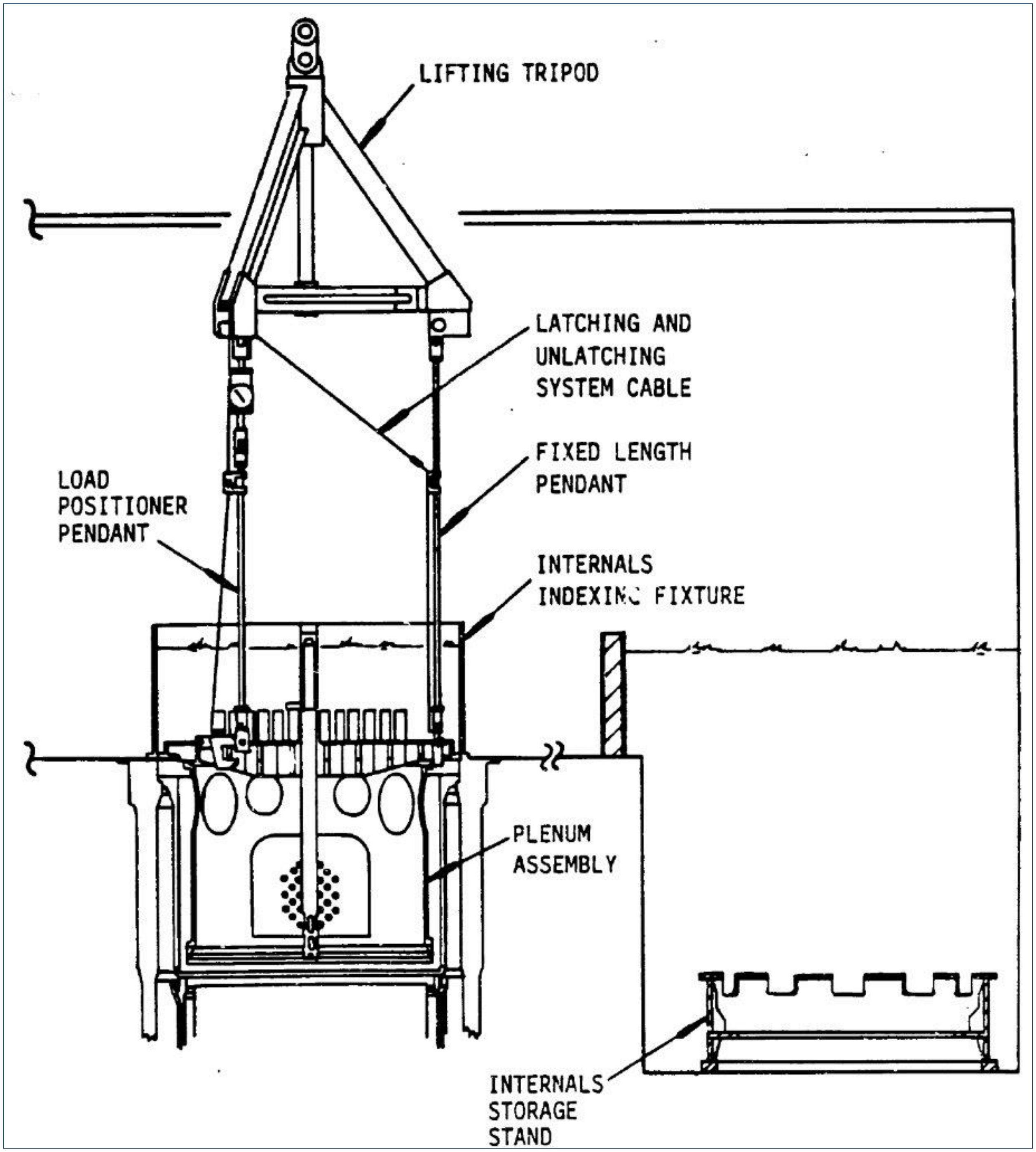
Source: (1984-03-09) GPU Safety Evaluation, Head Removal, Rev. 5

A.5.5 Load-Impact Areas in Containment Building



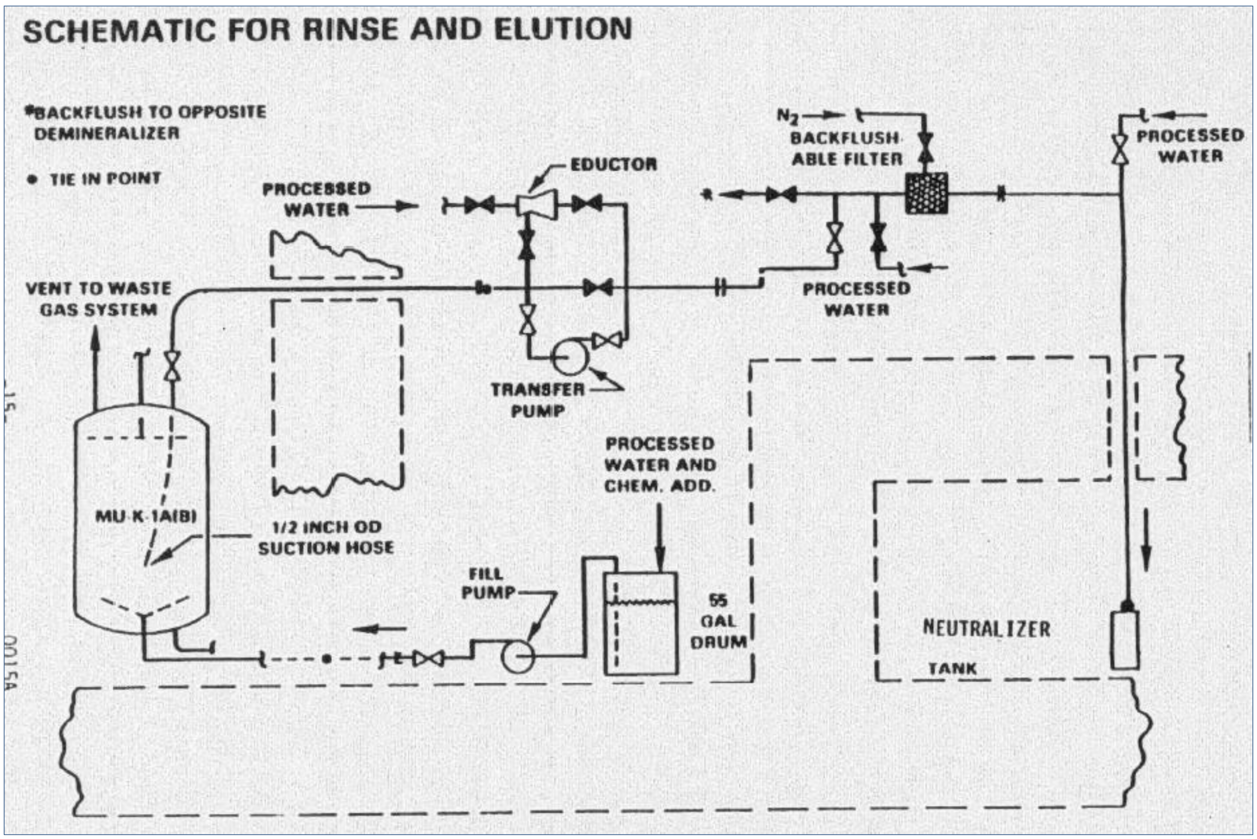
Source: (1986-06-02) GPU Safety Evaluation, Heavy Load Handling Inside Containment, Rev. 3

A.5.6 Reactor Vessel Upper Plenum Lift Rigging



Source: GEND-054, TMI-2 Reactor Vessel Plenum Final Lift (1986-01)

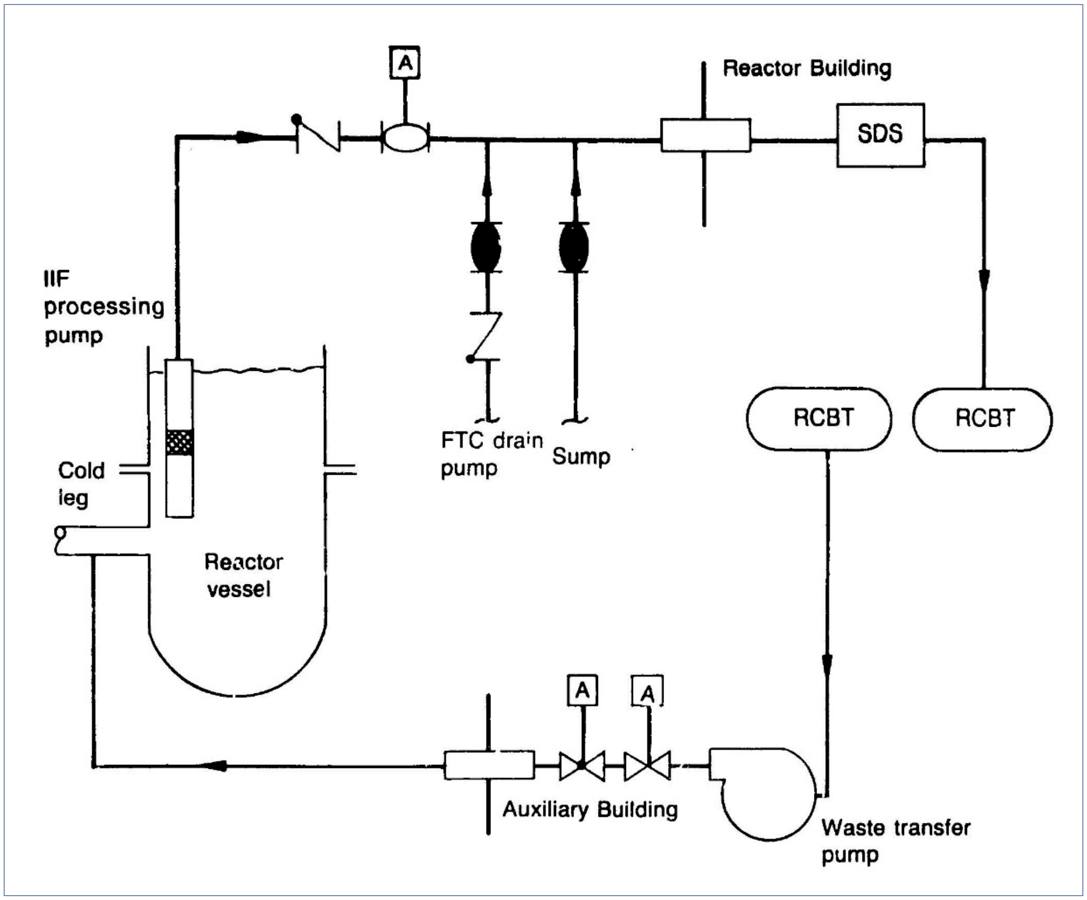
A.5.7 Makeup and Purification Demineralizer Cesium Elution Arrangement



Source: (1984-07-19) GPU Safety Evaluation, Makeup and Purification Cesium Elution SER

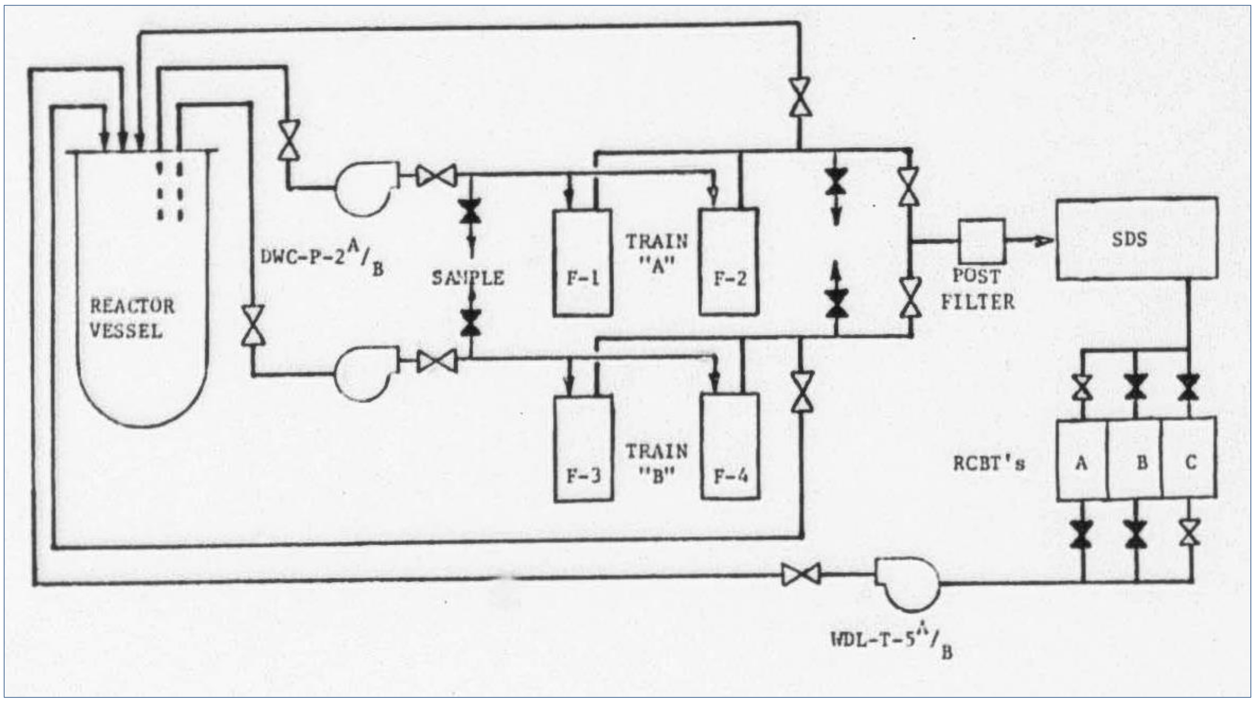
A.6 Defueling Tools and Systems

A.6.1 Internals Indexing Fixture Water Processing System



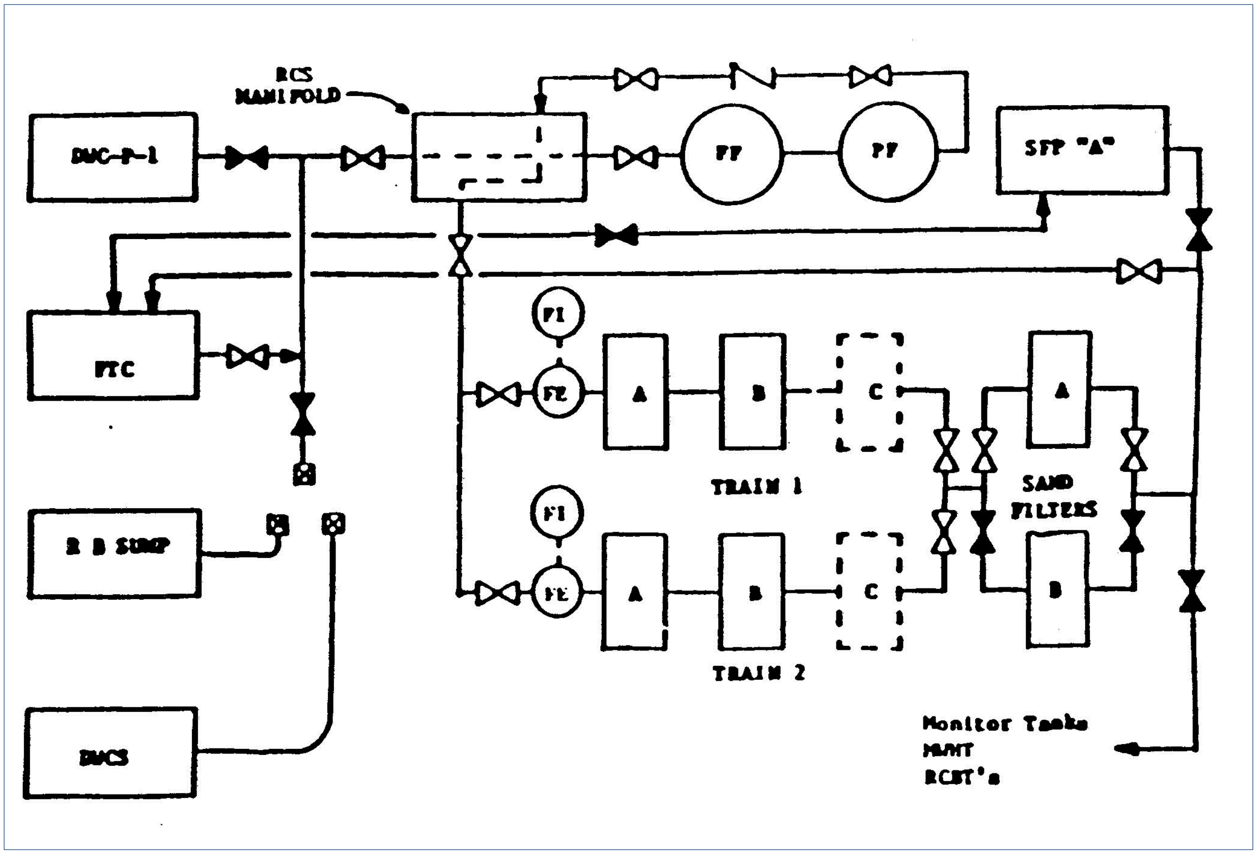
Source: GEND-044, TMI-2 Reactor Vessel Head Removal (1985-09)

A.6.2 Defueling Water Cleanup-Reactor Vessel Filtration System (Early Configuration)



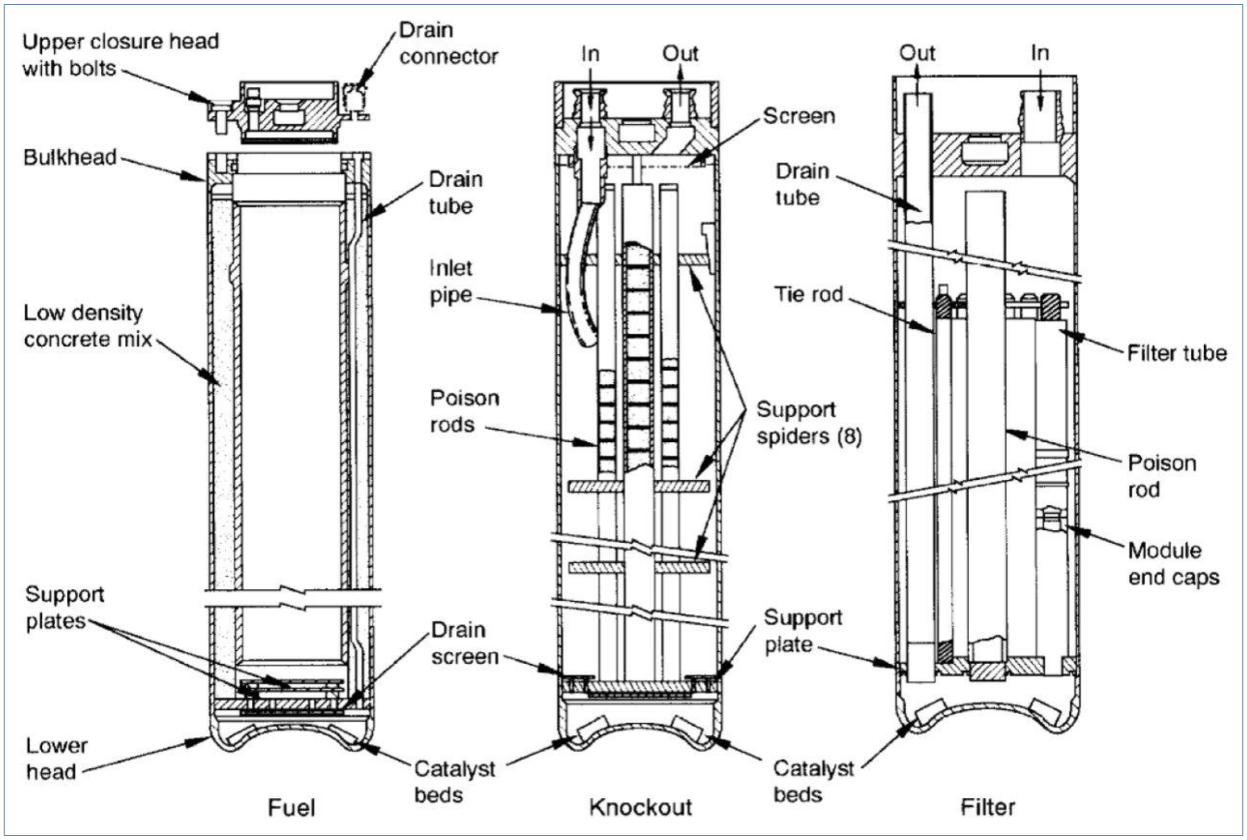
Source: (1985-08-16) GPU Technical Evaluation, SDS, Rev. 3 [Final configuration is described in source: (1987-04-29) GPU Technical Evaluation, Defueling Water Cleanup System, Rev 10.]

A.6.3 Defueling Water Cleanup-Fuel Transfer Canal Processing



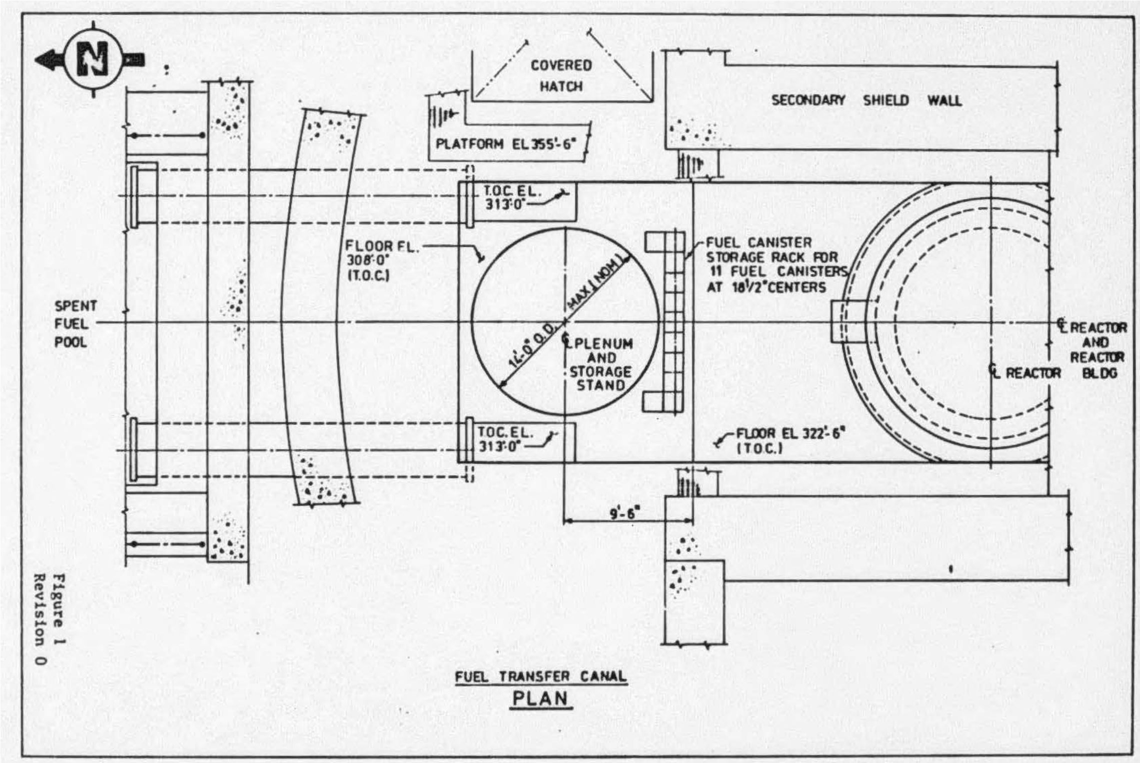
Source: (1987-09-28) GPU Technical Evaluation, SDS, Rev. 5

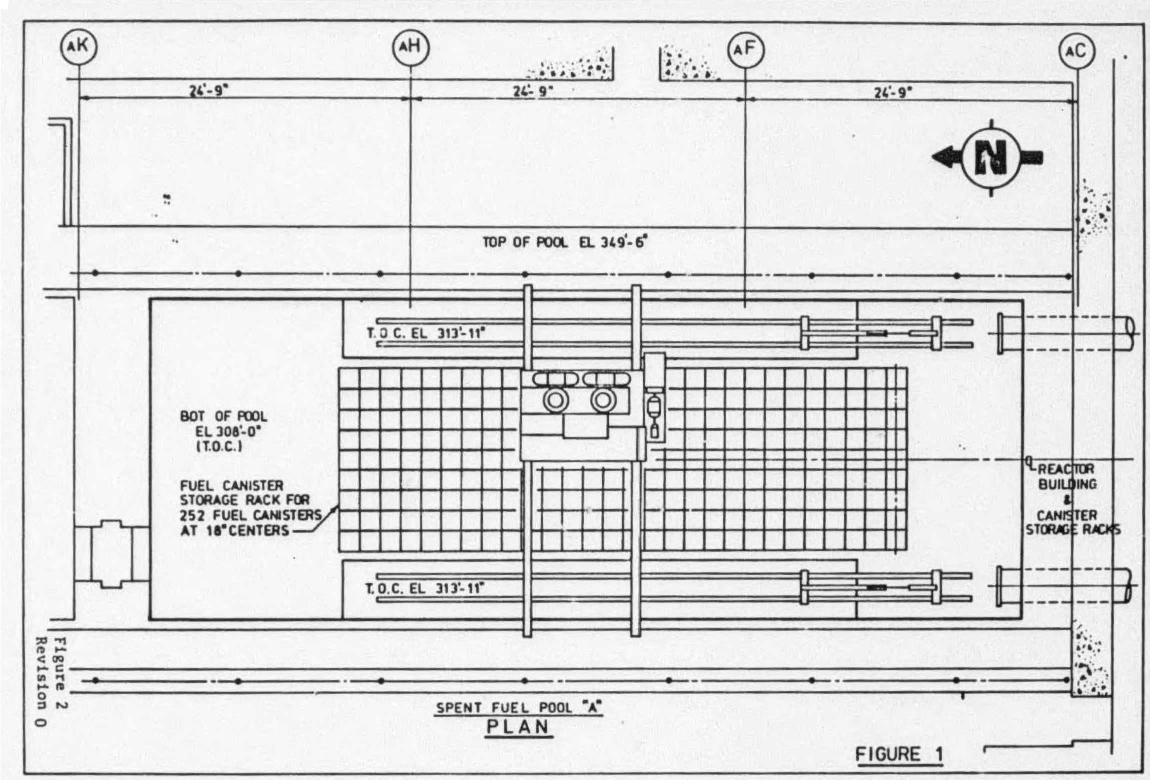
A.6.4 Defueling Canisters



Source: INL-CON-12-26246 (CONF), Lessons from TMI Packaging, Transport and Disposition that Apply to Fukushima (2012)

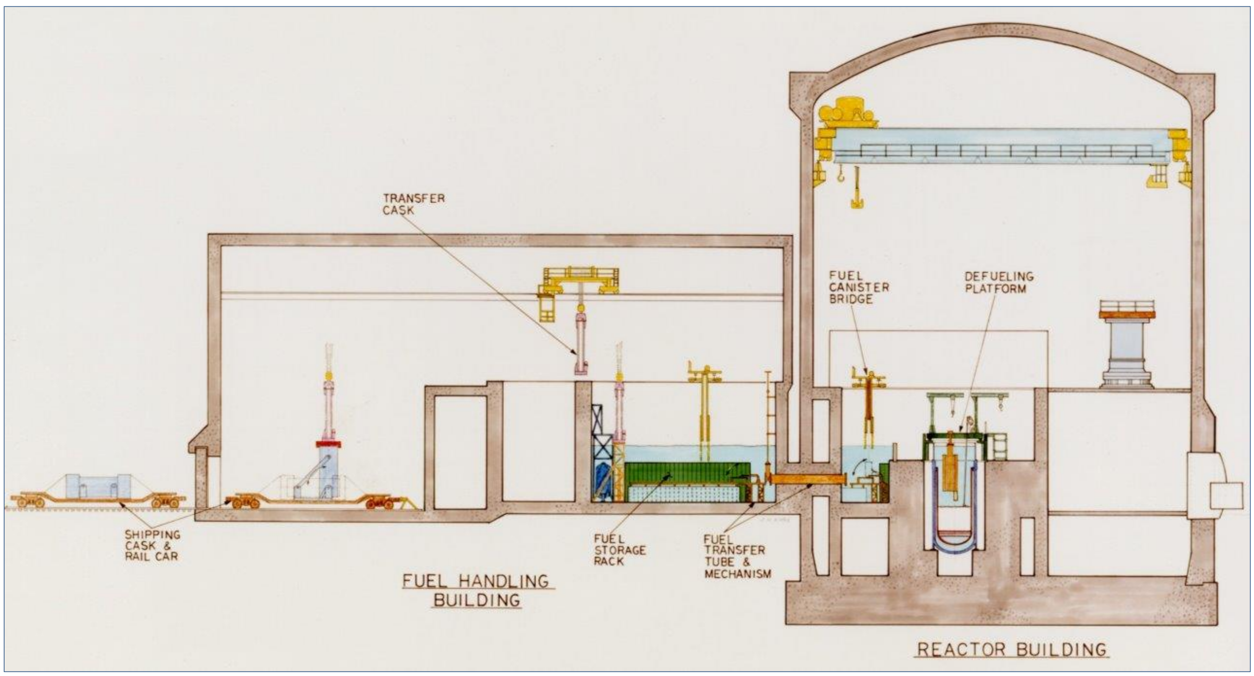
A.6.5 Fuel Canister Storage Racks (Containment and Spent Fuel Pool-A)





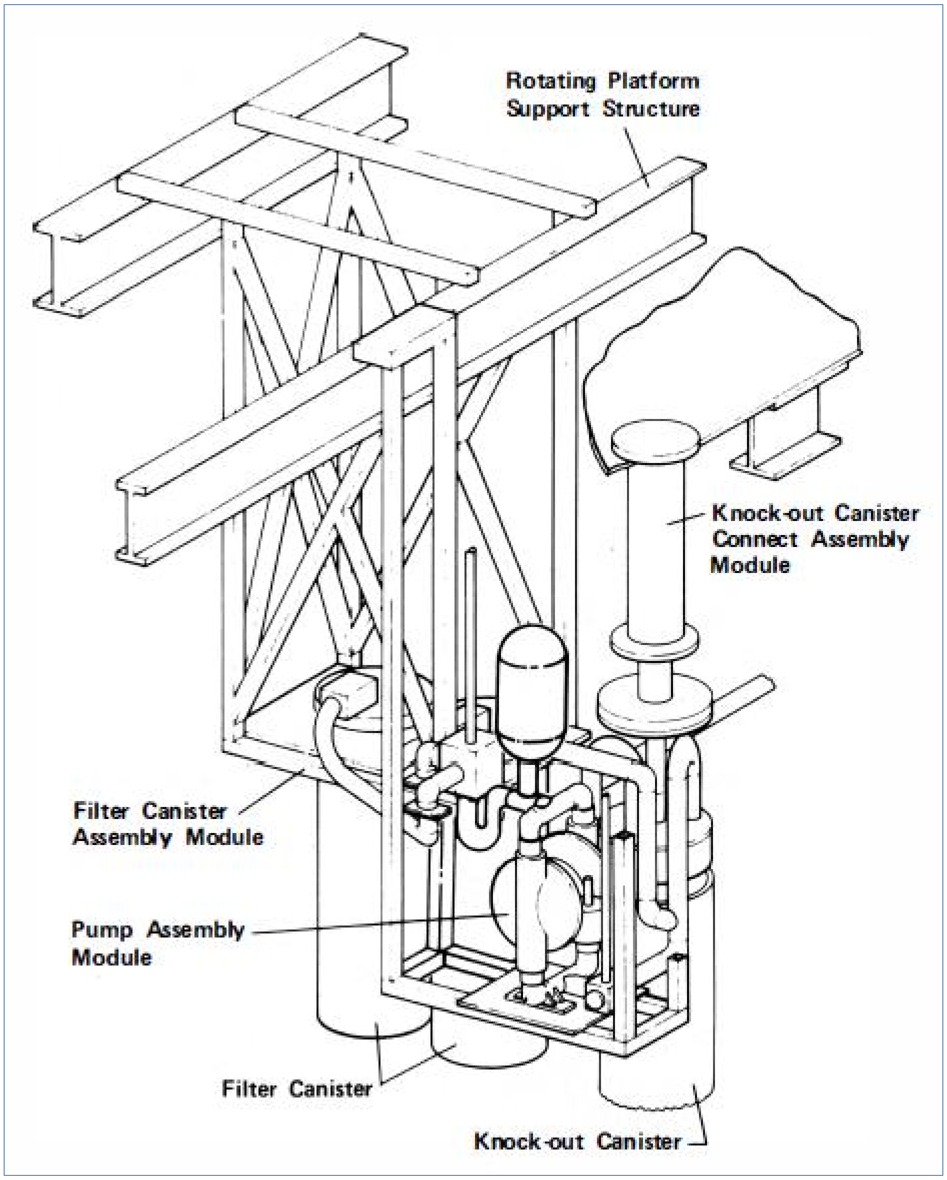
Source: GPU Technical Evaluation, Fuel Canister Storage Racks TER, Rev. 0

A.6.6 Canister Handling and Preparation for Shipment Arrangement



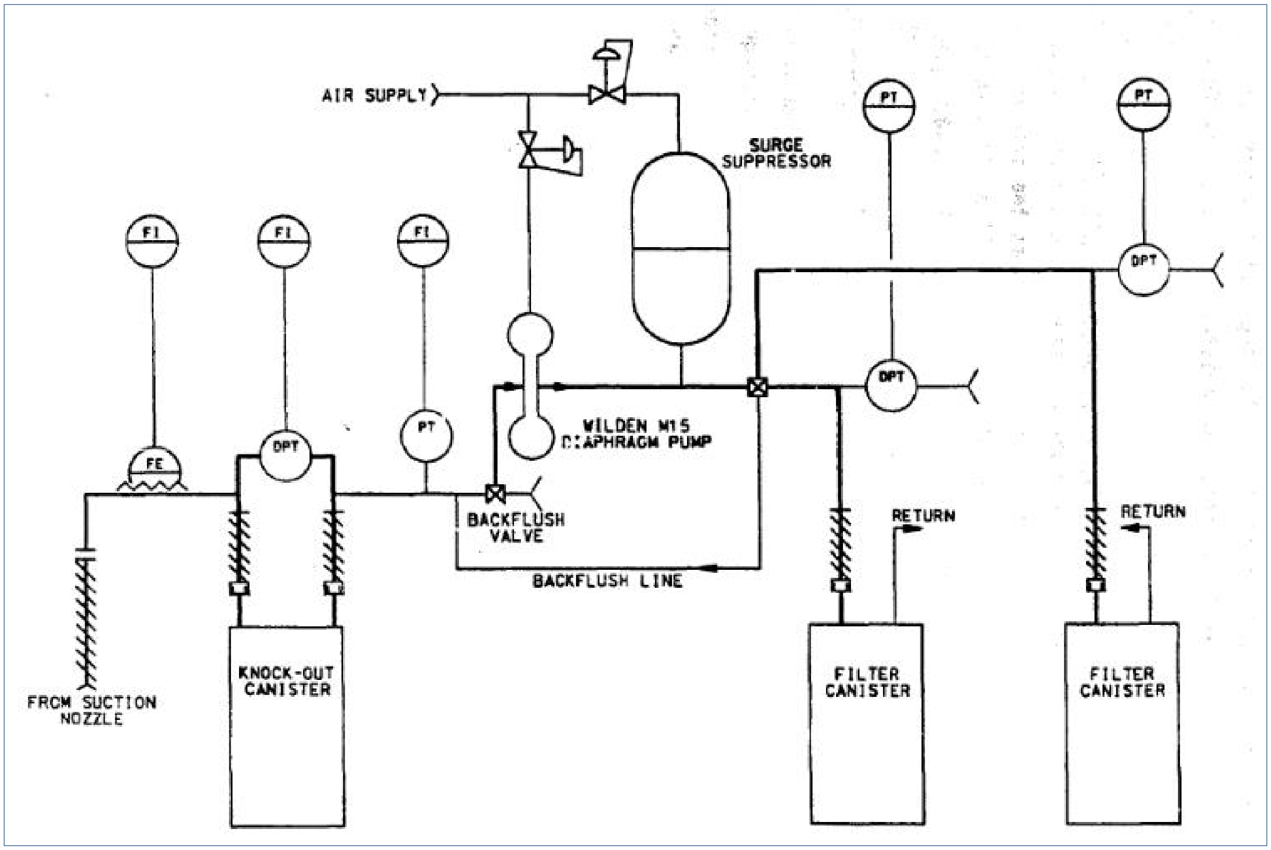
Source: Legacy vugraph based on DOE-ID-10276, TMI-2 Lessons Learned by the DOE 1979-1990 (1990-03)

A.6.7 Fines/Debris Vacuum System (Isometric)



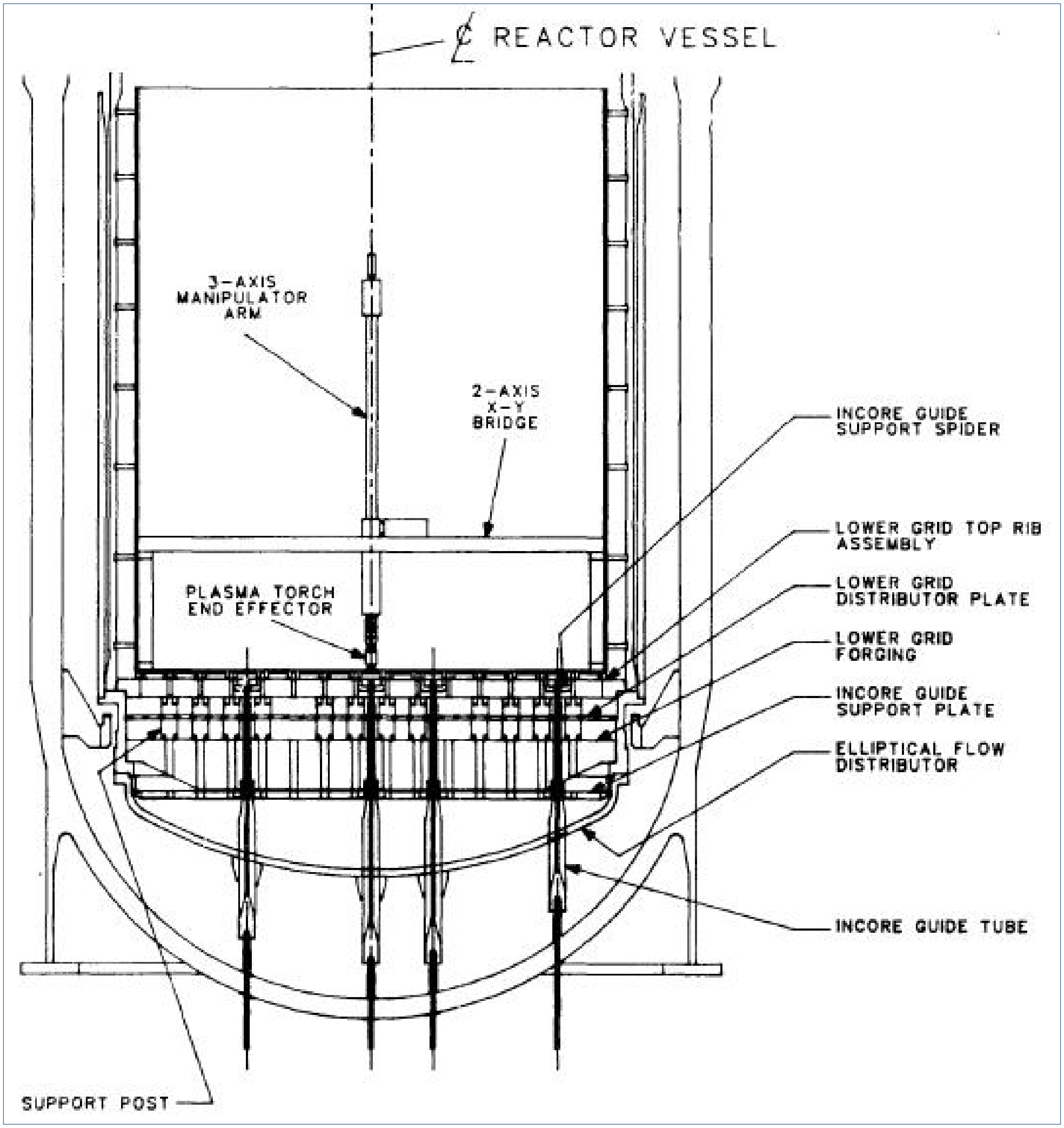
Source: GEND-INF-093, Lower Core Support Assembly Defueling Planning and Tools (1988-10)

A.6.8 Fines/Debris Vacuum System (Piping Diagram)



Source: GEND-INF-093, Lower Core Support Assembly Defueling Planning and Tools (1988-10)

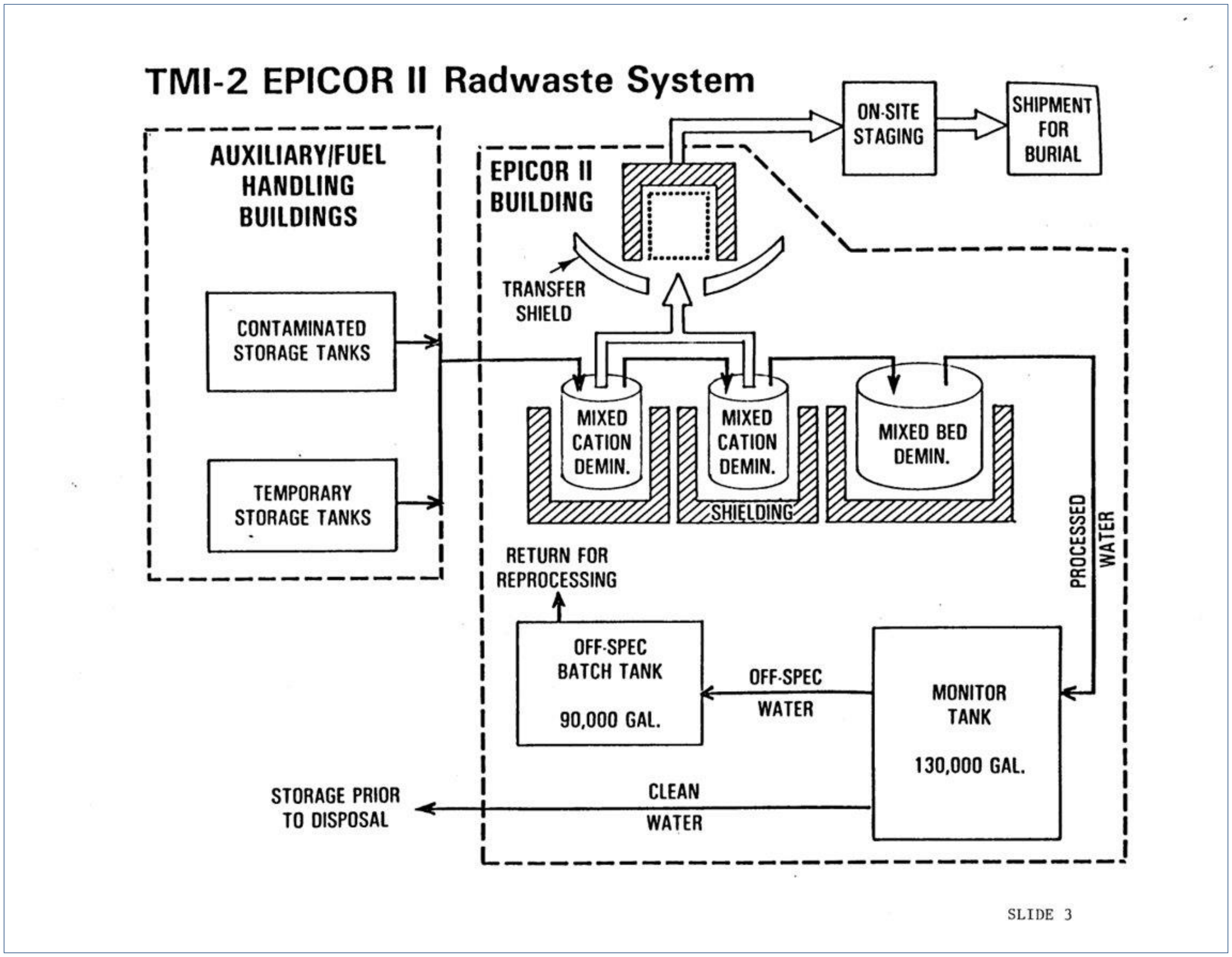
A.6.9 Plasma Arc Torch (Arrangement for Cutting the Lower Core Support Assembly)



Source: GEND-INF-093, Lower Core Support Assembly Defueling Planning and Tools (1988-10)

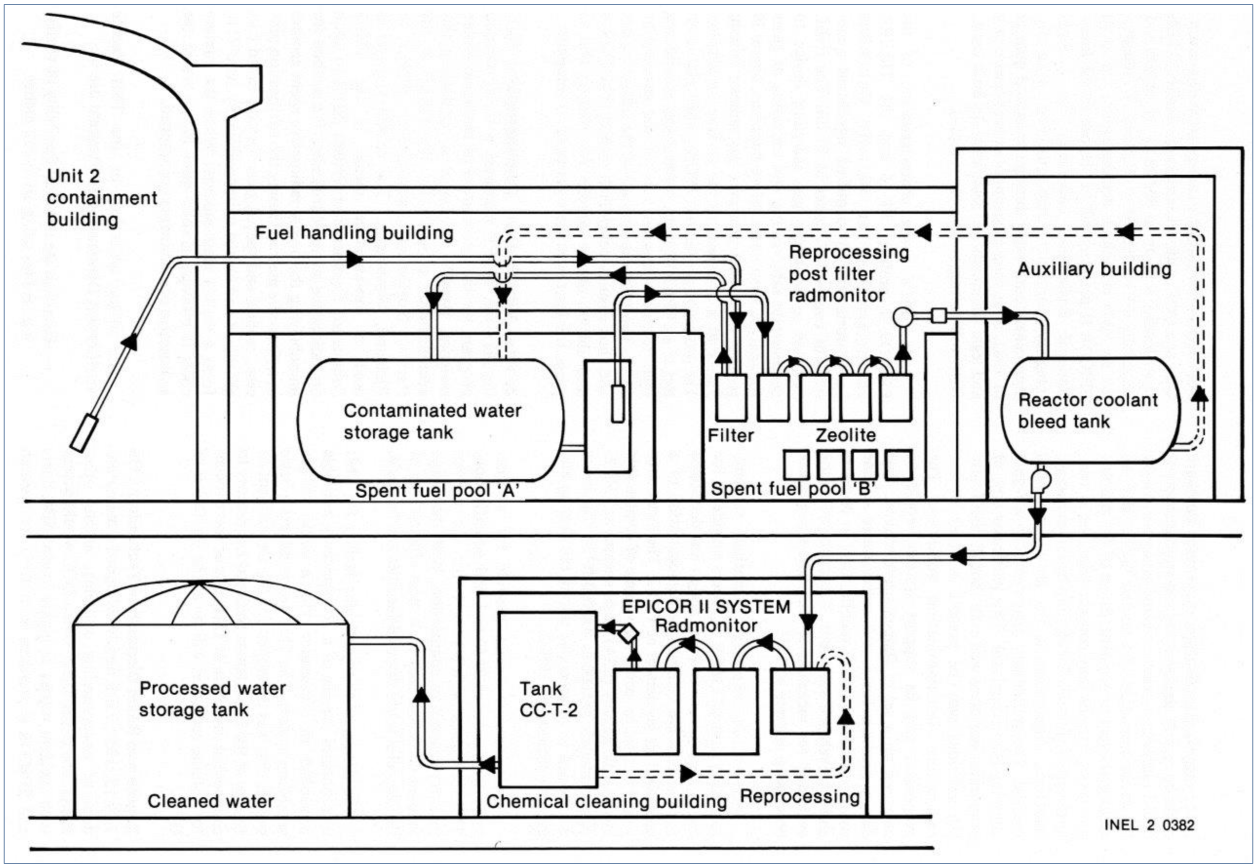
A.7 Waste Water Cleanup Systems

A.7.1 EPICOR II for Early Auxiliary Building Liquid Cleanup



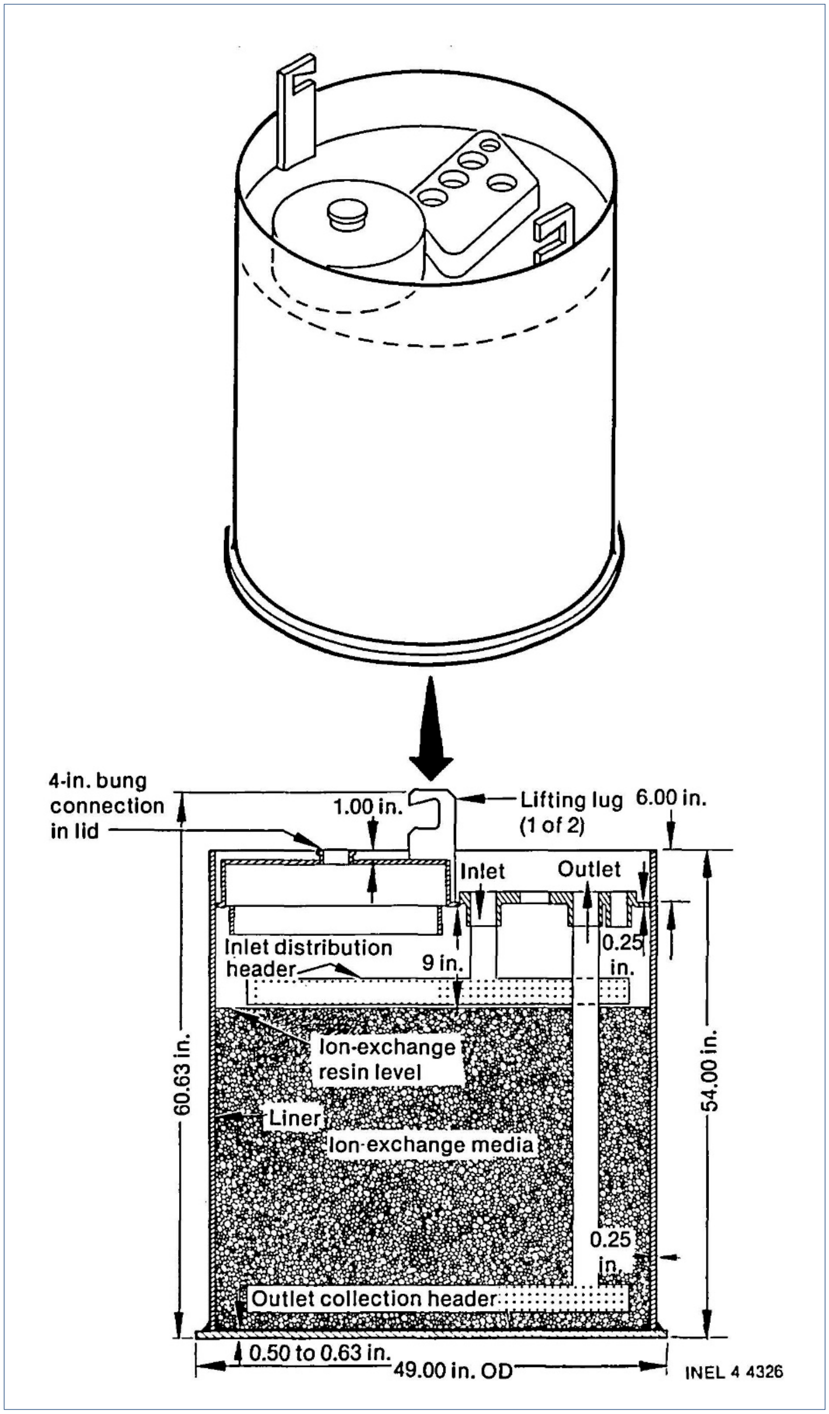
Source: Hovey, G., et al., "TMI-2 Status of Recovery and Future Plans," Presentation to the Joint ANS/ENS Meeting, Washington, DC, November 18, 1980. (Presentation slides handout.)

A.7.2 EPICOR II for Polishing the Submerged Deminerizer System Effluients



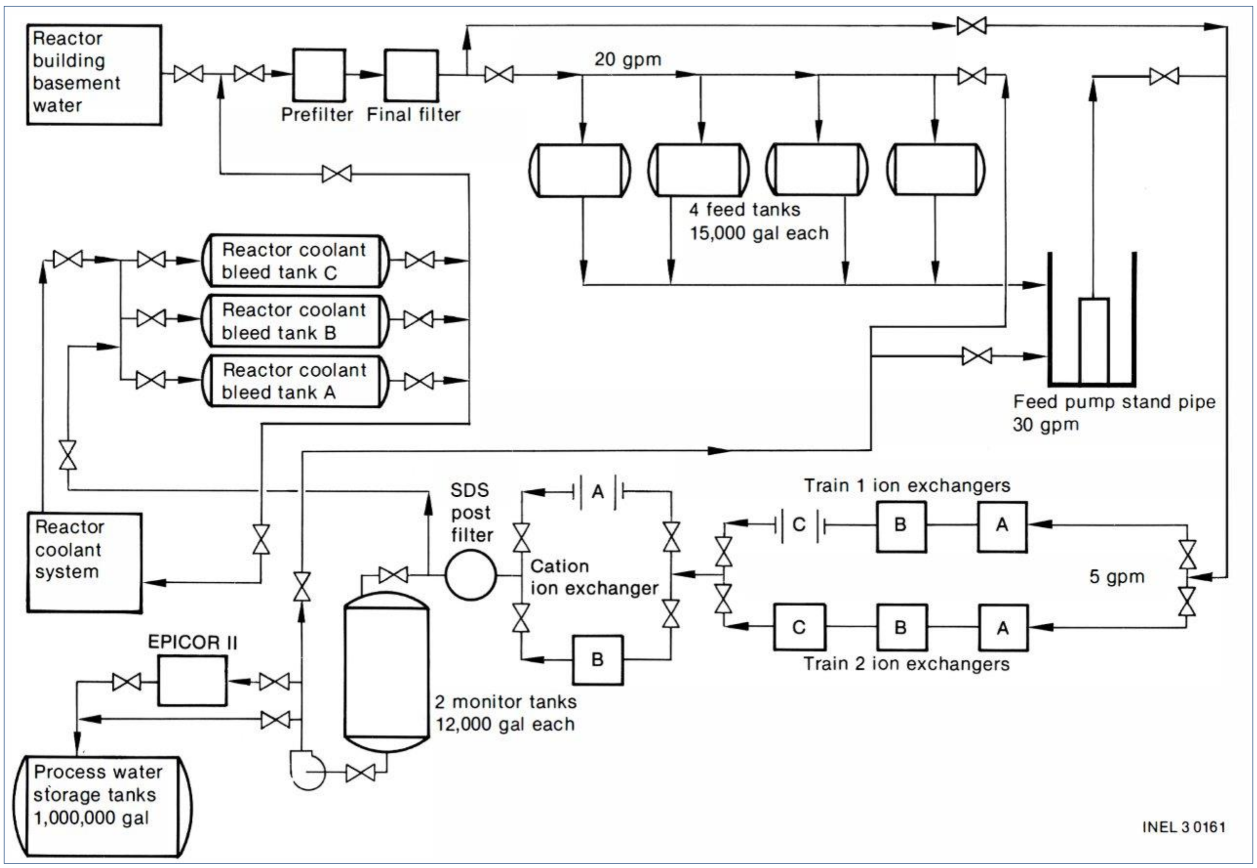
Source: GEND-022, TMI-2 Technical Information and Examination Program Annual Report 1981 (1982-04)

A.7.3 EPICOR II Typical Liner



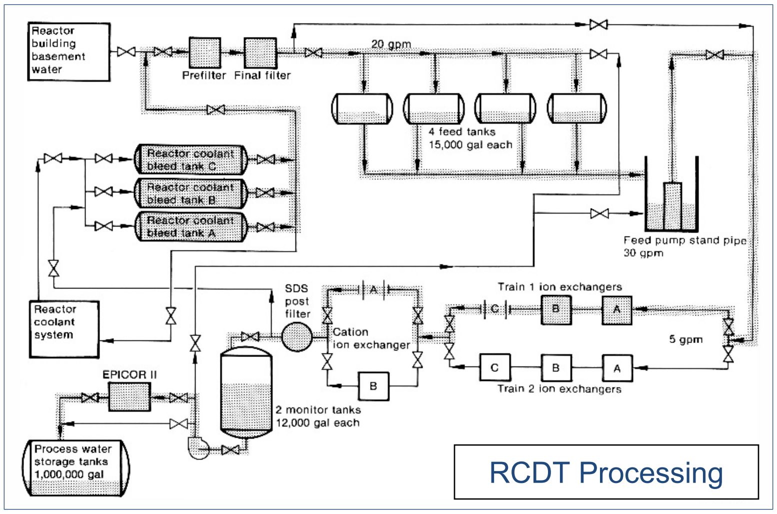
Source: NUREGCR-4150, TMI-2 EPICOR-II Resin Degradation Results from First Resin Samples of PF-8 and PF-20 (1985-07)

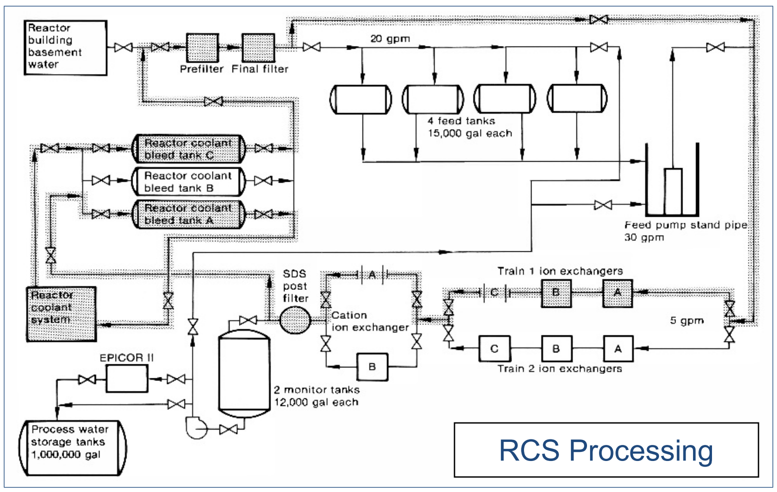
A.7.4 Submerged Deminerizer System (SDS) Flowsheet

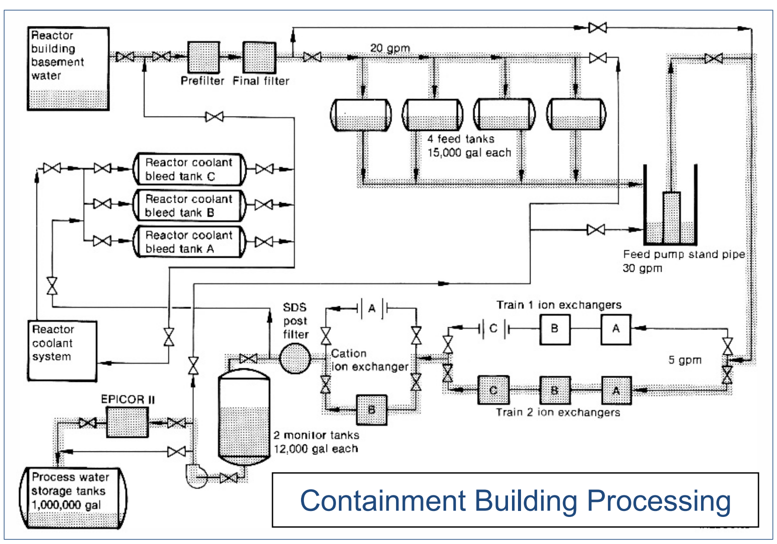


Source: GEND-031, Submerged Demineralizer System Processing of TMI-2 Waste Water (1983-02)

A.7.5 SDS Flowsheets for Cleanup Modes

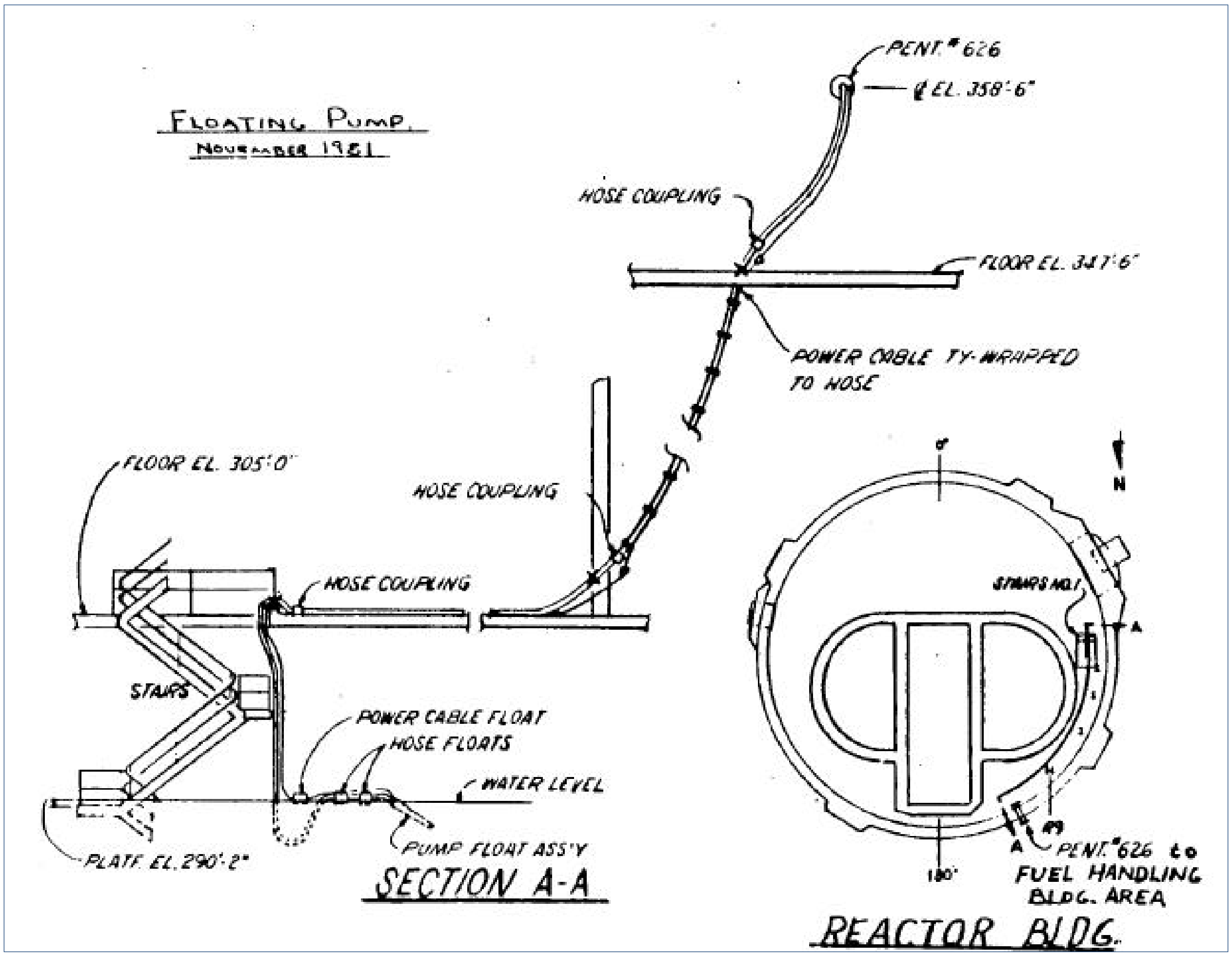






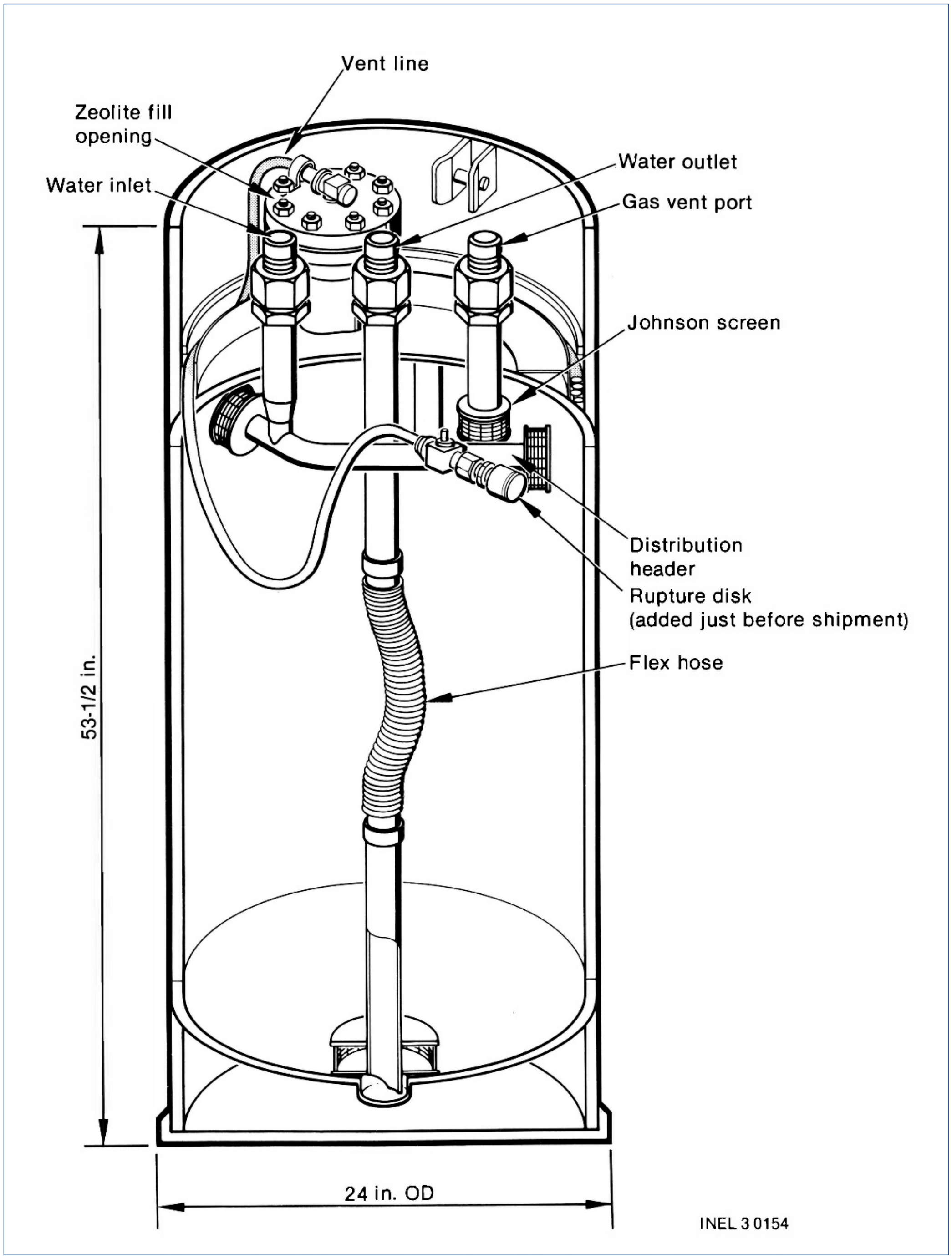
Source: GEND-031, Submerged Demineralizer System Processing of TMI-2 Waste Water (1983-02)

A.7.6 SDS Floating Suction Pump in Containment Building Basement



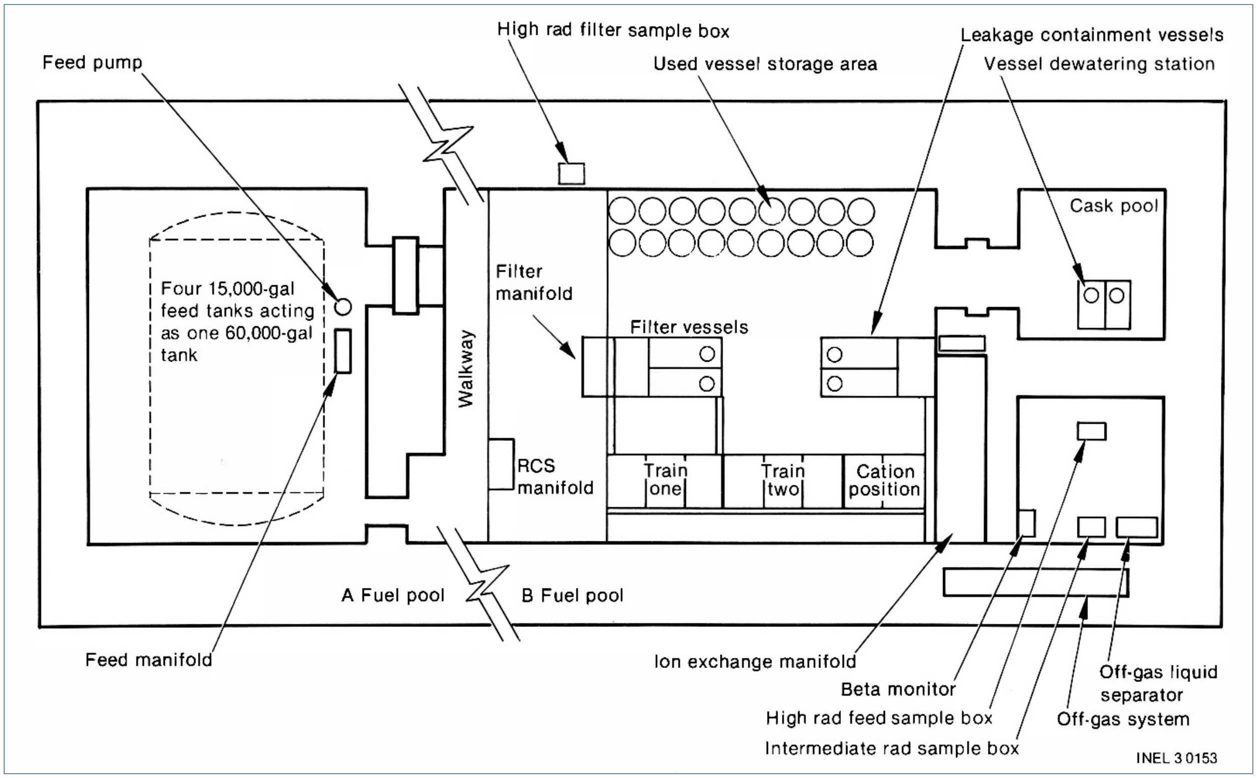
Source: (1982-11-14) TMI-2 Special Sessions (ANS Meeting)

A.7.7 SDS Ion Exchange Liner



Source: GEND-031, Submerged Demineralizer System Processing of TMI-2 Waste Water (1983-02)

A.7.8 SDS Arrangement Inside Spent Fuel Pool-B



Source: GEND-031, Submerged Demineralizer System Processing of TMI-2 Waste Water (1983-02)

NRC FORM 335 (12-2010) NRCMD 3.7

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

(See instructions on the reverse)

1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG/KM-0001
Supplement 3

2. TITLE AND SUBTITLE

Three Mile Island Accident of 1979 Knowledge Management Digest Cleanup Safety Evaluations

3. DATE REPORT PUBLISHED

MONTH	YEAR
June	2022

4. FIN OR GRANT NUMBER

5. AUTHOR(S)

Don Marksberry
Michel Call
Alice Chung

6. TYPE OF REPORT

Technical

7. PERIOD COVERED (Inclusive Dates)

8. PERFORMING ORGANIZATION - NAME AND ADDRESS (If NRC, provide Division, Office or Region, U. S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)

Division of Systems Analysis
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission Washington, DC
20555-0001

9. SPONSORING ORGANIZATION - NAME AND ADDRESS (If NRC, type "Same as above", if contractor, provide NRC Division, Office or Region, U. S. Nuclear Regulatory Commission, and mailing address.)

Same as 8, above

10. SUPPLEMENTARY NOTES

11. ABSTRACT (200 words or less)

After the accident at Three Mile Island Generating Station, Unit 2 (TMI-2) in 1979, safety evaluations were required by the licensee and NRC for most cleanup activities and new cleanup systems. This supplement (Supplement 3) details the safety evaluations of 64 cleanup activities that were performed during the 1979-1993 period and has been organized into the following groups: data collection; pre-defueling preparations; defueling tools and systems; defueling operations; and liquid waste management systems.

The safety evaluations for each of the cleanup activities are discussed by safety topics which include: criticality; boron dilution; decay heat removal; fire protection; hydrogen generation; industrial safety; instrument interference; impact on Unit I activities; load drop; occupational exposure, including internal and external exposures; pyrophoricity; radiation protection/as low as reasonably achievable practice; radiological release; reactor vessel integrity; seismic hazard; radiation shielding; and vital equipment protection.

Supplement 3 is part of the NUREG/KM-001 series and provides complimentary details of the TMI-2 cleanup activities which can be found in its preceding supplements. Supplement I provided summary descriptions of programs, activities, systems and tools that were long involved in the decade-long cleanup campaign of the damaged reactor core and severely contaminated equipment and buildings. The DVDs accompanying Supplement I contain most of the references cited in Supplement 3. Whereas Supplement 2 consolidated many of the experiences and lessons during the TMI-2 cleanup that had been recorded in numerous reports and papers.

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)

Accident Cleanup
Accident Recovery
Safety Evaluation
Knowledge Management
Lessons Learned
Reactor Core Damage
Severe Accident
Three Mile Island Unit 2
TMI-2

13. AVAILABILITY STATEMENT

unlimited

14. SECURITY CLASSIFICATION

(This Page) unclassified

(This Report) unclassified

15. NUMBER OF PAGES

16. PRICE

CONVERSIONS

Radiation Dose

1 mrem (1 millirem, 10^{-3}) = *10 microsieverts (10 μSv, 10^{-5})

100 mrem = *1 millisievert (1 MSv)

1 rem = *10 mSv

100 rem = *1 Sv

Radioactive Concentration

27 picocuries (27 pCi, 2.7×10^{-11}) = *1 becquerel (1 Bq)

1 millicurie (1 mCi, 0.001) = *37 megabecquerels (37 MBq, 3.7×10^{7})

1 curie (1 Ci) = *37 gigabecquerels (37 GBq, 3.7×10^{10})

Radiation Absorbed Energy

1 roentgen = *0.877 rad = *0.00877 gray (Gy)

100 rad = *1 Gy

Length

1 inch (in.) = *2.54 centimeters (cm)

1 foot (ft) = 0.3048 meter (m)

Volume and Weight

1 gallon (gal) = 3.7854 liters (L)

1 pound (lb) = 0.4536 kilograms (kg)

1 ton (U.S.) = *2000 lb = 907.1847 kg

Pressure

1 pound per square inch (psi) = 6.8948 kilopascals (kPa)

1 atmosphere (atm) = *101.325 kPa

Temperature

Degrees Celsius (°C) = 5/9 × (°F - 32)

Degrees Fahrenheit (°F) = (9/5 × °C) + 32

* Exact conversion factor

UNITED STATES NUCLEAR REGULATORY COMMISSION

NUREG/KM-0001, Supplement 3
June 2022

@NRCgov