UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

February 26, 1996

Mr. Donald E. Erb, Acting Project Manager
Office of Nuclear Energy
U.S. Department of Energy
Washington, D.C. 20585

SUBJECT: DRAFT COPY OF PREAPPLICATION SAFETY EVALUATION REPORT (PSER) ON THE MODULAR HIGH-TEMPERATURE GAS-COOLED REACTOR (MHTGR)

Dear Mr. Erb:

Enclosed is a draft of the final PSER which documents the staff's preapplication review of the MHTGR design. In accordance with your letter of July 17, 1995, the PSER does not contain Applied Technology information and, thus, does not carry that designation.

The enclosed PSER contains minor revisions to the draft PSER that was submitted to you in my letter of June 30, 1995, for a review for Applied Technology information. The revisions are based on an internal review after the June 30, 1995, letter. The enclosed PSER was submitted to the Commission in SECY-95-299, "Issuance of the Draft of the Final Preapplication Safety Evaluation Report (PSER) for the Modular High Temperature Gas-Cooled Reactor (MHTGR)," on December 19, 1995.

The enclosed PSER is (1) Volume 1, which contains the documentation of the staff's preapplication review of the MHTGR design and the conclusions of the staff on the design from this review, and (2) Volume 2, which contains the appendices, without copies of the documents that are referenced in the PSER and available through the NRC Public Document Room. These documents, which would be in Appendices C through J of Volume 2, are not essential for the staff's discussion of MHTGR licensability and policy issues and are, therefore, not included in the enclosed PSER to reduce its size. These documents were provided to you in our letter of June 30, 1995, and will be provided in the final PSER.

Please provide comments to the Nuclear Regulatory Commission (NRC) on the technical discussions and conclusions in the enclosed PSER within 6 weeks of the receipt of this letter. These comments will be considered for inclusion into the PSER before it is submitted to the Commission as the final PSER for the staff's preapplication review of the MHTGR. Section 4.2.9 of the PSER states that the Department of Energy (DOE) should provide in its design approval application for the MHTGR the basis for designating design information as being required to be withheld from the public. That PSER section states further that DOE should include in the application an explanation as to how information designated as Applied Technology falls within the scope of the Atomic Energy Act. In your comments, you are also requested to address the discussion on the Applied Technology designation in Sections 1.8 and 4.2.9 of the enclosed PSER.

In the comments you provide, you are also requested to address your plans and schedule for submitting an application for design certification for the MHTGR design.

This reporting requirement affects nine or fewer respondents and, therefore, is not subject to Office of Management and Budget review under P.L. 96-511. If you have any questions regarding this request, or want a copy of the documents not included in Appendices C through J of Volume 2, please contact me at (301) 415-1307.

Sincerely,

Jack Donohew

Jack Donohew, Project Manager
Standardization Project Directorate
Division of Reactor Program Management
Office of Nuclear Reactor Regulation

Project No. 672

Enclosure: Draft PSER

cc w/enclosure:
See next page

In the comments you provide, you are also requested to address your plans and schedule for submitting an application for design certification for the MHTGR design.

This reporting requirement affects nine or fewer respondents and, therefore, is not subject to Office of Management and Budget review under P.L. 96-511. If you have any questions regarding this request, or want a copy of the documents not included in Appendices C through J of Volume 2, please contact me at (301) 415-1307.

Sincerely,

Jack Donohew, Project Manager
Standardization Project Directorate
Division of Reactor Program Management
Office of Nuclear Reactor Regulation

Project No. 672

Enclosure: Draft PSER

cc w/enclosure:
See next page

DISTRIBUTION:
Central File
PDST R/F
MHTGR R/F
DMCrutchfield
BGrimes
TQuay
JNWilson
JMoore, O-15 B18
ACRS (11)
LShotkin
WDean, O-17 G21

DOCUMENT NAME: A: PSER-DOE.REV
To receive a copy of this document, indicate in the box: "C" = Copy without attachment/enclosure "E" = Copy with attachment/enclosure "N" = No copy

OFFICE	PM:DRPM:PDST	E	SC:DRPM:PDST	N	D:DRPM:PDST	C
NAME	JDonohew:sg		JNWilson		TRQuay	
DATE	02/23/96		02/ /96		02/ /96	

MHTGR
Project No. 672

cc: Mr. Sterling M. Franks
Acting Director
Office of Nuclear Energy, NE-45
U.S. Department of Energy
Washington, DC 20585

Salma El-Safwany
U.S. Department of Energy
San Francisco Operations Office
1301 Clay Street
Oakland, CA 94612-5208

Dr. Daniel L. Mears
Gas-Cooled Reactor Associates
10240 Sorrento Valley Road, Suite 300
San Diego, CA 92121-1605

Mr. Dave Pettycord
Modular HTGR Plant Design
Control Office - West
P.O. Box 85608
San Diego, CA 92186-9784

Mr. Ray R. Mills
Modular HTGR Plant Design
Control Office - East
3206 Tower Oaks Boulevard
Suite 300
Rockville, MD 20852

A. J. Neylan, Vice President
Power Reactor Group
General Atomics
P.O. Box 85608
San Diego, CA 92186-9784

Sunil Ghose
Bechtel National, Inc.
50 Beale Street
P.O. Box 193965
San Francisco, CA 94119

Mr. Walter J. Parker
Stone and Webster Engineering
Corporation
245 Summer Street
Boston, MA 02107

Sten Caspersson
ABB/Combustion Engineering
1000 Prospect Hill Road
Windsor, CT 06095

Mr. Phil Rittenhouse
Oak Ridge National Laboratory
P.O. Box 6138
Building 4500S
Oak Ridge, TN 37831-6138

Mr. Steve Goldberg, Budget Examiner
Office of Management and Budget
725 17th Street, NW
Washington, DC 20503

Dr. Gerald Garvey
Office of Science and Technology
OEOB, White House
Washington, DC 20500

NUREG-1338

Preapplication Safety Evaluation Report for the Modular High-Temperature Gas-Cooled Reactor (MHTGR)

Draft Copy of the Final Report
Volume 1: Chapters 1 through 9

Manuscript Completed : June 1995
Date Published: December 1995

Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

ABSTRACT

This preapplication safety evaluation report (PSER) documents the results of the preapplication review, from 1986 through the present, of the conceptual design of the Modular High-Temperature Gas-Cooled Reactor (MHTGR) design, Nuclear Regulatory Commission (NRC) Project No. 672. The Department of Energy (DOE), the preapplicant for the MHTGR design, submitted the advanced reactor design to NRC in response to the Commission's Advance Reactor Policy Statement (51 *Federal Register* 24643). This policy statement provides for the early NRC review and interaction with designers and preapplicants before the submittal of a design approval application for preliminary design approval, final design approval, or standard plant design certification under 10 CFR Part 52.

The MHTGR reactor plant design is a small, modular, graphite-moderated, helium cooled, high temperature, thermal-power reactor plant design similar to (but at a lower power density than) that of Fort St. Vrain plant, which was licensed by NRC. The standard plant design consists of four identical reactor modules, each rated at 350 MW(t), coupled with two steam-generator sets. The total plant electrical output rating is 540 MW(e), with a power conversion efficiency of about 39 percent. Each module is located in its own below-grade silo. The most significant departures from current light-water-reactor (LWR) plants is the encapsulation of the fuel in small multi-coated microspheres (as was the case for Fort St. Vrain core) and a completely passive safety-related decay heat removal system (as was not the case for Fort St. Vrain).

The review approach and criteria used by the staff was directed toward meeting the guidance in the Commission's Advance Reactor Policy Statement which states that advanced reactors must, as a minimum, provide the same degree of protection for the public and the environment that is required by current-generation LWRs. As defined in NUREG-1226, current-generation LWRs are the evolutionary designs which have been reviewed by NRC for standard plant designs under 10 CFR Part 52, such as the Advanced Boiling Water Reactor. Further, the Commission expects advanced reactors to provide enhanced margins of safety. This PSER addresses how the MHTGR meets the Commission's Advance Reactor Policy Statement.

Because of timing and resource limitations, the preapplication review for the MHTGR was directed toward meeting the objective in the Policy Statement that licensing guidance should be provided to the reactor designers while the design is being developed, before a design approval application is submitted. The discussions in the PSER on issues involving the evolutionary LWRs and passive advance LWRs that were considered applicable to the MHTGR design were to provide as much licensing information for the MHTGR as practicable.

The PSER is based on draft NUREG-1338 issued by NRC on the MHTGR in 1989, contractor reports completed since draft NUREG-1338 was issued, reports completed on other high temperature gas-cooled reactors, such as Fort St. Vrain, reports completed on the evolutionary LWRs and passive advanced LWRs on matters relevant to the MHTGR, and the Commission guidance for the advanced reactors, including the evolutionary LWRs and passive advanced LWRs.

The staff has used and built upon the applicable existing regulations and guidelines for safety to develop additional criteria when necessary to address the unique characteristics of the MHTGR and to ensure that the unique characteristics were assessed for enhanced safety margins in comparison with the current-generation LWRs. In the application of the existing regulations and guidelines, the staff, in some cases, has had to interpret the guidance developed for LWRs for application to the MHTGR. In making such interpretations, the staff maintained limits and criteria at least equivalent to those of current-generation LWRs, providing for conservatisms to account for plant-specific uncertainties in the designs. The staff also maintained consistency with the guidance being developed for the evolutionary LWRs and passive advanced LWRs in the treatment of severe accidents.

The PSER does not cover all aspects and systems of the design, including the balance of plant and areas in which the technologies to be used are the same or similar to those approved for Fort St. Vrain.

The PSER discusses licensability and policy issues, and provides an assessment of the designer's proposed general design criteria which, in the designer's judgement, apply to the design. The staff also reviewed confirmatory research and development programs and plans for prototype testing. The PSER identifies areas in which additional information will be required to support a design approval application. The overall conclusion is that DOE has not demonstrated the necessary performance required of the fuel to assure the licensing of the MHTGR design; however, an MHTGR design could be licensed with a lower level of fuel performance than currently proposed and a more leak-tight containment.

CONTENTS

3. IDENTIFICATION OF LICENSABILITY ISSUES (continued)

4. LICENSABILITY ISSUES

5. POLICY ISSUES

5. POLICY ISSUES (continued)

6. NRC CONTRACTOR REPORTS

APPENDICES (continued):

List of FIGURES

List of TABLES

ABBREVIATIONS

ABB	Asea Brown Boveri
ABWR	Advanced Boiling Water Reactor (reactor design)
ACI	American Concrete Institute
ac	alternating current (electric power)
ACRS	Advisory Committee on Reactor Safeguards
AEC	Atomic Energy Commission (U.S.)
ALARA	as low as is reasonable achievable
ALWR	advanced light-water reactor
ANSI	American National Standards Institute
AOO	anticipated operational occurrence
ASME	American Society of Mechanical Engineers
ASME Code	ASME Boiler and Pressure Vessel Code
ATWS	anticipated transient without scram
AVR	Arbeitsgemeinschaft Versuchs Reaktor (German reactor)
BE	bounding event
BISO	Trade name for HTGR multi-coated microsphere fuel
BLBE	beyond licensing basis event (DOE term)
BNL	Brookhaven National Laboratory
BOP	balance of plant
BTP	Branch Technical Position
CE	Combustion Engineering
CFR	Code of Federal Regulations
CLR	core lateral restraint
COL	combined operating license
CR	control room
DBA	design basis accident
DBE	design basis event
dc	direct current (electric power)
DHRS	decay heat removal system
DMS	data management system
DOE	Department of Energy (U.S.)
D-RAP	design reliability assurance program
EAB	exclusion area boundary
EC	event category
ECA	Energy Conversion Area
EDCPS	essential dc power system
EDO	Executive Director for Operations (NRC)
EES	economizer-evaporation superheater

EPA	Environmental Protection Agency (U.S.)
EPBE	emergency preparedness bounding event (DOE term)
EPRI	Electric Power Research Institute
EPZ	emergency planning zone
EUPSS	essential uninterruptible power supply system
FDA	final design approval
FIMA	fissions per initial metal atom
FR	*Federal Register*
FRG	Federal Republic of Germany
FS	finishing superheater
FSAR	final safety analysis report
FSER	final safety evaluation report
FWS	feedwater supply
GA	General Atomics
GASSAR	General Atomics Standard Safety Analysis Report
GCRA	Gas-Cooled Reactor Associates
GDC	general design criteria
GLRWS	gaseous and liquid radioactive-waste system
GRWS	gaseous radwaste system
GT-MHR	Gas Turbine-Modular Helium Reactor
HEPA	high efficiency particulate filter
HPS	helium purification system
HTGR	high temperature gas-cooled reactor
HTS	Heat Transport System
HVAC	heating, ventilation, and air conditioning
I&C	instrumentation and control
INCA	inner neutron control assembly
ITAAC	inspection, testing, and acceptance criteria
LBP	lumped burnable poison
LEU	low enriched uranium
LNS	Liquid Nitrogen System
LOCA	loss-of-coolant accident
LOFC	loss-of-forced-cooling
LR	letter report (NRC contractor report)
LRWS	liquid radwaste system
LTA	low-temperature adsorber
LWR	light water reactor

MC	main circulator
MCIG	miscellaneous control and instrumentation group
MCS	main circulator system
MCSS	metallic core support structure
MHTGR	Modular High Temperature Gas-Cooled Reactor
MLSV	main loop shutoff valve
MSS	main steam system
MSSSs	main steam and feedwater supply systems
MW(t)	megawatt thermal
MW(e)	megawatt electric
NCA	neutron control assembly
NCSS	neutron control subsystem
NDTT	nil-ductility transition temperature
NEACDS	nonessential ac distribution system
NEDCPS	nonessential dc power system
NEUPSS	nonessential uninterruptible power supply system
NI	Nuclear Island
NICWB	nuclear island cooling water building
NPR	New Production Reactor
NRC	Nuclear Regulatory Commission (U.S.)
NRR	NRC Office of Nuclear Reactor Regulation
NSSS	nuclear steam supply system
OBE	operating basis earthquake
ONCA	outer neutron control assembly
O-RAP	construction/operation-reliability assurance program
ORNL	Oak Ridge National Laboratory
PAGs	protective action guides
PCDIS	plant control, data, and instrumentation system
PCRV	prestressed concrete reactor vessel
PDA	preliminary design approval
PDCO	Plant Design Control Office (DOE contractor)
PFPAS	plant fire protection and alarm system
PFPCDS	plant fire protection carbon dioxide subsystem
PFPHS	plant fire protection Halon subsystem
PFPS	plant fire protection system
PFPWS	plant fire protection water subsystem
PIUS	Process Inherent Ultimate Safety (reactor design)
PPICS	plant protection, instrumentation, and control system
PPIS	plant protection and instrumentation system
PRA	probabilistic risk assessment
PRISM	Power Reactor Innovative Small Module (liquid metal reactor)
PSAR	preliminary safety evaluation report
PSB	Personnel Service Building
PSC	Public Service Company of Colorado

PSCS	plant supervisory control subsystem
PSER	preapplication safety evaluation report
PSID	Preliminary Safety Information Document
PSR	permanent side reflector
RAB	Reactor Auxiliary Building
RAP	reliability assurance program
RB	Reactor Building
RCCS	Reactor Cavity Cooling System
RCPB	reactor coolant pressure boundary
RCSS	reactor core subsystem
RES	NRC Office of Nuclear Regulatory Research
RG	regulatory guide
RPCWS	reactor plant cooling water system
RPS	reactor protection system
RSA	remote shutdown area
RSB	Reactor Service Building
RSCE	reserve shutdown control equipment
RTDP	Regulatory Technology Development Plan
RTNSS	regulatory treatment of non-safety-related systems
RV	reactor vessel
RVACS	reactor vessel auxiliary cooling system
RWMB	Radioactive Waste Management Building
SAMDA	severe accident mitigation design alternative
SAR	safety analysis report
SBWR	Simplified Boiling Water Reactor (reactor design)
SCCS	shutdown cooling circulator subsystem
SCHES	shutdown cooling heat exchanger subsystem
SCS	Shutdown Cooling System
SCWS	shutdown cooling water subsystem
SD	shutdown
SECY	NRC Commission paper
SG	steam generator
SGIV	steam generator isolation valve
SGV	steam generator vessel
SLSV	shutdown loop shutoff valve
SNAIS	special nuclear area instrumentation subsystem
SPSS	safety protection subsystem
SRDC	safety-related design condition
SRM	staff requirements memorandum
SRP	standard review plan
SRWS	Solid Radwaste System
SSCs	structures, systems, and components
SSE	safe shutdown earthquake
SU	startup
SWDS	Steam and Water Dump System
SWS	Service Water System

TDN	technology development need
TDP	technology development program
TER	technical evaluation report (NRC contractor report)
THTR	Thorium High Temperature Reactor (German reactor)
TID	technical information document
TRISO	Trade name for MHTGR multi-coated microsphere fuel
UofT	University of Tennessee
U.K.	United Kingdom
UPS	uninterruptible power supply
UPTPS	upper plenum thermal protection structure
USS	ultimate shutdown system
VS	vessel system
V&V	verification and validation

1. INTRODUCTION

1.1 Introduction

The staff of the United States Nuclear Regulatory Commission (NRC) has prepared this final preapplication safety evaluation report (PSER) to document its preapplication review of the Modular High Temperature Gas-cooled Reactor (MHTGR). In 1986, the Department of Energy (DOE) submitted to NRC its conceptual design of the MHTGR, as part of its advanced gas-cooled reactor program. This report points out the licensability issues for the MHTGR design. For these issues, the MHTGR design departs significantly from current NRC practices and the resolutions of the issues could fundamentally alter the MHTGR design.

In 1986, DOE also submitted the Preliminary Safety Information Document (PSID) (DOE-HTGR-86-024) for the MHTGR design to NRC in accordance with the Commission's Advanced Reactor Policy Statement (51 FR 24643). From 1986 through 1992, DOE has amended the PSID, up to Amendment 13. The staff reviewed the PSID in accordance with the process and guidelines outlined in NUREG-1226, "Development and Utilization of the NRC Policy Statement on the Regulation of Advanced Nuclear Power Plants."

DOE also submitted information on the MHTGR in the following documents:

- DOE-HTGR-86011, "Probabilistic Risk Assessment for the Standard Modular High Temperature Gas-Cooled Reactor"
- DOE-HTGR-87001, "Emergency Planning Bases for the Standard Modular High Temperature Gas-Cooled Reactor"
- DOE-HTGR-86-064, "Regulatory Technology Development Plan for the Standard Modular High-Temperature Gas-Cooled Reactor
- DOE-HTGR-87089, "MHTGR Assessment of NRC LWR [Light Water Reactor] Generic Safety Issues"
- DOE-HTGR-90257, "MHTGR Fuel Process and Quality Control Description"
- DOE-HTGR-86004, "Overall Plant Design Specification Modular High Temperature Gas-Cooled Reactor"
- DOE-HTGR-88311, "Containment Study for MHTGR"
- DOE-HTGR-90321, "450 MW(t) MHTGR Source Term and Containment Study"

The initial phase of this preapplication review was conducted, from 1986 through 1990, by NRC's Office of Nuclear Regulatory Research (RES) because RES was identified in the Commission's Advanced Reactor Policy Statement as the focal point in NRC for preapplication reviews of advanced reactors. The second and last phase was conducted since 1991 by NRC's Office of Nuclear Reactor Regulation (NRR) because NRR became this focal point in 1990. The reference to the NRC staff in this report is a reference to the work done by RES and NRR from 1986 to the present.

The staff's preliminary findings about the MHTGR were based on the PSID through Amendment 10 and on the four licensing issues (accident selection, adequacy of containment, adequacy of offsite emergency planning, and siting source term) for the MHTGR in SECY-88-203, "Key Licensing Issues Associated with DOE Sponsored Advanced Reactor Designs." These findings were documented in draft NUREG-1338, "Draft Preapplication Safety Evaluation Report for the Modular High-Temperature Gas-Cooled Reactor," in March 1989. After 1989, the staff continued its review of the MHTGR design, had contractors review selected technical areas of the design, and identified policy and technical issues that required Commission guidance for design certification for the design (SECY-93-092), which superseded SECY-88-203. DOE also further amended the PSID in submitting Amendments 11, 12, and 13 (DOE-HTGR-86-024) in response to staff comments in draft NUREG-1338 and four staff requests for additional information (NRC letters dated August 8, 1991; March 4, 1992 (2 letters); and August 26, 1993).

This report is based on the staff's conclusions in draft NUREG-1338. However, in light of the Commission's policy decisions on advanced reactors, the contractor reports on the MHTGR design, and the completion of a preapplication review of an advanced reactor since draft NUREG-1338 was issued, the staff's conclusions in draft NUREG-1338 have been evaluated and discussed in this report to determine the licensability issues for the MHTGR design.

This report is not an approval of the MHTGR design, or any part of the design. It documents a preapplication review of the MHTGR design for the purpose of providing guidance early in the design process on the licensing acceptability of the design preceding the submittal of an application for NRC staff review under 10 CFR Part 52. This report is intended to aid DOE in developing an application for design approval (i.e., preliminary design approval (PDA) or final design approval (FDA) under Appendix O of 10 CFR Part 52, or standard plant design certification under Subpart B of 10 CFR Part 52).

The Commission can only make a licensing determination on a standard reactor plant design after the design has been submitted for review under 10 CFR Part 52. For this review, DOE will have to comply with the administrative processes for licensing standard nuclear power plants, including public notification and participation, required in Title 10, "Energy," and Title 40, "Protection of the Environment," of the *Code of Federal Regulations* (CFR).

1.2 Preapplication Review

The Commission developed its advanced reactor policy to encourage early interaction between the NRC and advanced reactor designers (i.e., the applicant for the preapplication review), and for NRC to establish licensing guidance applicable to these advanced designs. The policy is discussed in NUREG-1226 and the objectives of this policy were the following:

- Allow for the earliest possible interactions of the applicant and NRC

- Encourage greater safety margins through the use of simplified, inherent, passive, or other innovative means for safety design

- Provide all interested parties, including the public, with the Commission's views concerning the desired characteristics of advanced reactor designs

- Express the Commission's intent to issue timely comment on the implications of such designs for safety and the regulatory process

The staff issued NUREG-1226 to present and discuss the Commission's final version of its Advanced Reactor Policy Statement. It provides guidance to designers and the staff for the preapplication reviews.

As stated in NUREG-1226, the Commission's Advanced Reactor Policy applies to reactor designs for which licensing requirements are not covered by the light water reactor (LWR) standard review plans in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition." The Commission stated in NUREG-1226 that high-temperature gas-cooled reactor (HTGR) designs, such as the MHTGR, are advanced reactor designs. NUREG-0800 has no review plans for HTGR designs.

The preapplication review is an early interaction with NRC before the applicant submits its design for design approval. This review is performed and completed while the design is still being developed and lacks details that would be required in the application under 10 CFR Part 52. This review is not intended to be complete because it takes place at such an early stage of the design and because details of the design are still not determined by the designer. It is also not intended to be the basis for NRC accepting any part of the design under 10 CFR Part 52.

In Section 1.1 of the PSID, DOE stated that the MHTGR design was at a conceptual stage and that the level of detail and completeness of the supporting analyses and assessments were representative of this stage of design. The detail given in the PSID is less than that in a preliminary safety analysis report (PSAR) submitted with an application for a PDA under 10 CFR Part 52.

In its letter of May 4, 1994, DOE stated that its gas-cooled reactor program had been redirected to a gas turbine—modular helium reactor (GT—MHR) and was in the preconceptual stage. The GT—MHR is not the design described in the PSID, a steam generator—turbine generator design, and indicates the continuing evolution of the MHTGR design as is expected during the preapplication review stage. Because there is so little information on the GT—MHR design, and because of timing and resource limitations, the implications of the change to a GT-MHR concept are not addressed in this report. However, because the licensability issues discussed in Chapter 4 of this report are independent of whether the MHTGR concept has a steam generator or a gas turbine, these issues should also apply to the GT-MHR.

Therefore, the conclusions of the staff in this report, particularly in Chapter 4 on licensability issues and Chapter 5 on policy issues, are not intended to be complete discussions of these issues or to close out any reviews of the staff during a design approval review in the future. The staff conclusions given here merely report the staff's insights on licensability problems of the design at this time, and the designer is expected to address these problems in its PDA, FPA, or design certification application.

1.3 List of Meetings and Correspondence

NRC and DOE have met many times to discuss the MHTGR design. The meetings in the initial phase of the preapplication review are listed in Table 1.1 of draft NUREG-1338. Since draft NUREG-1338 was issued, the staff met with DOE on the following dates and discussed the following subjects:

• April 23, 1991	Restarting project and status of project
• October 23, 1991	Fuel performance and research
• December 17–20, 1991	Fuel performance and fission product transport
• January 22, 1992	Equipment safety classification
• February 20, 1992	Fuel performance
• June 24, 1992	Advanced reactor policy issues
• July 1, 1993	Final PSER schedule
• September 29, 1994	Final PSER content and schedule

The staff issued meeting summaries on June 24, 1991; January 13, March 10, April 10 and 15, and August 20, 1992; July 8, 1993; and October 7, 1994, respectively. DOE also presented details of the MHTGR design to the NRC staff on June 4, 5, and 6, 1991, and on May 23 and 24, 1994. The staff issued summaries of these presentations on July 31, 1991, and July 12, 1994, respectively.

DOE has also met with NRC's Advisory Committee on Reactor Safeguards (ACRS) on the design. The meetings conducted before draft NUREG-1338 was issued are listed in Table 1.1 of that document. The meetings held after draft NUREG-1338 was issued are listed in Appendix A of this report.

The chronology of correspondence between the NRC and DOE is in (1) Chapter 18 of draft NUREG-1338, for the initial phase of the preapplication review of the MHTGR, and (2) Appendix A of this report, for the correspondence after draft NUREG-1338 was issued.

1.4 Consistency of DOE's Approach with NRC's Advanced Reactor Policy

The Commission's objectives for the preapplication review of advanced reactor designs are stated in Section 1.2 of this report. In NUREG-1226, the Commission further stated that designers of advanced reactors should consider the NRC regulations, regulatory guides, and other established guidelines as the defense-in-depth philosophy, standardization, the Commission's safety goal and severe-accident policies, and industry codes and standards. Guidance was given in NUREG-1226 on these and on such other considerations as operating experience, technology development programs, probabilistic risk assessment, and prototype tests.

DOE has addressed the NUREG-1226 considerations in the PSID and other supporting documents submitted to the staff on the MHTGR design. However, in these documents, DOE has submitted the information under its Applied Technology designation which restricts the dissemination of information on the MHTGR design. This keeps a significant amount of information on the MHTGR design from the public; dissemination of such information is one of the objectives of the Commission's Advanced Reactor Policy Statement. The Applied

Technology designation and the information being withheld from the public is discussed in more detail in Section 1.8 of this report.

Except for the continued use of the Applied Technology designation, DOE has been fully responsive to the Commission's Advanced Reactor Policy Statement.

1.5 DOE Approach to the MHTGR and DOE Objectives

The objectives of the DOE MHTGR project, as stated in Chapter 1 of the PSID (DOE-HTGR-86-024) and in the "Concept Description Report" (DOE-HTGR-86-118), are to develop a safe, economical plant design by the turn of the century that meets both NRC and utility (or user) power plant requirements by providing defense in depth through the pursuit of the following four general goals:

- Maintain plant operation by reliably maintaining the plant functions necessary for startup, shutdown, operation, and refueling.
- Maintain plant protection by providing design features or systems to prevent plant damage.
- Maintain control of radionuclide releases by providing design features or systems to ensure containment of radionuclides.
- Maintain emergency preparedness by providing adequate means to protect the health and safety of the public.

DOE stated that its objectives were to develop an advanced reactor design with passive safety characteristics that would be reliable, economical, and competitive with alternative electric power generation sources available to the electric utility industry for large power plants and would also be deployable in small, or modular, increments of power.

DOE also stated that the overall programmatic objective for the MHTGR is its development for a broad range of applications using the following unique safety and high-temperature characteristics of the MHTGR design:

- reduced power density
- large negative core doppler coefficient
- large core heat capacity providing slow responses to core-heatup events
- very high core temperatures before degradation of the fuel
- inert, single phase, and non-core-reactivity-effects coolant
- minimal reliance on active systems and operator actions

These characteristics lead to a nuclear power reactor design that DOE stated would have the following attributes:

- simplified plant design compared to that of current LWRs
- inherent passive reactor-shutdown and decay-heat-removal features
- reduced need for operator action
- insensitivity to operator error
- long time intervals for operators to take corrective actions

DOE stated that given these characteristics the preliminary MHTGR design would reduce the number of structures, systems, and components that would need to be classified as safety related in comparison to both LWRs and other HTGRs.

DOE's safety philosophy (i.e, to reach the stated goal of maintaining control of radionuclide release) for the MHTGR is to provide defense in depth by conforming to the following very broad ("top-level") regulatory criteria:

- Commission's "Policy Statement on Safety Goals" for the operation of nuclear power plants (51 FR 28044)
- 10 CFR Part 20, "Standards for protection against radiation."
- Appendix I to 10 CFR Part 50, numerical guidelines for nuclear power plant effluents to meet the "as low as is reasonably achievable" standard
- 40 CFR Part 190, environmental radiation protection standards for nuclear power plant operations
- 10 CFR Part 100, "Reactor Site Criteria"
- Environmental Protection Agency (EPA) "Protective Action Guides [PAGs] and Protective Actions for Nuclear Incidents

DOE stated that the basis for these criteria was that such criteria are quantifiable and sufficient to protect the health and safety of the public, independent of the reactor type and site. DOE concluded in its preliminary probabilistic risk assessment (DOE-HTGR-86011) that the MHTGR will behave in a benign manner with limited offsite releases during even extremely unlikely (i.e., severe) accidents.

DOE stated that the MHTGR design goals exceed the Commission's safety goal objectives for nuclear power plants (51 FR 28044). These goals are the following: (1) a core damage probability of less than 1 in 10,000 per year of reactor operation and (2) a large release probability from a nuclear accident of less than 1 in 1 million per year of reactor operation. DOE stated that the design is orders of magnitude within the latent fatality limit of the Commission's safety goal objectives and has no acute fatality risk (DOE-HTGR-86011, Section 9.3.2).

The core will be designed with a low-power density so that core temperatures should remain below values causing fuel damage, or melting, and the core will not lose a coolable geometry. The fission products released from the fuel will not exceed the lower level EPA PAGs at the plant site boundary in even extremely unlikely accidents. The intact microspheres containing the fuel will retain the fission products, and the increase in fuel failures at higher fuel temperatures during accidents will be insignificant.

DOE stated that the approach to safety for the design was to apply what it called a "top-down" approach. This began with the identification of the broad ("top-level") regulatory requirements stated above. Next, the specific functions, requirements, and design selections to achieve the top-level requirements are developed using an integrated systems engineering approach to the design. The application of the top-level regulatory criteria to the "goals" of the MHTGR design listed above were presented in Table 3.1-1 of the PSID. The product, or lower-level result, is the specific plant design described in the PSID. The staff review, however, is not concerned with the user requirements for the design, except where these requirements affect the safety aspects of the design.

DOE is developing the MHTGR with the support of a user utility group, Gas Cooled Reactor Associates, and a team of contractors. This team comprises the nuclear steam supply system (NSSS) vendors (General Atomics and Asea Brown Boveri—Combustion Engineering, ABB—CE), the architect-engineers (Bechtel National, Inc., and Stone and Webster Engineering Corporation), and the research and development support of Oak Ridge National Laboratory and EG&G Idaho, Inc. General Atomics is responsible for the design of the reactor, fuel, and primary-system equipment. ABB—CE is responsible for the design of the reactor, cross duct, and steam generator system. Bechtel is responsible for the design of the nuclear island. Stone and Webster is responsible for plant control and the balance of plant (i.e., the non-NSSS part of the design).

DOE's activities with NRC were initiated in June 1984 with a technical briefing on the MHTGR design. Among other DOE activities with NRC before the PSID was submitted in 1986 were the submittal of a draft licensing plan to the staff; briefings of the staff on design criteria, accident-selection criteria, safety criteria, and concept criteria; and briefing the Subcommittee on Advanced Reactors of the Advisory Committee on Reactor Safeguards (ACRS) and the full ACRS on design and programmatic objectives. Table 1.1 of draft NUREG-1338 lists the selected milestones in the MHTGR review process by both NRC and DOE from 1984 through 1988.

1.6 General Plant Description

A general description of the MHTGR plant is provided below in this section. A detailed technical description from draft NUREG-1338 is in Appendix B of this report.

The MHTGR will be a helium-cooled and graphite-moderated thermal power reactor. Figure 1.1 is a cut-away view of the reactor module. The fuel is composed of millions of ceramic coated microspheres (four major coatings or layers) held in a cylindrical organic binder and placed in hexagonal graphite blocks. The helium is a single phase coolant, chemically inert, and neutronically inert. Although it will be graphite moderated, the MHTGR will have no water in the core lattice as did the Chernobyl pressure tube reactor.

The most significant departures from current LWRs will be (1) the encapsulation of the fuel in small multi-layered microspheres of diameters about 575 um (fissile fuel) and 690 um (fertile fuel) and (2) the completely passive safety-grade decay heat removal system discussed in Sections 3.2.3.2, 5.2.6, and 4.2.6 of this report. The fuel particles are designed to maintain their integrity during normal operation and at elevated temperatures under

350 MW(t) MODULAR HTGR ELEVATION

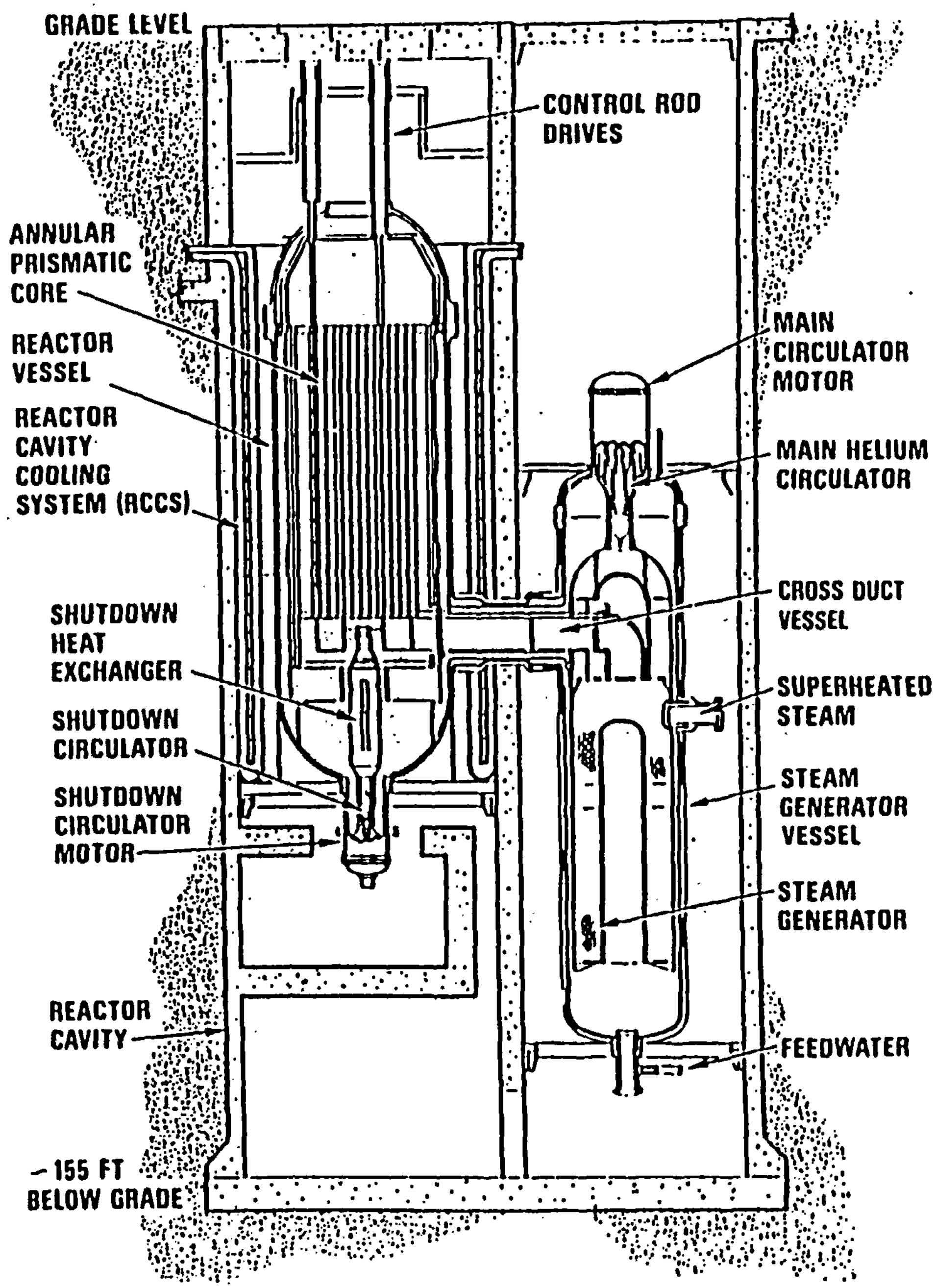


FIGURE 1.1 MHTGR REACTOR MODULE

transient conditions or under conditions of chemical attack by water or air. Figure 1.2 depicts the fuel, which is low-enriched uranium (enriched to about 19.9 percent), although significantly more enriched than LWR fuel. The fuel is in TRISO (trade name) multicoated microspheres (with a silicon carbide, SiC, layer) which has been used in the following reactors: Fort St. Vrain, Thorium High Temperature Reactor (THTR), and Arbeitsgemeinschaft Versuchs Reaktor (AVR).

The standard plant design consists of four identical reactor modules coupled with two steam turbine-generator sets. The simplified flow diagram for a reactor module is shown in Figure 1.3. Each module will be designed for a thermal output of 350 megawatts thermal, or MW(t), and a total plant (four modules) electrical output of 540 megawatts electric, or MW(e). This is about half the output of current LWR plants. The major plant characteristics are listed in Table 1.1.

The plant site would be separated into the Nuclear Island and the Energy Conversion Area. The Nuclear Island would be the portion of the plant containing the reactor modules, steam generators, and safety-related structures, systems, and components dedicated to reactor shutdown, heat removal, fission-product retention, and security, and will be the nuclear protected and secured area. The Energy Conversion Area will be the remaining portion of the plant outside the Nuclear Island, containing the turbine-generators and transmission equipment.

The reactor core would be low-power density (5.9 w/cc) and have the coated micro fuel particles in small organic bonded cylindrical compacts. The compacts are placed in small vertical holes in the hexagonal graphite block fuel assemblies. The fuel assemblies are cooled through passages in the blocks. There are about 660 graphite blocks in the 66-column annular core region between the inner and outer reflector regions.

The core will be in a steel vessel located, with the steam generator vessel, in the reactor building below ground to reduce seismic loads on the vessel and core. The reactor vessel will be above the steam generator vessel to prevent natural circulation (which prevents the hottest helium gas from contacting the inner surfaces of the two vessels) and will be connected to this vessel by a horizontal crossduct vessel. The reactor and steam generator vessels will be in separate cavities of the reactor building. The secondary-side water will be superheated in the steam generator. The secondary-side pressure will be higher than pressure on the primary side, so water would leak into the coolant with a steam generator tube leak or rupture.

There will be two safety-related reactor protection systems (RPSs) (control rods and boron carbide balls), which will be diverse and redundant, and one non-safety-related system (control rods). The non-safety-related system will be independent from and redundant to the safety-grade ones.

According to the PSID if the core heated up during an event, such as water inleakage, it would be immediately shut down from the rising core temperature and the large prompt Doppler coefficient . If both non-safety-related, active decay heat removal systems (DHRSs) failed to function, decay heat would be

FIGURE 1.2 MHTGR REFERENCE FUEL

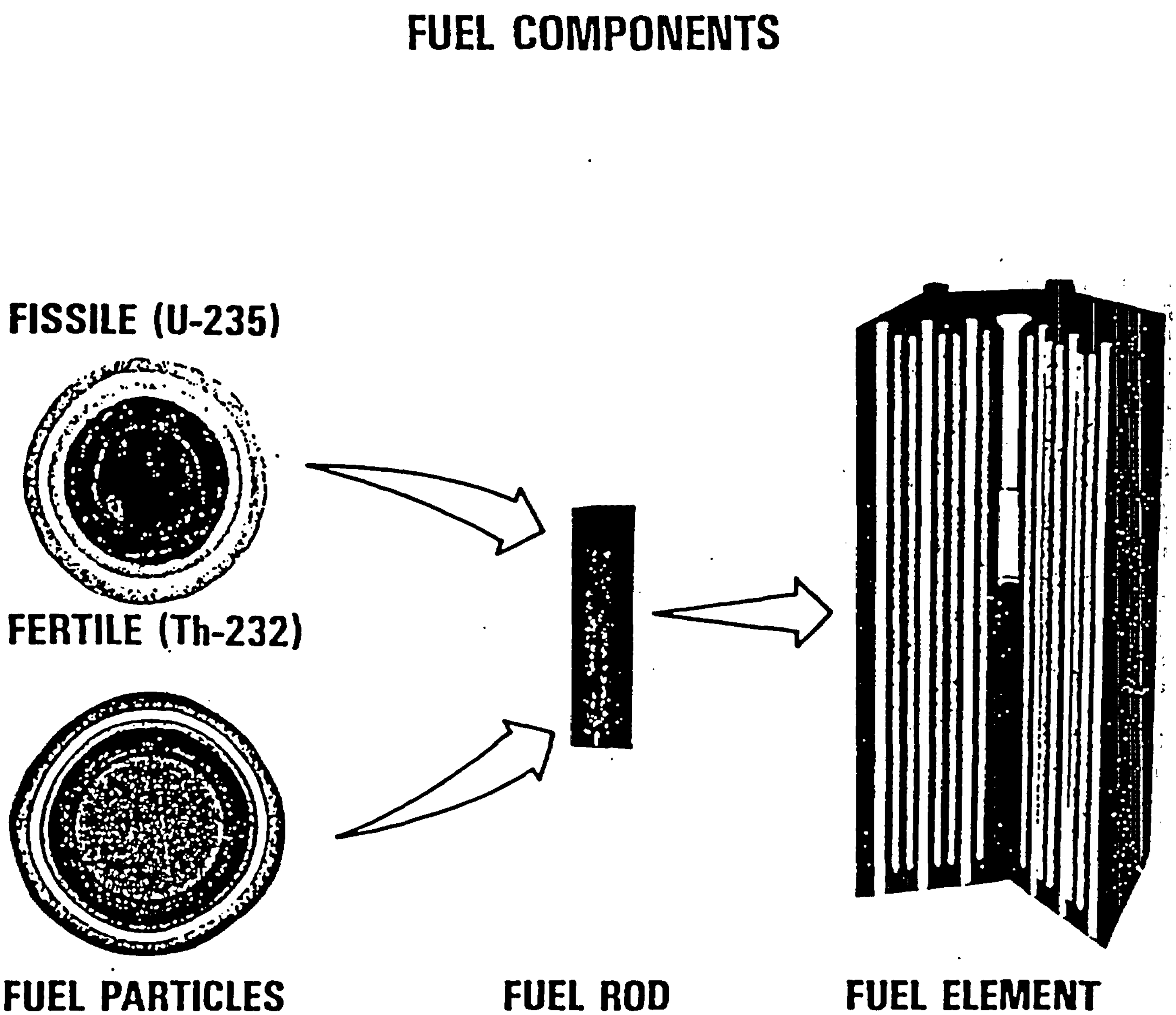


FIGURE 1.3 SIMPLIFIED MHTGR REACTOR MODULE FLOW DIAGRAM

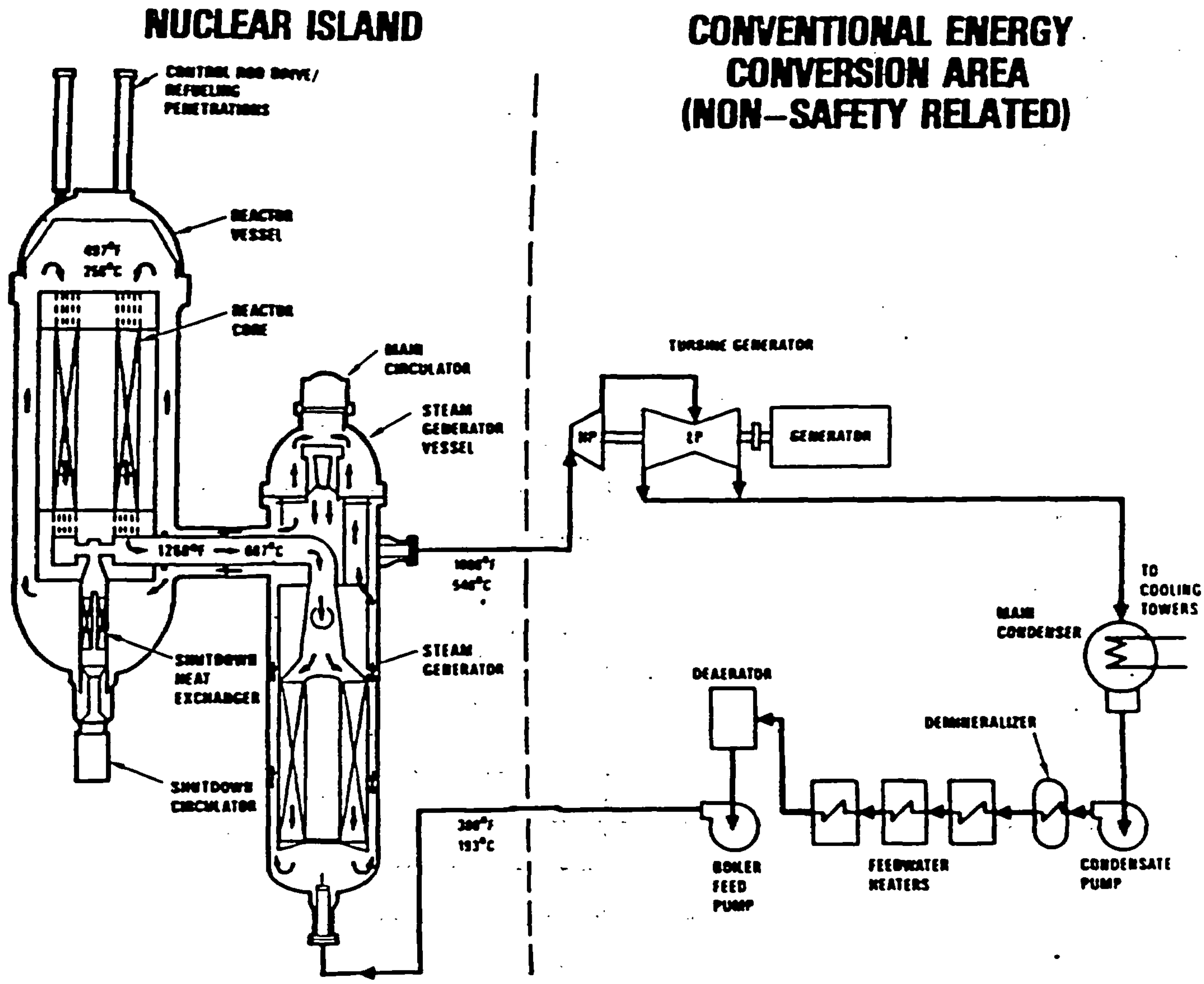


TABLE 1.1 MHTGR PLANT CHARACTERISTICS AND DESIGN DATA

Plant Characteristic	Design Data
Number of reactors per module	1
Number of modules	4
Net electrical output	550 MW(e)
Steam pressure	2500 psig
Net station efficiency	39.2%
Reactor Module	
Thermal power	350 MW(t)
System pressure	6.4 MPa (925 psig)
Core inlet/outlet temperatures	259/687°C (497/1268°F)
Fuel temperature (max/ave)	1060/677°C (1940/1250°F)
Reactor Core	
Fuel	$UCO+ThO_2$ microparticles
Coating	Ceramic (PyC/SiC/PyC)
Moderator	Graphite
Coolant	Helium
Power density	5.9 w/cc

conducted through the core to the reactor vessel and radiated to the passive safety-grade Reactor Cavity Cooling System (RCCS) panels surrounding the vessel. If the RPSs failed to function, the core would slowly return to critically in about 38 hours with the decay of xenon and the cooling of the core. With the return to criticality, the core power level would rise to a peak of about 20 to 25 MW(t) and then decrease to an equilibrium level of 3 MW(t) with increasing core temperature. In all cases, the RCCS would be designed to keep core fuel and reactor vessel temperatures below critical values. DOE has stated that the licensed operators will not be required to take any action.

As shown in the PSID, the RCCS, the safety-related decay heat removal system, will consist of panels surrounding the reactor vessel below grade with a header connection to four inlet and exhaust ports located above grade. This will allow hot air to rise, thus removing heat radiated from the reactor vessel while cold air would be drawn from outside into the panels. The RCCS will be entirely passive with no moving components. It will operate continuously, will not be capable of being turned off or on, will automatically respond to rising temperatures in the core through thermal radiation and natural circulation, and will have flow path redundancy to the cooling panels through the cross-connected header. In addition, there will be two other highly reliable, but non-safety-related, active heat removal systems: (1) the shutdown cooling system in the bottom of the reactor vessel and (2) the main circulator/steam generator in the primary cooling loop.

The barriers to the release of fission product radioactivity will be the multiple-coated fuel microspheres, the reactor coolant pressure boundary (RCPB), and the containment. DOE proposed that the microspheres function as both the initial fission-product barrier and the containment system for fission products.

The containment will be the reactor building below ground with containment isolation valves to isolate the secondary side from the steam generator. It will be a vented, high-leakage structure containing the reactor vessel and steam generator vessel with dampers that will open to relieve the pressure pulse following a depressurization of the RCPB or a steam-line break. It will not be a conventional, essentially leaktight, LWR containment in that the MHTGR containment will not retain the gases released from a rapid RCPB depressurization and is designed to have a leak rate of less than one building volume per day after this depressurization.

A non-safety-related helium purification system will pressurize and depressurize the RCPB, and purify the coolant.

1.7 Comparison with Other High Temperature Gas-Cooled Reactors

Worldwide experience in the last 50 years with gas-cooled nuclear power reactors has been considerable. This experience base contains the operation of a number of HTGR facilities. Several major power facilities and their characteristics, including the MHTGR, are listed in Table 1.2, on the next two pages.

TABLE 1.2 COMPARISON OF HTGR DESIGNS

Features	MHTGR[a]	Fort St. Vrain	THTR[a]	AVR[a]	Peach Bottom 1	Dragon	GASSAR[a]
<u>Design Origin</u>	U.S.	U.S.	FRG	FRG	U.S.	U.K.	U.S.
<u>Years of Power Production</u>	None	1976-1989	1985-1989	1967-1989	1967-1974	1966-1975	Plant not built
<u>Plant Output</u>							
MW(t)/MW(e)[b]	4 x 350/540	842/330	750/300	46/15	115/40	20/0	3000/1120
<u>Reactor Core</u>							
Active core dimensions (m)							
Diameter	3.5 (OD)[c] 1.65 (ID)[c]	6.0	5.6	3.0	2.8	1.1	8.5
Height	7.9	4.8	6.0	2.5	2.5	1.6	6.8
Core power density (W/cc)	5.9	6.3	6.0	2.5	8.3	14.0	8.4
Fuel cycle	LEU/Th[d] 19.9% enriched	HEU/Th[d]	HEU/Th[d]	HEU/Th[d] and LEU	LEU/Th[d]	LEU/Th[d]	HEU/Th[d] 93% enriched
<u>Reactor Vessel</u>	Steel	PCRV[e]	PCRV[e]	Steel	Steel	Steel	PCRV[e]
<u>Primary Cooling System</u>							
Pressure (bar)	64	48	40	11	24	17	50
Core inlet gas temp (°C)	260	405	250	270	340	350	319
Core exit gas temp (°C)	690	785	750	950	725	750	755
<u>Fuel</u>							
Fissile particle	UCO[f] TRISO[f]	(Th,U)C_2 TRISO[f]	(Th,U)O_2 BISO[f]	(Th,U)C_2 BISO[f]	(Th,U)C_2 BISO[f]	UO_2-TRISO[f] (Zr,U)C	UC_2 TRISO[f]
Fertile particle	ThO_2-TRISO[f]	ThC_2-TRISO[f]	(Th,U)O_2 BISO[f]	(Th,U)O_2 BISO[f]	(Th,U)C_2 BISO[f]	(Th,U)C BISO[f]	ThO_2 TRISO[f]
Fuel element type	Prism	Prism	Sphere (pebble bed)	Sphere (pebble bed)	Cylinder	Hexagonal rods pin-in-block	Prism
Fuel element lifetime (yr)	3.3	6	3	3	3	Varies	4

TABLE 1.2 COMPARISON OF HTGR DESIGNS (continued)

Features	MHTGR	Fort St. Vrain	THTR	AVR	Peach Bottom-1	Dragon	GASSAR
<u>Circulator</u>							
Number	4 (1 per module)	4	6	2	2	6	6
Compressor type	Single-stage axial	Single-stage axial	Single-stage radial	Single-stage radial	Single-stage radial	Single-stage radial	Single-stage axial
Bearing	Magnetic	Water	Oil	Oil	Oil	Gas	Water
<u>Steam Generator</u>							
Number	4 (1 per module)	12	6	1	2	6	6
Type	Helical, non-reheat	Helical, with gas reheat	Helical, with gas reheat	Evolvent, non-reheat	U-tube, with steam drum	Helical heat exchanger	Helical, with gas reheat
<u>Residual Heat Removal</u>							
Primary	Main loop	Two separate main loops	Two separate main loops	Main loop	Two separate main loops	Main loop	Two separate main loops
Second	Shutdown Cooling System	Main loop with alternate motive force	Main loop with alternate motive force	Vessel cooling	Main loop with alternate motive force	Emergency natural circulation boiler	Core Auxiliary cooling system
Tertiary	Reactor Cavity Cooling System	PCRV* liner cooling	None	None	Reactor vessel cooling panels	None	None
Reactor building	Confinement below grade vented to atmosphere	Confinement above grade vented to atmosphere	Confinement above grade vented to atmosphere	Containment above grade	Containment above grade	Containment above grade	Containment above grade

Source: draft NUREG-1338 (as modified to bring the table up to date)

* Mixed fissile/fertile particle used

NOTES:

a	MHTGR	=	modular high-temperature gas-cooled reactor
	THTR	=	Thorium High Temperature Reactor
	AVR	=	Arbeitsgemeinschaft Versuchs Reaktor
	GASSAR	=	General Atomics Standard Safety Analysis Report
b	MW(t)	=	megawatt thermal
	MW(e)	=	megawatt electric
c	OD	=	outside diameter
	ID	=	inside diameter,
d	LEU/Th	=	low enriched uranium/thorium
	HEU/Th	=	high enriched uranium/thorium
e	PCRV	=	prestressed-concrete reactor vessel
f	UCO	=	uranium oxicarbide fuel
	TRISO	=	type of coated fuel particle (includes SiC layer)
	BISO	=	type of coated fuel particle (does not include SiC layer).

Commercial gas-cooled reactors began with the graphite-moderated, carbon dioxide—cooled "Magnox" reactors developed in the early 1950s in the United Kingdom (U.K.) and France. The first station in the U.K. consisted of the four units at Calder Hall, which first became operational in 1956 and which are expected to operate for a 40-year life. The U.K.'s investment in gas-cooled power reactors includes the 26 Magnox reactors, which use natural uranium fuel (22 are still in operation), and 14 advanced gas-cooled reactors,which use slightly enriched uranium. Japan's first nuclear power reactor, which began commercial operation in 1966, was a Magnox design; Japan is also currently building an experimental HTGR.

The HTGR concept was developed in the United States (late 1950's) with a graphite core, multi-layered fuel microspheres, prismatic fuel blocks, and helium coolant. This development resulted in the 40-MW(e) Peach Bottom Atomic Power Station (Unit 1), which operated from 1967 to 1974, and the 330-MW(e) Fort St. Vrain Nuclear Generating Station, which operated from 1976 to 1989.

In the U.K., the Dragon HGTR reactor operated from 1964 to 1977; it was a research reactor and did not produce electric power. Also, in the late 1950s, the Federal Republic of Germany (FRG) began designing the "pebble-bed" type of HTGR based on the coated-fuel developments in the United States. Two HTGRs have been operated in the FRG: the experimental 15-MW(e) AVR and the prototype THTR. The AVR operated from 1967 to 1989 and the THTR operated from 1985 to 1988.

There have been about 50 gas-cooled reactors worldwide, totaling about 1000 reactor-years of operation. Of this total, about 50 reactor-years have been with HTGRs.

The BISO and TRISO (trade names) multi-layered microspheres has been the fuel form for HTGRs. The BISO fuel form, a fuel kernel with essentially only two major layers, has been used at Peach Bottom 1, THTR, and Dragon. The TRISO fuel form, a fuel kernel with four major layers (including a silicon carbide, SiC, layer) and the reference fuel for the MHTGR, has been used at Fort St. Vrain, Dragon, and AVR. To meet higher fuel-integrity requirements, the TRISO design has replaced the BISO design in recent HTGR concepts, except at THTR.

DOE maintains agreements with the FRG, France, and Japan for the exchange of technical information; some of the data submitted to the NRC on the integrity of the MHTGR fuel is this type of information. Experiments have been conducted in France at Comēdie under such an agreement. These agreements are part of the Technology Development Program being conducted by DOE for the MHTGR design. This includes post-irradiation testing of development fuel at Oak Ridge National Laboratory.

Regulatory experience with HTGRs has been on designs that have been built and operated in the United States (i.e., Peach Bottom 1 and Fort St. Vrain), and that were proposed and never built (1000 MW(e) HTGR Study, Summit and Fulton plants, Gas-Cooled Fast Breeder Reactor, and GASSAR -- a General Atomics standard large HTGR plant). These are listed in Table 1.2 of draft NUREG-1338.

Major trends in recent HTGR designs, including the MHTGR, are the following: (1) increased system pressure; (2) the choice of steel pressure vessels for the smaller HTGRs, including the MHTGR, instead of the prestressed concrete reactor vessel (PCRV) used for larger HTGR designs as Fort St. Vrain; (3)

greater fuel integrity (i.e., smaller failed fuel fraction); and (4) lower enriched uranium fuel.

1.8 Applied Technology Designation

DOE's Applied Technology designation on MHTGR design information restricts the dissemination by NRC, or any other organization, of the information on the MHTGR as follows:

> Any Further Distribution by any Holder of this Document or of Other Data Herein to Third Parties Representing Foreign Interests, Foreign Governments, Foreign Companies, and Foreign Subsidiaries or Foreign Divisions of U.S. Companies Shall be Approved by the Associate Deputy Assistant Secretary for Reactor Systems, Development and Technology, U.S. Department of Energy. Further Release May Require DOE Approval Pursuant to Federal Regulation 10 CFR Part 810, and/or May be Subject to Section 127 of the Atomic Energy Act.

In complying with this designation, the staff has not placed Applied Technology information in the NRC Public Document Room and, therefore, has not released this information on the MHTGR to the public.

In its four letters dated March 12, July 8, July 23, and July 28, 1993, NRC has requested that DOE review documents that were prepared by NRC on the MHTGR design to determine if they contain Applied Technology information. In several instances, NRC has requested DOE by phone conference call to perform this review and DOE has responded to NRC by telephone. DOE reviewed draft NUREG-1338 before NRC issued it and determined that the document did not contain Applied Technology information.

DOE has designated most of the information it has submitted on the MHTGR design as Applied Technology information. Entire documents within submittals from DOE are designated this way. Applied Technology restrictions include not listing references containing Applied Technology information.

The staff, in its letter dated April 29, 1993, stated its concern about complying with the Commission's objective of public disclosure of advanced reactor designs, in the Commission's Advanced Reactor Policy Statement, if the staff's PSER was based on documents controlled by the Applied Technology designation. The staff also stated that the Applied Technology designation in itself was not sufficient for NRC to justify withholding the documents it was reviewing on the MHTGR in their entirety from the public.

Section 2.790 of 10 CFR requires that NRC make available, for copying and inspection in the NRC Public Document Room, all final NRC records and documents regarding NRC reviews in the absence of a compelling reason for nondisclosure to the public. These records and documents include, but are not limited to, correspondence to and from NRC. Section 2.790 does not specifically address Applied Technology information; however, it requires a "balancing of the interests of the person or agency urging nondisclosure and the public interests in disclosure" and Paragraphs 2.790(a) to (e) list exceptions to the disclosure requirement. Some of these paragraphs may apply to the Applied Technology information; however, DOE has not provided the basis for withholding design information from the public. For an applicant other than DOE, this designation would be determined to not meet the requirements of

10 CFR 2.790. Because this is a preapplication review (which is not approving any part of the MHTGR design), the staff has continued to work with DOE and to discuss NRC's concerns about the application of this designation; the staff has not made the Applied Technology information available for copying and inspection in the NRC Public Document Room.

In its response of May 26, 1993, DOE removed the Applied Technology designation from information submitted on the PRISM design and stated that it would delay removing the designation from the information for the MHTGR until the MHTGR PSER would be published.

In its letter of February 8, 1995, DOE removed this designation from the PSID and the probabilistic risk assessment report for the MHTGR (DOE-HTGR-86011). These documents and their amendments have been placed in the NRC Public Document Room. However, DOE's Applied Technology designation still applies to a significant amount of information that has been submitted on the MHTGR design; therefore, a significant part of the MHTGR information submitted to NRC is still not available to the public. This nondisclosure of significant portions of the design information on the MHTGR, apart from information withheld under 10 CFR Part 810 or the Atomic Energy Act, has prevented DOE from fully meeting the Commission objective in its Advance reactor Policy Statement of public disclosure of information on the MHTGR design.

The extensive application of the Applied Technology designation to the MHTGR design by DOE is a licensability issue for the design; it is discussed in Section 4.2.9 of this report.

1.9 DOE-Requested NRC Response

In Section 1.1.5 of the PSID, DOE stated that the overall licensability statement by NRC on the MHTGR design should address the following questions:

- Is the standard MHTGR design licensable?
- Are the interfaces between the standard Nuclear Island and the Energy Conversion Area, and the site appropriately identified and characterized?
- Are the top-level (broad) regulatory criteria acceptable and can they remain valid through final design approval?
- Is the methodology for proceeding from the top-level regulatory criteria through risk assessments and other safety analyses to the licensing basis acceptable, and can it remain in use through final design approval?
- Is the approach for emergency planning acceptable?
- Is the Regulatory Technology Development Plan adequate for final design approval?
- Is the proposed application procedure in the licensing plan of HTGR-85-001 acceptable?

The staff believes the responses to these questions would be important to the designer in preparing the design approval application for the MHTGR. The questions are addressed in Section 8.2 of this report.

1.10 Principal Reviewers

This report is based in large part on draft NUREG-1338. The principal individuals who participated in the MHTGR review documented in draft NUREG-1338 are listed in Section 1.9 of that document. The principal reviewer in this report is Jack N. Donohew, NRR Project Manager for the MHTGR Project.

2. REVIEW APPROACH

2.1 Introduction

The staff conducts preapplication reviews of advanced reactor designs so that the staff and the public can learn about the design during the development of the design details, and the designers can learn about the licensability issues of the design before the design details are completed. This review process, which is only done for advanced reactors, originates from the Commission's Advanced Reactor Policy Statement (51 FR 24643) and is discussed in NUREG-1226. This review precedes an application for design approval: preliminary design approval, final design approval, or standard plant design certification in accordance with 10 CFR Part 52.

Advanced reactors are defined as "those reactors that are significantly different from current generation light-water reactors [LWRs] under construction or in operation and to include reactors that provide enhanced margins of safety or utilize simplified inherent or other innovative means to accomplish their safety functions" (NUREG-1226). The MHTGR is an advanced reactor and the current generation of LWRs are the following evolutionary reactors: the General Electric Advanced Boiling Water Reactor (ABWR) design and the Asea Brown Boveri—Combustion Engineering (ABB-CE) System 80+ design. The staff has reviewed these LWRs for standard plant design certification, but the reactors have not been constructed or operated.

The staff has completed one preapplication review of an advanced reactor and issued NUREG-1368, a final preapplication safety evaluation report (PSER) on the sodium-cooled Power Reactor Innovative Small Module (PRISM) reactor design. Because of timing and resource limitations, the review approach taken herein on the MHTGR design differs from that in NUREG-1368.

2.2 Scope of the Review

The preapplication review is a non-licensing review of an advanced reactor design in terms of the requirements and guidance in the Commission's Advanced Reactor Policy Statement. In that policy statement, the Commission stated the following:

- Advanced reactor designs must, as a minimum, have the same degree of protection of the public and environment as is required for the current generation of LWRs.

- Enhanced margins of safety and/or use of inherent passive or innovative systems in the design are expected.

- The supporting technology, operating experience, technology development, and prototype testing for the design must be provided.

- The proposal of less-prescriptive or nonprescriptive design criteria in the regulatory process for the design is encouraged.

In NUREG-1226, the staff stated that the results of the staff preapplication review would be a safety evaluation report that documented the key safety issues associated with the design, gave guidance on the licensing criteria applicable to the design, assessed the adequacy of the applicant-sponsored research and development programs proposed in support of the design, and determined any obvious impediments to licensing the design.

Therefore, the preapplication review is to (1) determine the conformance of the advanced reactor design application to the preceding statements in the Commission's Advanced Reactor Policy Statement and (2) discuss the key safety, research and development, and licensing criteria issues that are potential impediments to licensing the design.

The initial phase of the preapplication review was documented in draft NUREG-1338. The present report documents the completion of the preapplication review by the staff of the MHTGR design.

2.3 Review Approach

In the preapplication review documented in draft NUREG-1338, the staff evaluated the following for the MHTGR design:

- conformance to the Commission's Advanced Reactor Policy Statement.
- safety and policy issues.
- research and development plans.
- proposed new licensing criteria.

The review approach documented in draft NUREG-1338 was, in general, the same approach used by the staff for licensing LWRs: a review of the plant against the Standard Review Plan (SRP) sections in NUREG-0800 (July 1982), the regulations in 10 CFR Parts 50 and 100, regulatory guides, general design criteria (GDCs) (10 CFR Part 50 Appendix A), the designer's probabilistic risk assessment, and staff-endorsed industry codes and standards. The staff evaluated the defense-in-depth factors that contribute to LWR safety to ensure that similar factors were in the MHTGR design. The key examples of this evaluation approach are in Table 1.4 of draft NUREG-1338. The review by the staff documented in draft NUREG-1338 only lacked the final design details in a design approval application to be a review to license the design.

Because of timing and resource limitations, the approach taken to complete the preapplication review was the following:

- take what had been done by the staff in draft NUREG-1338

- review the draft NUREG-1338 taking into account the following new information on the design since draft NUREG-1338 was issued:

  - submittals by the Department of Energy (DOE)
  - contractor reports
  - Commission papers on policy and technical issues,

- consider the staff's conclusions in staff documents on HTGRs (i.e., Fort St. Vrain) and in NUREG-1368 on the preapplication review of the advanced PRISM design

- discuss the following:

  - The most significant safety issues, or licensability issues, that may be impediments to licensing the design.
  - The policy issues for advanced reactors, including licensing criteria, that are applicable to the design.

Recent contractor reports were used to assist the staff in the (1) identification of the MHTGR licensability issues and (2) revision, if any, of the discussions in draft NUREG-1338 on the adequacy of the MHTGR research and development programs.

The conclusions on the conformance of the MHTGR design application to the Commission's Advanced Reactor Policy Statement and on the DOE approach to the MHTGR design using the broad (top-level) licensing criteria discussed in Section 1.5 of this report were based on the staff's evaluation in draft NUREG-1338 and the staff's review of new information since draft NUREG-1338 was issued. The conclusions are given in Section 2.7 below.

The staff has not evaluated all of the new information submitted by DOE since draft NUREG-1338 was issued; however, the new information was reviewed by a contractor or by the staff to determine if it involved a licensability issue and to update the discussion in draft NUREG-1338. Because the preapplication review is not performed to approve any part of the MHTGR, the new information was reviewed at least to the extent that it involved the licensability issues identified in Chapters 3 and discussed in Chapter 4 of this report.

The systems in the MHTGR are discussed in this report to the extent that they identify licensability issues that may be impediments to licensing the MHTGR design. Therefore, this report is a discussion of licensability issues for the design rather than a review of the plant systems and their safety issues, which was performed in draft NUREG-1338.

The licensability issues identify areas that the designer must address at the design approval review stage. Because the preapplication review is performed while the design is still developing and still lacks details that would be required in a design approval application, other licensing issues may appear during the design approval review. The preapplication review is not intended to be complete or the basis for NRC accepting any part of the design. This design will be approved through the separate and complete review of the design in accordance with 10 CFR Part 52.

The staff identified certain key safety issues of a policy nature as well as of a technical nature that required Commission review and guidance for advanced reactors because of the departures from current guidance in NUREG-0800 for licensing LWRs. These issues were identified in policy issue papers submitted to the Commission on the advanced reactor designs, including the MHTGR, and on the advanced and evolutionary LWRs, which contain subjects that also apply to the MHTGR design. These policy issues are discussed in Chapter 5 of this report.

The recent contractor reports which were either completed after draft NUREG-1338 was issued or not discussed in that report are discussed in Chapter 6 of this report.

2.4 Information Needed From The Preapplicant

In NUREG-1226, the staff stated that to perform a meaningful preapplication review the following information on the design was needed:

- description of the plant design and its proposed design, safety, and licensing criteria, including analyses of major accident scenarios demonstrating acceptable plant response
- probabilistic risk assessment
- description of those applicant sponsored research and development programs considered necessary to support development and licensing of the design

DOE submitted to the staff information on the MHTGR in the following major documents:

- HTGR-86-024, "Preliminary Safety Information Document (PSID) for the Standard MHTGR
- DOE-HTGR-86011, "Probabilistic Risk Assessment for the Standard Modular High Temperature Gas-Cooled Reactor"
- DOE-HTGR-87-001, "Emergency Planning Bases for the Standard Modular High Temperature Gas-Cooled Reactor"
- DOE-HTGR-86-064, "Regulatory Technology Development Plan for the Standard Modular High-Temperature Gas-Cooled Reactor
- DOE-HTGR-87089, "MHTGR Assessment of NRC LWR Generic Safety Issues"
- DOE-HTGR-90257, "MHTGR Fuel Process and Quality Control Description"
- DOE-HTGR-86004, "Overall Plant Design Specification Modular High Temperature Gas-Cooled Reactor"
- DOE-HTGR-88311, "Containment Study for MHTGR"
- DOE-HTGR-90321, "450 MW(t) MHTGR Source Term and Containment Study"

On the basis of its assessment of these documents, the staff concludes that DOE submitted sufficient information for a preapplication review of the MHTGR design.

2.5 Licensability Issues

Licensability issues are those issues that may be resolved by fundamentally altering the proposed design. That is, the changes to the design to make the design "acceptable" may fundamentally alter the design from what was originally proposed. Examples of solutions to problems that fundamentally change the design are: changing the reactor coolant, replacing pressure tubes in the core for a reactor vessel, significantly changing the fuel enrichment, and replacing a vented, high-leakage containment with a conventional, essentially leak-tight, pressure-retaining LWR containment.

These issues need to be raised at the preapplication stage. They are the key safety issues and licensing impediments of the design that the designers should understand before the design is submitted for design approval (i.e., preliminary or final design approval, or standard plant design certification, in accordance with 10 CFR Part 52). In many cases, the designer may only need to develop more justification for the proposed design.

The licensability issues for the MHTGR design are discussed in Chapter 4 of this report. Licensability issues may also include aspects of a design that are significant departures from past NRC acceptance practices (i.e., the high leakage containment for the MHTGR). These aspects of the design would be raised as policy issues to the Commission for guidance. Where the significant departure from past acceptance practices for a design has been raised to the Commission and the Commission has accepted the design's departure, the significant departure by the design would not be a licensability issue because the Commission has accepted it. Also, if the staff has approved a significant departure from past practice for a new design and the basis for approval applied to the MHTGR design, this departure would also not be a licensability issue. The policy issues applicable to the MHTGR design are discussed in Chapter 5.

2.6 Technical Areas Needing Review for Licensability Issues

Because the preapplication review of the MHTGR was performed to identify licensability issues and not to approve the design, the staff did not review all the technical areas of the MHTGR design described in the PSID to identify licensability issues that may be impediments to licensing the MHTGR design.

The staff addressed technical areas for the MHTGR design that did not need a preapplication review in Section 1.4 of draft NUREG-1338. The staff stated in that section that a preapplication review was not performed in those areas in which conventional approaches or experiences with early HTGRs have been fully satisfactory. Table 1.3 of draft NUREG-1338 listed the deferred areas for the initial preapplication review and the sections in draft NUREG-1338 in which the areas were discussed. The staff stated in draft NUREG-1338 that it believed these areas were capable of successful resolution at the design approval stage for the following reasons:

- The staff had resolved analogous problems in licensing a similar plant design.

- There were standard, conventional methods to resolve the potential problems.

- The design is not a significant departure from previous NRC acceptance practices, and the methods to resolve the potential design problems would not fundamentally alter the plant design.

There are several significant departures from past NRC acceptance practices that were proposed by DOE for the MHTGR. These, like the high leakage containment, were submitted by the staff to the Commission for its guidance. They are discussed in Section 5.2 of this report on advanced reactor policy issues directly applicable to the MHTGR design.

Draft NUREG-1338, the policy issues in Commission papers, and the Safety Evaluation Reports that licensed Fort St. Vrain (FSERs dated June 21, 1968 and January 20, 1972) were reviewed to determine the technical areas that had licensability issues for the MHTGR design. These technical areas are the following:

- fuel design and performance
- containment performance
- modeling of fission-product transport and related phenomena
- reactor cavity cooling system
- reactor vessel
- safety classification of structures, systems, and components

2.7 DOE Responsiveness to Advanced Reactor Policy Statement

In its Advanced Reactor Policy Statement for preapplication reviews, the Commission stated that advanced reactor designers (1) should submit a design with at least the same level of protection as current-generation LWRs and the technology development needed to complete the design, (2) are expected to provide safety enhancements and/or use of inherent passive or innovative systems in the design, and (3) are encouraged to propose new regulatory approaches that NRC may apply to the design.

The level of protection and the safety enhancements for the MHTGR design were discussed in Section 1.8 and Appendix D of draft NUREG-1338. The staff concluded at that time that the MHTGR can provide a greater level of protection to the public than current-generation LWRs. The specific reactor design attributes and how they have been met by the MHTGR design were discussed in Table D.1 of draft NUREG-1338 and contained the following attributes:

- highly reliable and less-complex shutdown and decay heat removal systems (i.e., the passive and inherent negative temperature coefficient and reactor cavity cooling system)

- longer time constants and sufficient instrumentation to allow time for more diagnosis and management before reaching safety systems challenge and/or exposure of vital equipment to adverse conditions during accidents

- simplified safety systems that, where possible, reduce required operator actions, equipment subjected to severe environmental conditions, and components needed for maintaining safe-shutdown conditions

- minimized potential for severe accidents and their consequences by providing sufficient inherent safety, reliability, redundancy, diversity, and independence in safety systems

- reduced potential radiation exposure to plant personnel

- Incorporated defense-in-depth by maintaining multiple barriers against radiation release and by reducing the potential for the consequences of severe accidents

- design features proved by citation of existing technology or established to a suitable technology development program.

The staff, however, further concluded in draft NUREG-1338 that the design had only the potential for this enhanced level of safety. This was because the staff's final conclusion on safety enhancements of the MHTGR design had to be based on the final plant design and the technology development information presented at the design approval stage.

In completing the preapplication review, the staff concludes that sufficient proof for some of the safety enhancement attributes discussed in Appendix D.1 of draft NUREG-1338 exists and proof was shown in the licensing of Fort St. Vrain that the MHTGR provides at least the same level of protection to the public and the environment as the current generation of LWRs. This is based on the proven large negative temperature coefficient, multicoated fuel particles, reduced potential for occupational radiation exposure, and slow thermal response to accidents. In the discussion of licensability issues in Chapter 4 of this report, the lack of demonstration of the proposed fuel performance and the need to demonstrate the passive ultimate heat sink does not detract from the overall safety enhancements of the MHTGR design.

DOE submitted the Regulatory Technology Development Plan (DOE-HTGR-86-064) to describe the research and development programs, planned and in progress, for the MHTGR. In each topical area of the plan, a summary of the existing database was provided. This plan is discussed in Chapter 7 of this report.

In designing the MHTGR, DOE proposed a top-down approach starting from broad (top-level) requirements which does not use the comparison of the design to the SRP sections, regulations, and GDC for LWRs. This is discussed in Section 1.5 of this report.

In reviewing the top-level criteria proposed by DOE, the staff concluded in Section 1.5 of draft NUREG-1338 that DOE's approach is a systematic and useful approach for designing a nuclear power plant; however, it is not an adequate replacement for NRC's regulatory approach to licensing a nuclear power plant or an adequate assurance of protection of the public health and safety from plant operation. To assure the new designs provide at least an equivalent level of safety as current-generation LWRs, the staff stated that one part of this assurance must be a comparison of the new design to the NRC licensing requirements and guidance, including SRP sections, regulations, regulatory guides, GDCs, and endorsed industry codes and standards. These latter criteria define the safety margins for the new designs and provide assurance that the top-level criteria have been met.

DOE provided its conclusions about the applicability of the GDC to the MHTGR design in its response to Comment G.3-1 of PSID Chapter R. There were many GDC which DOE believed were not applicable to the MHTGR because of DOE's positions on the top-level criteria, containment design and isolation, protection provided by the fuel, and safety classification. Because of time and resource limitations, the staff did not evaluate DOE's positions on each GDC criterion, but discussed the top-level criteria, containment, fuel, and safety classification in Chapters 3 through 8 of this report. An example of how the staff may apply the GDC, including possible new GDC, to the MHTGR is given in Section 3.2, of the final PSER for PRISM, NUREG-1368. The staff concluded in NUREG-1368 that almost all the GDC were applicable, directly or with revisions, to the PRISM design.

In completing the preapplication, the staff concludes that the statements made in Section 1.5 of draft NUREG-1338 remain valid as an evaluation of DOE's approach to designing the MHTGR. Also, in NUREG-1226, the staff requested that designers clearly explain how their design criteria offers the same level of protection as the NRC regulatory approach. DOE should do this for the MHTGR at the design approval review stage.

On the basis of its preapplication review, the staff concludes that DOE has been responsive to the Commission's Advanced Reactor Policy Statement.

3. IDENTIFICATION OF LICENSABILITY ISSUES

3.1 Introduction

This chapter discusses the identification of licensability issues for the Modular High Temperature Gas-Cooled Reactor (MHTGR) using the following documents:

- Final Safety Evaluation Report (FSER) for the licensing of Fort St. Vrain (NRC, 1968 and 1972)
- supplement to Applicant's Decommissioning Environmental Report, Post Operating License Stage, for operational problems at Fort St. Vrain (PSC, 1991 and 1992)
- draft NUREG-1338, draft Preapplication Safety Evaluation Report (PSER) on the MHTGR design
- new information on the MHTGR submitted by the Department of Energy (DOE) since draft NUREG-1338 was issued
- implications of the accident at Chernobyl in NUREG-1251
- Commission papers discussed in Chapter 5 of this report on policy and technical issues for current-generation and advanced reactors
- contractor reports discussed in Chapter 6 of this report which were completed since draft NUREG-1338 was issued

Fort St. Vrain was a high-temperature gas-cooled reactor (HTGR) similar in design to the MHTGR. The FSER which provided the staff's technical basis for licensing the HTGR and the experience of operating the plant provide insights as to possible licensability issues for the MHTGR.

Draft NUREG-1338 documented the staff's preliminary preapplication review of the licensability of the MHTGR design. Because the staff concentrated its review on those features, issues, and research and development activities considered important to the safety and licensability of the design, the key safety issues in draft NUREG-1338 which could significantly alter the design by their resolution would become licensability issues for the design. In discussing draft NUREG-1338 in Section 3.4 below, the statements in draft NUREG-1338 that remain valid for the MHTGR design are also noted.

In the discussions that follow, references will also be made to the information in the Preliminary Safety Information Document (PSID) ([DOE]-HTGR-86-024) for the MHTGR, the final PSER for the PRISM advanced reactor (NUREG-1368), and the letter from the Advisory Committee on Reactor Safeguards (ACRS) on draft NUREG-1338 (October 13, 1988).

3.2 Fort St. Vrain Operational Problems

3.2.1 Operational History

Fort St. Vrain was an HTGR owned and operated by the Public Service Company of Colorado (PSC). The operational history of the plant is discussed in the Decommissioning Environmental Report (PSC, 1991 and 1992).

Construction was authorized by the Atomic Energy Commission on September 17, 1968 and the plant received a full-power operating license on December 21, 1973; however, extensive preoperational testing and resulting engineering modifications delayed commercial operation until 1979. Fort St. Vrain operated at a reduced capacity of 200 MW(e) instead of the original design capacity of 330 MW(e). It operated for about 11 years, from 1979 to 1989, and then was shut down to be decommissioned.

The plant had an inconsistent record of operation with a historical capacity factor of less than 15 percent as a result of such technical problems as:

- core thermal and neutron oscillations
- moisture ingress into the reactor vessel from the helium circulator water bearings
- helium circulator material failures
- multiple control rod drive material failures and failures to automatically scram
- inadequate original design analyses (which limited maximum capacity to 82 percent power)
- cracking of steam generator main steam outlet piping assemblies
- major fire damage to turbine building

The turbine building fire and implementation of the environmental qualification program are not pertinent to a discussion of licensability issues for the MHTGR design because the design would be reviewed against the current standards for fire protection and environmental qualification for design approval. The material problems listed above were also not considered because there was little information available on these operational problems and for the MHTGR design, unless the problems were discussed in draft NUREG-1338.

3.2.2 Evaluation of Operational Problems

Core Thermal and Neutron Oscillations

The operational problem at Fort St. Vrain concerning fuel block oscillations was solved by installing mechanical restraints at the tops of the fuel stacks. This problem was recognized in the MHTGR design by the mechanical design of the upper-plenum elements and by the flow modeling test described in Section 4.4.4 of the PSID ([DOE]-HTGR-86-024). This is also discussed in draft NUREG-1338 Section 4.4.4, Item B.

Although the inner core of the MHTGR comprises reflector blocks, which was not the case for Fort St. Vrain, the MHTGR core structure is similar to that approved for Fort St. Vrain and the operational problem was addressed in the MHTGR design; therefore, these oscillations should not be a licensability issue for the MHTGR design.

Helium Circulator Water Ingress

The circulators for Fort St. Vrain constantly introduced water into the core. The MHTGR circulators are designed with a single-stage, axial-flow compressor with magnetic bearings for the main circulator (at the top of the steam generator vessel) and a two-stage, axial-flow compressor with oil-lubricated bearings for the shutdown cooling system (at the bottom of the reactor vessel), as described in PSID Sections 5.3.2 and 5.4.2, respectively. The magnetic bearings for the main circulator are discussed in Comment 5-27 in Chapter R of the PSID.

The MHTGR circulators will not have water bearings to reduce the potential for the water ingress problems suffered at Fort St. Vrain; however, these circulators will not have the missile barrier that was provided in the Fort St. Vrain circulators to protect safety-related equipment in the prestressed concrete vessel (PCRV). This lack of missile barriers in the MHTGR is involved in the safety classification policy issue for the MHTGR, because it was based on the reactor and steam generator vessels not being considered safety-related systems for the MHTGR. The PCRV for Fort St. Vrain, which housed the reactor and steam generator vessels, however, was approved by the staff as being a safety-related system.

Safety classification is a licensability issue for the MHTGR design because the MHTGR design has few safety-related systems, and changing the safety classification criteria may fundamentally alter the proposed MHTGR design by significantly increasing the number of safety-related systems. This issue is also a policy issue for advanced reactors, and is discussed in Sections 4.2.5 (licensability issue) and 5.2.8 (policy issue) of this report.

Control Rod Drive Failures

The control rods and reserve shutdown system are the reactor control systems for the MHTGR and are described in PSID Sections 4.2.4 and 4.3. As discussed in Section 4.3.5.H of draft NUREG-1338 and in Comment 4-24 in Chapter R of the PSID, DOE identified the similarities and differences between the Fort St. Vrain and MHTGR control rod equipment, including materials employed, and discussed the improvements for the MHTGR based on the experience at Fort St. Vrain. The staff concluded in draft NUREG-1338 that it had confidence in the MHTGR control rod system; therefore, reactor control is not a licensability issue for the MHTGR design.

Inadequate Original Design Analyses

Because there is no information on the specific problems involved, the staff addressed this operational problem by reviewing draft NUREG-1338 to determine licensability issues. This review is discussed in Section 3.3 of this report.

Conclusions

On the basis of its review of the Fort St. Vrain operational problems, the staff concludes that safety classification is the only licensability issue for the MHTGR identified from the operational problems at Fort St. Vrain.

3.3 Differences between MHTGR and Fort St. Vrain Designs

The staff reviewed the Fort St. Vrain FSER (NRC, 1968 and 1972) to determine if there were any significant differences between the Fort St. Vrain design and the MHTGR design that would point to a licensability issue for the MHTGR. The licensability issues determined from this review are discussed below.

Fuel Design and Containment

The fuel proposed for the MHTGR is described in PSID Section 4.2 ([DOE]-HTGR-86-024). It is essentially the same fuel as that approved for Fort St. Vrain and the staff stated in the Fort St. Vrain FSER that the manufacture of the coated fuel particles is a well developed process and the particles were tested, with satisfactory results, to burnups in excess of 20 percent and to temperatures as high as 3600 °F (1980 °C). However, the fuel for Fort St Vrain was approved for about 1-percent failures of coatings in normal operation and about 50 - 100 percent in accidents. The staff reported, in the Fort St. Vrain FSER, that PSC stated that about 5 percent of halogens and 100 percent of noble gases would be released from the fuel during accidents.

The as-manufactured, normal inservice, and accident failed-fuel fractions, discussed in PSID Section 4.2.5.2.2 (Table 4.2-4) for the MHTGR design, are less than or equal to the following: 4.2×10^{-4}, 2.0×10^{-4}, and 6.0×10^{-4}, respectively, at 95-percent confidence. The fraction of fuel particles with missing or defective outer pyrolytic carbon layer is not included in the as-manufactured failed-fuel fraction because the silicon carbide layer is intact and the fission products will be kept within the fuel. These fractions are significantly lower than the fraction approved by the staff for Fort St. Vrain, and have not been sufficiently justified by DOE.

The failed-fuel fraction affects the containment design and the acceptable containment leak rate because the dose consequences from accidents are directly related to the failed-fuel fraction (fraction of fission products released from the fuel to the containment) times the containment leak rate (fraction of containment radioactivity released to the environment). This is discussed by DOE in a report (DOE-HTGR-90321) submitted to NRC by letter dated April 13, 1993.

For Fort St. Vrain, the containment leak rate was 0.2 percent building volumes per day, a value comparable to that for a conventional, leak-tight, pressure-retaining, light-water reactor (LWR) containment. By comparison, the containment for the MHTGR is designed to leak at less than one building volume per day. The staff has not approved such a high leak rate for a containment structure; therefore, the MHTGR design leak rate would significantly depart from previous NRC practices. A failed-fuel fraction like that approved for Fort St. Vrain may require a leak-tight containment for the MHTGR design.

The MHTGR containment, described in PSID Section 6.1, is not a conventional, leak-tight, pressure-retaining structure as was the PCRV. Therefore, the containment and its leak-tightness is a potential licensability issue for the MHTGR design. It is involved in the containment policy issue discussed in Section 5.2.3 of this report; however, because the Commission accepted the possibility of a high-leakage containment, a high-leakage containment in itself is not a licensability issue.

However, if the staff does not accept the proposed failed-fuel fractions for the MHTGR design, the MHTGR containment design would be affected. Such a change may fundamentally change the MHTGR design. Therefore, the low fuel-failure rate and the high containment leakage for the acceptable fuel-failure rate are licensability issues for the MHTGR design. They are discussed in Sections 4.2.1 and 4.2.4 of this report.

Accident Analyses

On the basis of the staff's conclusions on accident analyses in the Fort St. Vrain FSER, the accident analyses for the MHTGR design should not be a licensability issue. The accident selection and evaluation is a policy issue for the MHTGR; however, the criteria approved by the Commission do not appear to conflict with criteria proposed by DOE. This is discussed in Section 5.2.1 of this report.

Although the staff did not consider that a licensability issue existed within the technical area of accident analyses, it had not evaluated the HTGR fission-product transport codes in (1) licensing Fort St. Vrain and (2) draft NUREG-1338 for the MHTGR design. Therefore, to complete the preapplication review, and because the accident dose consequences from containment leakage and surface deposition are based on these codes, the staff had a contractor review these codes. As discussed in Section 6.4.1 of this report, the contractor concluded that these codes had not been verified and validated.

Because changes to the codes could significantly change the calculated dose consequences for accidents and, therefore, the MHTGR design based on these consequences, these computer codes are a licensability issue for the MHTGR design; they are discussed in Section 4.2.2 of this report.

Reactor Cavity Cooling System

For the MHTGR design, the reactor cavity cooling system (RCCS) is the only safety-grade cooling system for the design should there be a complete loss of forced flow or loss of the steam generator. The Fort St. Vrain plant did not have a similar safety system and relied on having at least one of four circulators and one of six steam generators operating.

The RCCS is a completely passive, non-powered system with closed ducts inside the reactor building surrounding the reactor vessel. The system operates on the principle that hot air rises to draw cooler air into the ducts from the outside. This safety system also has no precedent in the nuclear industry, although the PRISM design, which was discussed in NUREG-1368, has a similar passive cooling system. This system is a policy issue for the MHTGR design; it is discussed in Section 5.2.6 of this report.

If the RCCS is not approved, the MHTGR design would have to be significantly changed. Therefore, the RCCS is a licensability issue for the MHTGR design; it is discussed in Section 4.2.6 of this report.

Conclusions

On the basis of its review of the Fort St. Vrain FSER, the staff concludes that fuel design, containment leak-tightness, fission-product transport codes, and the reactor cavity cooling system are licensability issues for the MHTGR. These issues are discussed in Section 4.2 of this report.

3.4 Draft NUREG-1338 Report

The principal areas of review in draft NUREG-1338 were the fuel design, reactor physics, the vessel systems, the passive heat removal system, the support systems, the protection systems, the electrical power systems, the heat removal systems, and the safety analyses. The staff discussed the technical area, the design description and safety objectives, the scope of review, the review and design criteria, the identified research and development program, and the safety issues raised in the review.

The staff reviewed the safety issues discussed in draft NUREG-1338 to determine which ones, if any, were licensability issues for the design.

3.4.1 Safety Issues in Draft NUREG-1338 and DOE Responses

In its letter of July 31, 1991, DOE submitted its PSER Issues Tracking System, which prioritized the issues identified by the staff in draft NUREG-1338. Each issue and the DOE response are listed in this letter and in Appendix C of this report.

The DOE submittals on each issue, since draft NUREG-1338 was issued, were listed in a DOE handout for a meeting of September 29, 1994 (NRC meeting summary dated October 7, 1994) and are reproduced in Appendix D of this report. The staff has not reviewed all of these submittals because of timing and resource limitations; however, because the preapplication review is not done to approve any part of the MHTGR, the new information was reviewed by a contractor or by the staff at least to the extent that it involved the licensability issues. The work by contractors reviewing the MHTGR design since the draft NUREG-1338 was issued is discussed in Chapter 6 of this report.

3.4.2 Policy Issue Changes to Draft NUREG-1338

Some sections in draft NUREG-1338 Chapters 1, 3, 13, and 15 are marked by an asterisk indicating the section "is particularly sensitive to change by evaluation of forthcoming DOE information." This refers to the statement in the preface and in Section 1.7.3 of draft NUREG-1338 that the staff's positions and conclusions discussed in certain sections of this draft PSER are subject to change because of the staff proposals on the following policy issues discussed in draft NUREG-1338:

- accident selection
- mechanistic source term
- non-conventional containment
- reduced emergency planning

These issues are discussed in Sections 1.7, 3.2.2.2, 3.2.2.3, and 3.2.2.4 of draft NUREG-1338. The proposed review criteria for these issues were submitted to the Commission in SECY-88-203. This paper was withdrawn from the Commission and new criteria were submitted in SECY-93-092. The new criteria are discussed in Section 5.2 of this report.

In completing the preapplication review for the MHTGR, the staff reviewed draft NUREG-1338 to see what changes should be made to the document after the Commission acted on policy issues that are applicable to the MHTGR. These policy issues are discussed in Chapter 5 of this report; the Commission guidance on accident selection, containment performance, and emergency preparedness in SECY-93-092 (Sections 5.2.1, 5.2.3, and 5.2.4 of this report) is significantly different from what the staff proposed in SECY-88-203. Therefore, the staff's statements in draft NUREG-1338 associated with SECY-88-203 on accident selection, containment performance, and emergency preparedness are no longer valid.

Based on the staff's review of draft NUREG-1338, the following sections in the document that are marked with an asterisk are no longer valid because the review criteria discussed in the sections have changed:

- Section 1.7.1 EC-IV events are now not a separate event category (See the discussion on "Accident Analyses" below).
- Section 1.7.3 The criteria discussed in draft NUREG-1338 Section 3.2.2.3 are no longer valid.
- Section 1.7.4 The criteria discussed in draft NUREG-1338 Section 3.2.2.4 are no longer valid.
- Section 3.2.2.3 The criteria discussed in draft NUREG-1338 Section 3.2.2.3 are no longer valid.
- Section 3.2.2.4 The criteria discussed in draft NUREG-1338 Section 3.2.2.4 are no longer valid.

- Section 13.1.6 The criteria discussed in draft NUREG-1338 Section 3.2.2.4 are no longer valid.

The new review criteria for Sections 3.2.2.3 and 3.2.2.4, on containment performance and emergency preparedness, are discussed in Sections 5.2.3 and 5.2.4, respectively, of this report. The EC-IV events, as a separate event category for accident selection and mechanistic source term, were not included in the new criteria for these two issues as discussed in Sections 5.2.1 and 5.2.2, respectively, of this report. The other sections in draft NUREG-1338 marked with an asterisk remain valid.

3.4.3 Licensability Issues from Draft NUREG-1338

The sections that follow discuss the safety issues covered in draft NUREG-1338 and the conclusions of the staff in the document that are pertinent to the licensability of the MHTGR design. The intent of these sections is to determine the licensability issues for the MHTGR, discuss what the staff said about the licensability issues in draft NUREG-1338, explain new information submitted since draft NUREG-1338 was issued, and revise (or delete) statements in draft NUREG-1338 that the staff no longer considers valid. The new information comprised the policy issues discussed in Chapter 5 of this report and additional information submitted by DOE since draft NUREG-1338 was issued.

The staff's statements for the following technical areas, and the appropriate chapter, remain valid for the MHTGR preapplication review:

• Instrumentation and Control Systems	Chapter 7
• Electrical Systems	Chapter 8
• Service Systems (including fire protection)	Chapter 9
• Steam and Power Conversion	Chapter 10
• Operational Radionuclide Control	Chapter 11
• Occupational Radiation Protection	Chapter 12
• Prototype Testing	Chapter 14
• Quality Assurance	Chapter 17

No licensability issues for the MHTGR were identified in these areas. The remaining technical areas discussed in draft NUREG-1338 did have licensability issues and are discussed below:

3.4.3.1 Safety Classification and Design of Structures

In Sections 3.3, 3.4, and 3.5 of draft NUREG-1338, the staff discussed the safety classification of structures, systems, and components, the design of plant structures, and the plant seismic design. Safety classification is discussed throughout draft NUREG-1338 when the staff noted that it did not agree with the safety classification criteria being used by DOE and the few structures, systems, and components (SSCs) being classified as safety related. DOE stated that MHTGR safety-related SSCs were only those needed to limit the accident dose consequences to less than the guidelines in 10 CFR Part 100.

Because the safety classification criteria for the MHTGR are significantly different from criteria used by NRC for LWRs, including the evolutionary and passive advanced LWRs, these criteria are a policy issue for the MHTGR. The issue is addressed in Section 5.2.8 of this report and the Commission decided that the staff should continue its review of the DOE-proposed criteria.

Since draft NUREG-1338 was issued, DOE submitted additional information on safety classification in its responses to staff comments (G-28 and G-31 in Chapter R of the PSID). In addition, safety classification was discussed in a meeting with DOE on January 22, 1992 (NRC meeting summary dated April 15, 1992).

The staff will complete its review of this issue in the design approval review stage (i.e., preliminary design approval, final design approval, or design certification under 10 CFR Part 52). If the DOE-proposed criteria are not approved, the number of safety-related SSCs may change and significantly alter the MHTGR design. Therefore, safety classification is a licensability issue, and it is discussed in Section 4.2.5 of this report.

The staff's statements in Section 3.4 through 3.5 of draft NUREG-1338 remain valid until the licensability issue in Section 4.2.5 of this report is resolved. The staff's statements in Section 3.3 of draft NUREG-1338 are not valid because the Commission's decision for safety classification in Section 5.2.8 is different from what the staff stated in Section 3.3.

3.4.3.2 Reactor

In Chapter 4 of draft NUREG-1338, the staff discussed the fuel design, core structure, core thermal and hydraulic design, core reactor controls, and reactor internals.

Since draft NUREG-1338 was issued, DOE submitted additional information for the fuel design in draft NUREG-1338 Section 4.2. (These are DOE letters dated July 9 and 16, 1991; October 2 and December 9, 1991; and June 24 and 25, 1992.) The staff also held two meetings with DOE, on October 23 and December 17-20, 1991, on fuel design and fission-product transport (NRC meeting summaries dated March 10 and April 10, 1992). The changes that could be made to draft NUREG-1338 Section 4.2 because of the additional information submitted by DOE are the subject of a contractor report discussed in Section 6.4.2 of this report.

The staff's statements on the fuel design in draft NUREG-1338 Section 4.2 remain valid, including the following conclusions in Section 4.2.6:

- The staff believes that the fuel design and quality can be developed to meet the performance objectives proposed by DOE and required by the safety analyses, but notes that this conclusion is dependent on the successful outcome of a research program.

- The staff notes that actual fuel performance ... in the FRG [Federal Republic of Germany] reactors [discussed in Section 1.7 of this report], together with reported laboratory and in-pile tests, gives promise that the performance objectives can eventually be demonstrated.

The staff, however, is unsure when DOE may be able to demonstrate the fuel performance proposed for the MHTGR. Also, DOE has not explained the relationship among the fuel design, fuel manufacture, fuel test performance, and fuel operation in the core during normal and accident conditions in terms of showing that the dose consequences from accidents are acceptable. This is why the staff made the following statements in draft NUREG-1338 Section 4.2.5, Items A, B, C, and D:

- A discussion should be developed in a revised or subsequent document to show that the reference fuel will be...appropriate for the MHTGR.

- It needs to be clearly demonstrated that, when the parameters of the older fuel are used in the updated MHTGR fuel-failure model, the predictions are still applicable.

- The means for achieving 95- and 50-percent confidence levels need to be confirmed, and the associated Weibull probability distribution should be validated.

- Statistical quality control and assurance plans for fuel manufacture, including acceptance criteria, need to be considered in the RTDP [Regulatory Technology Development Plan (DOE-HTGR-86-064)] so that there will be assurance the actual reference fuel is of the specified quality and will perform as predicted.

- In effect, this would mean that fuel batches found acceptable by the quality control program would contain a recognized fraction of weak particles that are accounted for in the safety analysis.

The statements about "weak particles," including the one above, in draft NUREG-1338 Section 4.2.5.D remain valid. The "weak tails" in the strength of the silicon carbide layer in the fuel particle, discussed in Section 6.3.6 of this report on contractor reports, is an example of the "weak fuel."

As discussed in Section 3.3 (above), not approving the fuel design would significantly alter the MHTGR containment design and, therefore, fuel design is a licensability issue. It is discussed in Section 4.2.1 of this report.

For the discussion on nuclear design, thermal and fluid-flow design, and reactor internals in draft NUREG-1338 Sections 4.3 through 4.5, the important issues are the following:

- the concern about the design not achieving cold shutdown with the outer control rods and manual actuation of the reserve shutdown control equipment (RSCE) needed for cold shutdown (Section 4.3.5, Item C)

- the concern about the effects of neutron fluence on the reactor vessel (Section 4.3.5, Item E)

- the concern about the core structural graphite (Sections 4.4.5, 4.5.4, and 4.5.5, Item C)

The staff no longer considers that the automatic shutdown claimed by DOE requires the RSCE, a safety-grade backup shutdown system needed to reach "cold" shutdown for the MHTGR with the safety-grade control rods, to be free of any manual actuation. In Section 4.5, "Active Reactivity Control and Shutdown System," of NUREG-1368, the staff concluded for the ultimate shutdown system (USS), which serves the same function in PRISM as the RCCS in the MHTGR, that manual actuation is acceptable.

The concern about effects of the MHTGR neutron fluence on embrittlement of the reactor vessel is a licensability issue because not approving a steel reactor vessel for the RCCS would significantly alter the MHTGR design. This is true even if the fluence on the vessel may be sufficiently reduced by design changes that do not significantly affect the design of the plant (e.g., additional shielding or a larger reflector in the core). The potential impact on the design to address this issue is not known at this time. This is discussed in Section 4.2.7 of this report.

The MHTGR reactor internals consist of an arrangement of metallic and graphite structures that support and locate the graphite core fuel blocks and reflectors within the reactor vessel, and protect the reactor vessel from high-temperature helium and excessive neutron fluence. The safety design objectives for the reactor internals are to provide for normal and abnormal thermal loadings; thermal expansions and stresses; mechanical, fluid, and seismic loadings; and resistance to corrosion impurities in the helium coolant.

The staff believes, as stated in Section 4.5.6 of draft NUREG-1338, that the most important research and development needs for the structural graphite have been identified, but the research programs described by DOE (DOE-HTGR-86-064) may not be sufficiently comprehensive to meet these needs; also, the staff has not reviewed the proposed American Society of Mechanical Engineers (ASME) codes for nuclear grades of graphite. However, because Fort St. Vrain operated without significant problems with structural graphite, this technical area is not considered a licensability issue for the MHTGR.

Although DOE also submitted its recent reactor graphite development plan (DOE-HTGR-90358) in its letter dated July 16, 1993, the staff did not review it because this area is not considered a licensability issue and because of timing and resource limitations. The staff will review the graphite technology program in the design approval review.

3.4.3.3 Vessels and Heat Removal Systems

In Chapter 5 of draft NUREG-1338, the staff discussed the three vessels of the reactor coolant pressure boundary (i.e., the reactor, steam generator, and crossduct vessels) and the two forced-convection heat removal systems (i.e., the heat transport system for power operation and shutdown cooling system for decay heat removal without the steam generator).

In draft NUREG-1338 Sections 5.2.1, 5.2.5.D and 5.2.5.G, the staff stated that DOE made an application to the ASME Boiler and Pressure Vessel Code (ASME Code) Committee, Section III, Division 1, to extend the maximum allowable service temperature for the reactor vessel. This has been approved by the ASME Code Committee (DOE-HTGR-90286 and DOE letter dated November 7, 1991) and is a licensability issue for the MHTGR design because it is needed for the design and has not yet been approved by NRC.

The staff will review the ASME Code extension in the design approval review. If it is not approved by the staff, the MHTGR design will be fundamentally altered because the RCCS design may have to change to include active systems. Therefore, this ASME Code extension is a licensability issue and is discussed in Section 4.2.8 of this report.

The staff also stated, in draft NUREG-1338 Section 5.2.1, that DOE had applied to the ASME Code Committee to confirm DOE's approach to design the crossduct as a vessel meeting ASME Code, Section III. DOE, however, did not make such an application. The approach of considering the crossduct as a vessel, instead of a pipe, is discussed in a recent contractor report (discussed in Section 6.3.1 of this report). This section also discusses the probability of a gross vessel failure, which is addressed in draft NUREG-1338 Section 5.2.5.A.

The discussion of neutron irradiation of the reactor vessel in draft NUREG-1338 Section 5.2.5.B was addressed in the previous section and is discussed in Section 4.2.7 of this report, because it is a licensability issue for the MHTGR design.

The remaining statements made by the staff in Chapter 5 of draft NUREG-1338 remain valid.

3.4.3.4 Plant Arrangements and Containment

In Chapter 6 of draft NUREG-1338, the staff discussed the arrangement of the plant site; the reactor building which houses the reactor, steam generator, and crossduct vessels, and the containment structure; and the other buildings onsite.

The control room is located inside the Operations Center at the interfaces between the Nuclear Island, the Energy Conversion Area, and the non-protected portion of the plant site. In draft NUREG-1338 Sections 6.1.2 and 13.3.2.2 (Vital Areas), the staff stated that the control room should be in the Nuclear Island protected area because it is a vital area needing security for the reactor operators and the control room equipment. Since draft NUREG-1338 was issued, DOE submitted additional information on the location of the control room in R 13-17 of the PSID and has stated that the control room is within a protected area inside the operations center. Although the control room remains outside the Nuclear Island, it is in a protected area, and the remote shutdown area and the plant protection and instrumentation cabinets are located within the Nuclear Island of the protected area.

The new information appears to resolve the staff's concern in draft NUREG-1338 about the location of the control room; however, the details of having the control room protected area outside the Nuclear Island will be reviewed at the design approval review. However, the location of the control room is not a licensability issue.

The MHTGR containment design in Sections 6.2 and 6.3 of draft NUREG-1338 is not a conventional LWR containment, which is an essentially leaktight, pressure-retaining structure. Rather, it is a controlled and vented containment in which a large primary-coolant release from the reactor vessel would open blowout panels between the reactor and steam generator cavities of the reactor building and be released to the environment through louvers which open for high internal pressure and then close. A rapid depressurization of the reactor coolant pressure boundary would be released from the reactor building as a puff release to the environment.

The unconventional, high-leakage MHTGR containment was a policy issue. It is discussed in Section 5.2.3 of this report. Because the Commission accepted the possibility of a high-leakage containment, this is not a licensability issue; however, because the staff has not completed its review of the release of radioactivity from the containment during accidents, this release is a licensability issue. The MHTGR model for releases is significantly different from the model approved by the staff for Fort St. Vrain and described in the Fort St. Vrain FSER. Changing the release model may result in fundamentally changing the design of the containment. This is discussed under source term in Section 4.2.3 of this report.

Except for the preceding discussion, the staff's statements in Chapter 6 of draft NUREG-1338 remain valid.

3.4.3.5 Conduct of Operations

In Chapter 13 of draft NUREG-1338, the staff discussed emergency preparedness, the role of the control room operators, and safeguards and security. Emergency preparedness and the role of the operators are policy issues that are discussed in Section 5.2 of this report. Except for the statements about emergency preparedness review criteria in Section 13.1.6, as explained in Section 3.3, the staff's statements in Chapter 13 of draft NUREG-1338 remain valid.

For the role of control room operators, the staff stated in draft NUREG-1338 that DOE proposed that the MHTGR operators will have a role different from the role of operators of current LWR plants because the former will not need to perform any safety-related function. However, as discussed in Section 6.4.3 of this report, actions of an operator to start up the shutdown cooling system in the aftermath of certain events can lead to higher fuel temperatures in the core. In addressing the questions raised by the staff in Section 13.2 of draft NUREG-1338, DOE should address this and any other operator action that could lead to similar effects. However, changes to the role of the operator should not fundamentally alter the MHTGR design; therefore, this is not a licensability issue.

In response to the staff's concern in draft NUREG-1338 Sections 13.3 and 6.1.3 about the control room being outside the protected area, DOE stated that although the operators are not considered "vital," they will be afforded a secure operating environment insofar as, in addition to the Nuclear Island of the plant, the control room and portions of the Energy Conversion Area will be within the protected area. These are DOE responses to staff comments (R) 13-16, 13-17 and G-30 in PSID Chapter R ([DOE]-HTGR-86-024). Because this appears to address the staff's concern about the control room, this concern is not a licensability issue for the MHTGR; however, the final staff review for acceptability will be done in the design approval review.

3.4.3.6 Accident Analyses

In Chapter 15 of draft NUREG-1338, the staff discussed the accidents considered by DOE, the probabilistic risk assessment (Appendices A and B of draft NUREG-1338), the independent analyses by NRC contractors, and the accident source terms. The information submitted by DOE is in Chapter 15 and Chapter R of the PSID, the emergency planning bases report (DOE-HTGR-87-001), and the probabilistic risk assessment (DOE-HTGR-86011).

In draft NUREG-1338 Section 15.1.3, the staff stated that the staff and its consultants did not review the individual computer code modeling assumptions and input data, including the fission-product transport computer codes used to calculate the dose consequences of accidents. The staff concluded that its contractors performed independent analyses; however, the staff now believes that the computer codes used for draft NUREG-1338 may not have been independent of the codes used on the MHTGR. Therefore, the staff had a contractor review the MHTGR fission-product transport codes; the contractor's work is discussed in Section 6.4.1 of this report. Because the codes have not been verified and validated, the dose consequences calculated by the codes could change and the MHTGR design based on the dose consequences could significantly change. Therefore, these codes are a licensability issue and are discussed in Section 4.2.2 of this report.

It should be noted that draft NUREG-1338 was written in terms of (accident) event categories EC-I through EC-IV, instead of the event categories EC-I through EC-III presented in NUREG-1368. The difference between event categories EC-I through EC-III currently used by the staff and the event categories EC-I through EC-IV in draft NUREG-1338 are discussed in NUREG-1368 Section 3.1.2.1. The staff is still using the bounding event selection it used in draft NUREG-1338 Section 15.2.3.3 and these bounding events are also discussed in NUREG-1368 Section 3.1.2.1. The dose consequence guidelines for EC-I through EC-III are the same for the event categories in draft NUREG-1338 and the current staff event categories in NUREG-1368.

In the discussion of residual risks in draft NUREG-1338 Section 15.2.4, the staff stated that it and its consultants could not identify credible events that would exceed the lower dose limits of the Protective Action Guides (PAGs) (EPA-520/1-75-001). In draft NUREG-1338, the staff did not consider an event with high core temperatures, where fuel has failed, followed by a rapid depressurization of the reactor vessel. On the basis of the source term, discussed in draft NUREG-1338 Sections 3.2.2.2 and 15.5, fuel failures result from core temperatures exceeding 1600 °C (2900 °F). If the rapid depressurization occurred after the fuel had failed and fission products had migrated into the coolant, the radioactivity released from the fuel would be carried out of the containment in a puff release. This would be significantly higher than the dose consequences discussed in draft NUREG-1338.

In Comment 15-19 of PSID Chapter R, DOE investigated a pressurized conduction cooldown discussed in PSID Section 15.2, which had no release and assumed the pressure relief valve on the steam generator vessel failed at 120 hours after the peak fuel temperature is reached. The calculated thyroid dose consequences are within the consequences calculated for other accidents reported in the PSID; however, in Comment 15-17 for a moisture ingress event with a delayed depressurization, the thyroid dose consequences at 100 hours after the event were a factor of three higher than that reported in the PSID, but below the DOE proposed acceptance level (the lower PAGs of the Environmental Protection Agency). The staff does not agree that these two accidents bound all accidents that could involve a rapid depressurization of the reactor coolant pressure boundary at high fuel temperatures and, for the second accident, the worst meteorology was assumed not to occur during the depressurization event.

The discussion in draft NUREG-1338 Section 15.2.6.2 on graphite fires in the MHTGR core and on the low likelihood of a charcoal fire remains valid. The existence of the large negative Doppler coefficient and lack of a positive, and large, coolant void coefficient for the MHTGR would prevent the MHTGR core from experiencing the large positive reactivity insertion which occurred at Chernobyl and led to prompt criticality and a fire in the Chernobyl core. The non-flammable, gaseous helium coolant in the MHTGR core would prevent oxidation of the graphite, and the coolant would need to be displaced by significant quantities of air from the reactor building on a continuous basis to start and maintain a fire in the core. A graphite fire in the MHTGR would require at least two breaks in the MHTGR reactor pressure vessel to create a chimney effect and, before the fire, a positive force from outside the pressure vessel to displace the helium by air (NUREG/CR-4981). There was also oxidation of the large amounts of zirconium in the Chernobyl core that added energy to the core from hydrogen generation and then combustion, which would not occur in the MHTGR core.

The staff concluded in draft NUREG-1338 that a graphite fire in the MHTGR core is a very low probability event. As stated in NUREG/CR-6218 on air ingression during severe accidents, without two breaches of the reactor vessel to create a chimney effect, it is not likely that significant amounts of air will enter into the core.

The potential for graphite fires in the Fort St. Vrain core was also discussed in NUREG-1251 and was considered extremely improbable although the staff stated in NUREG-1251 that in the event of a fire at Fort St. Vrain the reactor building could be flooded to a level which would defeat the chimney effect for the core and stop the fire. The staff concluded in draft NUREG-1338 Section 15.2.6.2 that this conclusion about the extremely low probability of a charcoal fire at Fort St. Vrain could also be applied to the MHTGR design and that it should not be necessary to envision flooding the MHTGR reactor cavity.

Therefore, graphite fires are not a licensability issue for the MHTGR.

Although not explained in draft NUREG-1338, the accident dose consequences are based on a maximum containment leak rate of one building volume per day and on deposition of radioactivity on surfaces within the reactor building. This is discussed in DOE's responses to the staff's Comments 15-12, 15-16, and 15-18 in PSID Chapter R. The deposition of radioactivity inside containment during accidents was not considered in draft NUREG-1338 and it has not been evaluated in this report. This is part of the source term for the MHTGR. It is a policy issue discussed in Section 5.2.2 of this report, and the Commission decided that a mechanistic source specific to the design was acceptable. However, because the staff did not review this in its evaluations for draft NUREG-1338, the changes to the source term proposed by DOE could result in fundamental changes to the MHTGR design. Therefore, the source term is a licensability issue and it is discussed in Section 4.2.3 of this report.

The changes to draft NUREG-1338 Section 15.4, independent safety analyses for the MHTGR design, because of the additional information submitted by DOE is the subject of a contractor report; it is discussed in Section 6.4.3 of this report.

Except for what is discussed above, the staff's statements in Chapter 15 of draft NUREG-1338 remain valid.

3.4.4 Comparison to Design Certification Reviews

The staff has completed the FSERs (NUREG-1503 and NUREG-1462) for standard plant design certification reviews of the following two evolutionary LWRs: General Electric Advanced Boiling Water Reactor (ABWR) and ABB-CE System 80+. In these FSERs, the staff discussed the following technical areas which were either not discussed or only briefly discussed in draft NUREG-1338 for the MHTGR:

• Seismic Design	Section 3.7
• Radioactive Waste Management	Chapter 11
• Radiation Protection	Chapter 12
• Training	Section 13.2
• Operational Review	Section 13.4
• Plant Procedures	Section 13.5
• Physical Security	Section 13.6
• Human Factors Engineering	Chapter 18
• Generic Issues	Chapter 20

The review methods and criteria discussed in these FSERs would be used by the staff in a design approval review of the MHTGR. DOE should take into consideration the review criteria and staff conclusions in developing its design approval application.

3.4.5 Conclusions

Except as noted in Section 3.4.2 above, essentially all the staff's statements in draft NUREG-1338 remain valid for the MHTGR design. The fuel design and performance is the key licensability issue for the MHTGR design and is discussed in detail, with its effect on accident source term and dose consequences, containment design, and emergency preparedness, in Section 4.2.1 of this report. The other licensability issues identified are safety classification, neutron embrittlement of the reactor vessel, the ASME Code extension for high-temperature reactor vessel service, and MHTGR source term.

Accident selection, mechanistic source term, unconventional containment, reduced emergency planning, safety classification, and role of the operator are policy issues for the MHTGR design and are discussed in Section 5.2 of this report.

3.5 ACRS Letter on Draft NUREG-1338

Before draft NUREG-1338 was issued, the Advisory Committee on Reactor Safeguards (ACRS) sent a letter in 1988 to the Commission chairman (October 13, 1988) on its review of the MHTGR design. The letter is in Appendix C of draft NUREG-1338. The safety issues discussed by the ACRS in its letter, and the sections in this report or draft NUREG-1338 where the safety issue is discussed, are the following:

- The fuel particles must have the retention capabilities attributed to them in the PSID ([DOE]-HTGR-86-024).

  This is discussed in Sections 3.2.3.2 (Fuel Design), 4.2.1, and 5.2.2 of this report.

- The reactivity and temperature-reactivity characteristics used in the safety analyses need further verification.

  This is discussed in Section 4.3.5, Items A and B, of draft NUREG-1338.

- Inadvertent ingress of water or steam into the core must be precluded with high reliability.

  DOE has addressed this by designing the MHTGR circulators to have magnetic bearings to avoid the water ingress problems experienced in Fort St. Vrain, and by discussing specific accidents in PSID Chapter 15 involving water and steam ingress into the core.

- There must be assurance that decay and low-power heat transfer can be accomplished without excessively high core temperatures. Performance of the passive RCCS and the ability to conduct heat to the surrounding earth must be demonstrated.

  The reactor vessel elevated temperature service is discussed in Section 4.2.8 of this report.

- The properties of the structural graphite in the core must be demonstrated and assured.

  This is discussed in Section 3.3.2.2 of this report.

- The important safety benefits of the MHTGR design depend on the core geometry remaining unperturbed and questions about this remain concerning seismic resistance, aging, and cascading effects of accidents.

  This will be addressed at the design approval review stage.

- A major issue is whether a conventional containment structure, or other mitigation system or process, should be required.

  This is addressed in Sections 4.2.4 and 5.2.3 of this report.

- A substantial program of research and development must be continued to support the final design for the MHTGR, including the need to demonstrate the required fuel performance.

  This is discussed in Chapter 7 of this report.

- The staff should develop general guidance for designers of advanced reactors on designing against sabotage.

  Sabotage is discussed in Section 13.3 of draft NUREG-1338.

- Little is said about requirements for operation and staffing in that the advanced reactor designer's claims for needing only a small staff and for the design being less vulnerable to operator error have not been demonstrated.

  Staffing is discussed in Section 5.2.5 of this report.

The ACRS also recommended the following:

- A hot critical experiment may be necessary because the core is of an unusual geometry and has nuclear characteristics different from those in previous HTGRs.
- More extensive analysis is needed of plant response to accidents that might change the core geometry.
- A prototype should be built and tested before design certification.

DOE needs to address the concerns and recommendations of the ACRS for the MHTGR design at the design approval review stage.

The staff concludes that the ACRS letter discussed above does not identify any licensability issues for the MHTGR design that have not been identified in Sections 3.2 through 3.4.

3.6 Implications of the Chernobyl Accident

The implications of the Chernobyl accident are being discussed because of the public's concern about the Chernobyl accident and its significance to nuclear power plant designs in the United States which have similar features (i.e., the graphite-moderated Fort St. Vrain and MHTGR), and because of the identification of a severe-accident issue for DOE to address for the MHTGR design at the design approval review stage. The question is whether the implications of the Chernobyl accident raise a licensability issue for the MHTGR.

The staff discussed the implications of this accident on nuclear power plants in the United States in NUREG-1251 and NUREG-0933. The Chernobyl reactor was a graphite-moderated, light-water-cooled thermal reactor in which the graphite fire in the core was caused by a prompt criticality event. The Fort St. Vrain HTGR plant and the MHTGR design are also graphite-moderated reactors; however, the coolant for both HTGRs is helium and not light water, and helium causes significantly lower reactivity effects than light water.

In NUREG-0933, the staff stated that it had assessed the HTGR concept against the following issues raised by the Chernobyl accident: operations, design, containment, emergency planning, and severe-accident phenomena. The staff concluded that the only features that Fort St. Vrain and the MHTGR had in common with the Chernobyl design were the use of a graphite moderator and gravity-driven control rods. Important differences between the HTGRs and the Chernobyl design were (1) the slow response of HTGRs to plant transients, (2) the difference in fuel and the increased margin to fuel failure of HTGRs, and (3) the helium coolant for the HTGRs.

Further in NUREG-0933, the staff reported its assessment of the areas of operation, design, containment, emergency planning, and severe-accident phenomena, and found that the implications of the Chernobyl accident have not raised any new licensing concerns for HTGRs. The Chernobyl accident did reinforce the staff's concern about the integrity of graphite support structures in HTGR cores. Even before the Chernobyl accident, the staff had considered a limited probabilistic risk assessment and further experiments with structural graphite for Fort St. Vrain. While the Chernobyl accident supported the need for such work, the then-imminent shutdown of Fort St. Vrain in 1989 removed this need for Fort St. Vrain. Structural graphite integrity is discussed in Section 3.3.2.2 of this report for the MHTGR and is not a licensability issue for the MHTGR design.

In considering the potential for graphite fires in the MHTGR core, discussed in draft NUREG-1338 Section 15.2.6.2 and Section 3.3.2.6 (above), the staff has concluded that it is of very low probability. However, the Commission requested, in its discussion on the advanced reactors policy issue of containment performance, that the staff address the potential loss of the reactor coolant pressure boundary (RCPB) which could results in air ingress into the core from the chimney effect, a graphite fire in the core, failure of the fuel particles, and release of radioactivity from containment to the environment. This is discussed in Section 5.2.2 of this report.

The MHTGR design has a steam generator in the RCPB with the water and steam side at a significantly higher pressure than the helium coolant side. With a steam generator tube rupture, the helium coolant would be displaced from the core and replaced by light water. Also, the rapid ejection of a single control rod from the MHTGR core could cause the reactor to go prompt critical, as stated in draft NUREG-1338 Section 4.3.5.H, although the staff also stated in this section that a satisfactory level of mechanical performance can be achieved in the MHTGR design so that this rod ejection event would be precluded.

DOE considered water ingress into the MHTGR core in the PSID in Safety-Related Design Conditions (SRDCs) 6 through 9, which are discussed in PSID Sections 15.13.6 through 15.13.9, respectively ([DOE]-HTGR-86-024). A reactor trip would occur with safety-related equipment on high core power-to-flow ratio as a result of the insertion of water into the core, and the steam generator would be isolated on high RCPB pressure from the incoming water. The moisture monitors within the RCPB are not safety related and no credit was taken for them by DOE in the analysis of the event. This isolation limits the amount of water in the core by preventing further water from entering the steam generator and, thus, the RCPB. The calculated amounts of water introduced to the core were very small amounts between 1090 and 2200 kgm (2400 and 4850 lb) and would become steam within the hot helium coolant. Therefore, the Chernobyl situation of a light-water-cooled, graphite-moderated core is not approximated by the MHTGR core in SRDCs 6 through 9 events.

DOE considered the fission-product radioactivity released from the core because of the effect of water on the fuel in the PSID, but it did not consider the possibility of a prompt criticality event after the steam generator was isolated and the potential for a resulting fire in the core. The scenario would require at least two separate events because the reactivity effect of the water from the steam generator tube rupture and the increase in RCPB pressure from the water would actuate the two safety-grade reactor protection systems, both of which would shut down the core. The control rod ejection would be a separate event. However, to address the Commission's concern in its staff requirements memorandum (SRM) on the advanced reactors policy issue of containment performance, DOE should address, at the design approval review stage, this severe-accident event and the potential for a resulting fire in the MHTGR core with a significant release of radioactivity to the environment.

As discussed above, no licensability issues were identified for the MHTGR from the implications of the accident at Chernobyl.

3.7 Commission Policy Issues

There are many policy issues for which the staff has requested guidance from the Commission on applying review criteria to the advanced reactors and the evolutionary and advanced LWR designs. The issues that are applicable to the MHTGR design are discussed in Chapter 5 of this report.

The issues submitted to the Commission specifically on the MHTGR are discussed in Section 5.2 of this report. These issues were reviewed to determine if any had the potential, by their resolution, to fundamentally alter the MHTGR design. Of these issues, safety classification is the only one considered a licensability issue for the MHTGR design. The Commission did not accept the safety classification criteria proposed by DOE for the MHTGR and resolving these criteria may fundamentally alter the design. Safety classification is also identified as a licensability issue in Section 3.4.2.1 of this report, and it is discussed in Section 4.2.5 (licensability issue) and 5.2.8 (policy issue) of this report.

The issues submitted to the Commission on the evolutionary and advanced LWRs that are applicable to the MHTGR are discussed in Section 5.3 of this report.

3.8 Contractor Reports

The staff has engaged technical assistance on the MHTGR design since the draft PSER on the MHTGR design, NUREG-1338, was issued in March 1989. The reports which contractors completed since March 1989 are discussed in Chapter 6 of this report. The staff reviewed these contractor reports to determine if the reports included discussions and conclusions which

- identified a licensability issue for the MHTGR
- supported a licensability issue identified by the staff
- contradicted a licensability issue identified by the staff

In the review of these contractor reports, the staff identified a licensability issue on the computer codes used to calculate the fission product transport from the fuel through the containment to the environment. This is discussed in Section 4.2.2 of this report.

3.9 Conclusions

In this chapter, the staff identified the licensability issues for the MHTGR design which are discussed in Chapter 4 of this report. The licensability issues may involve the policy issues applicable to the MHTGR which are discussed in Chapter 5 of this report. The policy issues were specific questions addressed to the Commission on the evolutionary and advanced reactors and the Commission's responses to the questions provide guidance to the MHTGR designers.

4. LICENSABILITY ISSUES

4.1 Introduction

The staff performs preapplication reviews of an advanced reactor design in part to identify issues that may impede licensing the design. These licensability issues are where the design departs significantly from what NRC has accepted in the past or where changes to the design to resolve a staff concern may fundamentally alter the proposed design. These issues need to be identified at an early stage, so that the designers can address the issues in an application to NRC for design approval: the preliminary design approval (PDA), final design approval (FDA), or standard plan design certification under 10 CFR Part 52.

In this chapter, the staff discusses the licensability issues for the MHTGR design. The identification of these issues was discussed in the previous chapter (Chapter 3). The references in this chapter to the evolutionary light-water reactors (LWRs) and passive advanced LWRs are references to the plants listed in Section 5.1 of this report which have gone through or are going through design approval reviews by the staff.

4.2 MHTGR Licensability Issues

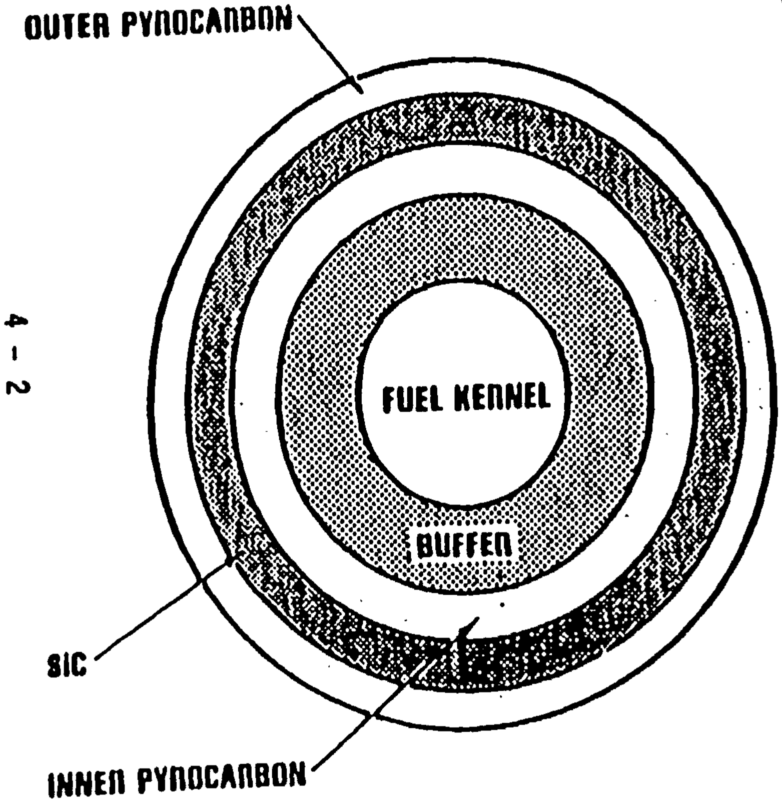
The nine licensability issues for the MHTGR design are as follows:

- Fuel Performance (Section 4.2.1)
- Fission Product Transport Computer Codes (Section 4.2.2)
- Source Term (Section 4.2.3)
- Unconventional Containment (Section 4.2.4)
- Safety Classification and Regulatory Treatment of Non-Safety-Grade Systems (Section 4.2.5)
- Completely Passive System for Ultimate Heat Sink (Section 4.2.6)
- Reactor Vessel Neutron Fluence Embrittlement (Section 4.2.7)
- Reactor Vessel Elevated Temperature Service (Section 4.2.8)
- Applied Technology Designation (Section 4.2.9)

4.2.1 Fuel Performance

The proposed fuel for the MHTGR is the TRISO multicoated microspheres which are discussed in Section 4.2 of the Preliminary Safety Information Document (PSID) ([DOE]-HTGR-86-024). It is essentially the same fuel as that approved for Fort St. Vrain, although the Department of Energy (DOE) has considered additional seal coats on the TRISO structure for the MHTGR. A picture of the fuel, from the DOE presentation of June 4-6, 1991, listed in Section 1.3 of this report, is shown in Figure 4.1. The fuel particles are formed into small, cylindrical compacts in the manufacturing process and the compacts are in large prismatic graphite blocks as shown in Figure 1.2 of this report. Fueled blocks and unfueled, or reflector, prismatic blocks will make up the core inside the reactor pressure vessel.

TRISO-COATED PARTICLE COMPONENTS



COMPONENT/PURPOSE

- FUEL KERNEL (UCO OR ThO_2)
  - — PROVIDE FISSION ENERGY
  - — RETAIN SHORT LIVED FISSION PRODUCTS
- BUFFER LAYER (POROUS CARBON LAYER)
  - — ATTENUATE FISSION RECOILS
  - — VOID VOLUME FOR FISSION GASES
- INNER PYROCARBON (IPyC)
  - — PROVIDE SUBSTRATE FOR SIC DURING MANUFACTURE
- SILICON CARBIDE (SIC)
  - — RETAIN GAS AND METAL FISSION PRODUCTS
- OUTER PYROCARBON (OPyC)
  - — PROVIDE BONDING SURFACE
  - — PROVIDE FISSION PRODUCT BARRIER IN PARTICLES WITH DEFECTIVE SIC

FIGURE 4.1 MHTGR FUEL PARTICLE

The fuel performance is related to the source term in that both are needed to determine the potential dose consequences of normal operations and accidents. The fuel performance is concerned with that fraction of the fuel that could be defective because it does not perform as designed or fails during normal operation or accidents. The source term is concerned with the composition, magnitude, and chemical and physical form of the radionuclides from the failed fuel. Source term is also a licensability issue for the MHTGR and it is discussed in Section 4.2.3 (below).

As discussed in Section 3.3 of this report, DOE has proposed a very high integrity fuel for the MHTGR, with a very small defective fuel fraction of 4.2×10^{-4} for fuel manufacturing and a failed fuel fraction of 8.0×10^{-4} for normal operation and accidents, at 95-percent confidence, to justify the high-leakage, unconventional containment for the MHTGR design. The containment is also a licensability issue for the MHTGR and it is discussed in Section 4.2.4 (below). DOE states that the fuel particles will not fail unless the fuel exceeds the threshold temperature of 1600 °C (2900 °F). The staff will need to clearly understand how the multicoated fuel particle design will ensure that this very low failed fuel fraction is not exceeded during normal operation and accidents.

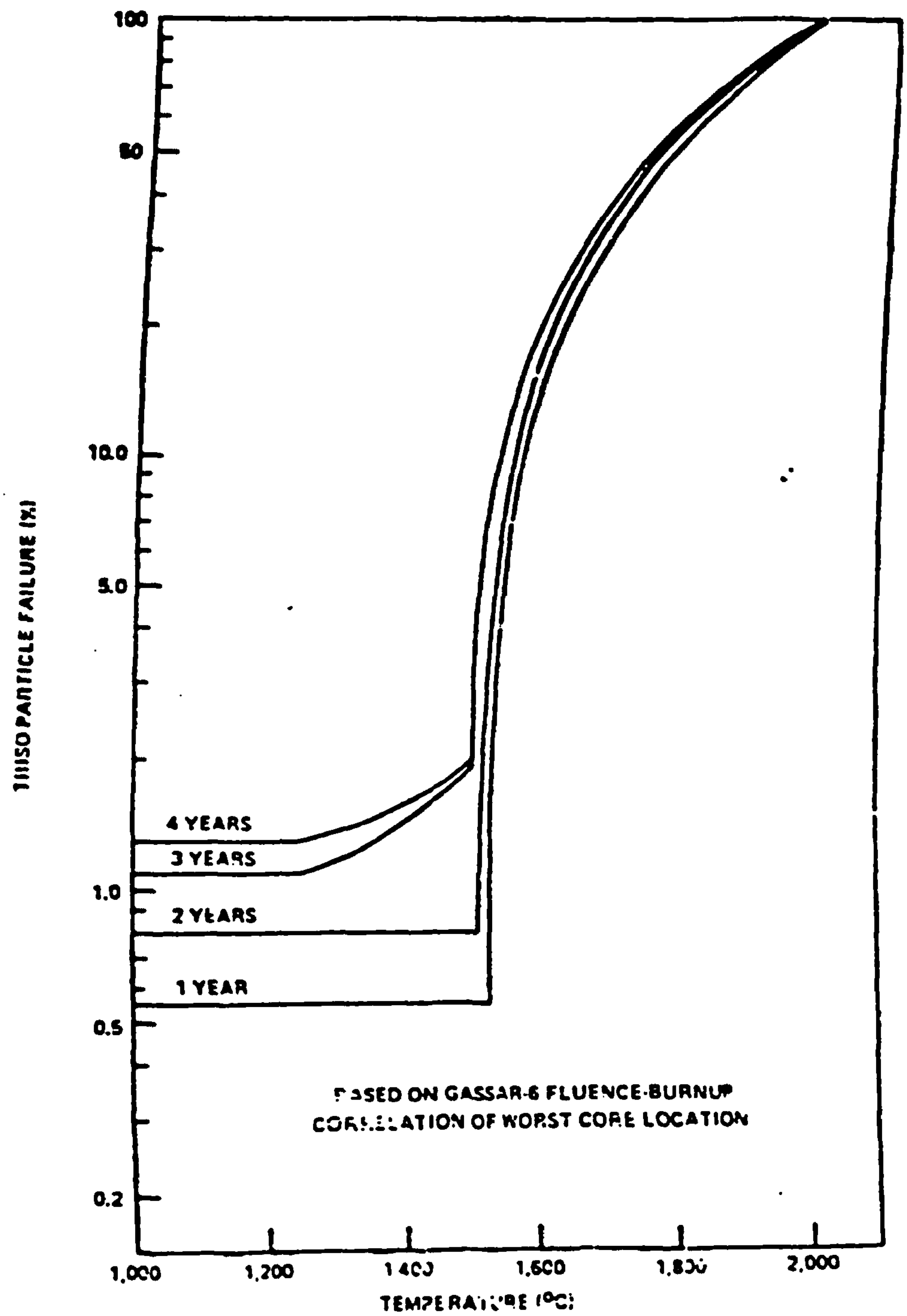
The fuel performance proposed by DOE for the MHTGR is significantly better than what the staff approved in licensing Fort St. Vrain. This is to say that the proposed failed fuel fraction for the MHTGR fuel is significantly lower than that accepted for Fort St. Vrain by about a factor of 100. When the staff licensed Fort St. Vrain, the plant did not require the high fuel integrity being proposed for the MHTGR because Fort St. Vrain had a low-leakage containment, an accepted leak rate less than 0.2 percent building volumes per day, for accidents. The MHTGR design, however, is being proposed with a vented containment and the significantly higher leakage rate of less than one building volume per day.

The staff has not approved the fuel performance proposed for the MHTGR design in any licensing action for a high-temperature gas-cooled reactor (HTGR) plant, even for Fort St. Vrain, the last HTGR licensed to operate in the United States. The staff concluded in the Final Safety Evaluation Report (FSER) for licensing Fort St. Vrain (NRC, 1968 and 1972) that the failed fuel fraction in the core could be as high as 1 percent. Even assuming such high fuel failures, the staff, however, concluded in the FSER that the manufacture of coated fuel particles was a well-developed process and the particles had been tested with satisfactory results to burnups greater than 20 percent and to fuel temperatures as high as 3600 °F (1980 °C).

The staff discussed HTGR fuel failures in Chapter V of NUREG-0111, during the staff reviews of the preliminary safety analysis reports for the Fulton (PSAR Fulton), Summit (PSAR Summit), and General Atomics standard HTGR plant (GASSAR, General Atomics Standard Safety Analysis Report, GA-A13200). In NUREG-0111, the staff discussed fuel failures for the TRISO-type fuel proposed for the MHTGR as a function of (1) the threshold fuel temperature for failure and (2) the fuel neutron fluence. This is displayed in NUREG-0111 Figure 28 and is reproduced in Figure 4.2 of this report. The fuel failures shown as a function of core-location temperature in Figure 4.2, for the 1-year to 4-year

FIGURE 4.2

TRISO PARTICLE COATING FAILURES



irradiation periods, account for (1) the defective fuel fraction from the manufacturing process (for time = zero years, which is not shown in the figure) and (2) the additional fuel failures per year from irradiation of the fuel (for time = 1 to 4 years).

Figure 4.2 could be coupled with calculations of the temperature and neutron fluence of the core during an event to determine the number of failed fuel particles in the separate regions of the core and averaged over the entire core to determine the failed fuel fraction for the core. The fraction of defective fuel in the core would be based on the fraction of the core which exceeded 1600 °C (2900 °F) and the annualized neutron fluence of the fuel particles. The time to failure would be based on the temperature rise in the delay-heatup of the core following an accident. The staff reported in NUREG-0111 Chapter V that the core-averaged failed-fuel fraction was 0.268 percent for the GASSAR standard HTGR plant, which is a significantly higher failed-fuel- fraction value than DOE has proposed for the MHTGR.

The staff's conclusions in NUREG-0111 disagree with the proposed fuel performance for the MHTGR.

The staff believes, as stated in draft NUREG-1338 Section 4.2.6, that the fuel design and quality can be developed to meet the performance objectives proposed by DOE for the MHTGR, even though it is uncertain when this will be demonstrated. However, at this time, DOE has not demonstrated the necessary design and quality of fuel to meet the performance objectives proposed for the MHTGR. DOE may need to consider a prototype MHTGR with a low-leakage containment to demonstrate the performance of the fuel for normal operation and accidents. This prototype would be licensed to operate based on a higher postulated fuel failure and lower leakage containment than the MHTGR.

For the staff to reach a determination on the MHTGR fuel, DOE needs to explain in its design approval application the relationships among the following:

(1) the design thicknesses of the fuel particle coatings and the bases for these thicknesses given the proposed fuel failures from manufacturing, normal operation (neutron fluence), and accidents (temperature)

(2) the quality control on the manufacturing process for the fuel and the resulting tolerances on the coatings

(3) the fuel performance of specific coated particles and coating tolerances demonstrated from radiation and temperature tests

(4) the expected fuel temperatures through the core during defined MHTGR accidents and the resulting volume-averaged failed fuel fraction

(5) the resulting potential dose consequences that are shown to be within acceptable limits

DOE also needs to explain why the MHTGR fission-product transport computer codes, discussed in Section 4.2.2 below, do not include the quality parameters of the manufacturing process as input to the codes and needs to address the statistical question of how many fuel particles are needed in the radiation and temperature tests to justify the proposed low failed-fuel fraction, within 95-percent certainty, for the millions of the fuel particles in the core. To address the statistical question, DOE needs to address the fraction of the core that exceeds the fuel design temperature limit during accidents and the maximum number of fuel particles in the core subject to failure from core temperature and neutron fluence. This maximum number may be small compared to the total number of fuel particles in the core.

DOE should also address the staff positions in NUREG-0111 and the problems raised by the staff in NUREG-0111 Chapter V. For example, the staff stated in NUREG-0111 that the staff fuel failure model has the rapid increase in fuel failures beginning, as shown in Figure 4.2, at about 1500 °C (2730 °F) and not at the 1600 °C (2900 °F) limit proposed for the TRISO fuel by General Atomics.

Since draft NUREG-1338 was issued, DOE has submitted its calculated fraction of the core above specific temperatures as a function of time for several accident events in its responses to Comment 15-13 of PSID Chapter R. The maximum fraction of the core above the 1600 °C (2900 °F) fuel failure limit was shown by DOE to be about 6 percent and lasting for about 100 hours.

DOE also submitted a report describing the MHTGR fuel processes and the quality control on these processes (DOE-HTGR-90257). The report documents the limiting values of the fuel coatings, but does not explain the technical basis for these values or the success of the quality control measures to meet these limits in the manufacturing process. DOE will need to submit this information in its design approval application.

The data needed by DOE to justify the fuel performance were discussed in several technical evaluation reports (TERs) on the MHTGR prepared by contractors. These reports are discussed in Chapter 7 of this report. The TERs discussed in Sections 7.3.1 and 7.3.3 stated that additional data were needed at this time. DOE also should address these data needs in its application for design approval.

4.2.2 Fission-Products Transport Computer Codes

This section discusses the General Atomics computer codes used to calculate fission-product transport from the degraded and defective fuel particles during normal operation and accidents. These codes are used to determine the dose consequences from accidents for the MHTGR to determine the acceptability of the design for the event sequences discussed in Section 5.2.1 of this report. These codes have not been approved by the staff.

In draft NUREG-1338, the staff stated that it and its contractors had not reviewed the computer codes used by DOE and discussed in Chapter 15.1 of the PSID ([DOE]-HTGR-86-024). This decision had been based on the assumption that the staff and its contractors had performed dose consequences and reactor performance analyses for draft NUREG-1338 independent from that presented by DOE in the PSID and there was, therefore, no need for the staff to review the MHTGR computer codes.

Since draft NUREG-1338 was issued, the staff has concluded that the computer codes used by the staff's contractors for draft NUREG-1338 may not have been independent of the MHTGR computer codes, and, thus, the fission-product transport codes used to calculate dose consequences for the MHTGR should be reviewed to ensure the dose consequences reported for the MHTGR were correct. The staff had a contractor, Oak Ridge National Laboratory, review the computer codes used to calculate the fission-product transport for normal operation and accidents in the MHTGR.

The contractor documented its review of the computer codes in TER 2-2-93. By letter dated July 8, 1993, the staff requested that DOE review the TER to determine what material in it was Applied Technology information; DOE responded in its letter dated August 26, 1993. The TER is not included in the appendices of this report (with other TERs and letter reports on the MHTGR design prepared by staff contractors) because of the extent of the Applied Technology information DOE stated was contained within the TER. This is discussed in Section 6.4.1 and the letters are listed in Table 6.2 of this report.

The DOE Applied Technology designation is discussed in Sections 1.8 and 4.2.9 of this report.

The contractor concluded that the basic DOE approach for the MHTGR fuel and fission-product analysis was sound; however, none of the computer codes that were reviewed were formally verified and validated. The following other points were made about the computer codes:

- The quality control parameters used in the manufacture of the fuel (i.e., microporosity, anisotropy, alpha versus beta phase content in the silicon carbide layer, and sphericity) are not input parameters for the computer codes.
- The approach was to validate integrated systems in the codes rather than individual models, which may not fully comply with Appendix K of 10 CFR Part 50.
- None of the codes has been used to predict the experimentally measured performance of the MHTGR fuel.
- From a theoretical perspective, some of the codes have significant deficiencies, as discussed in the TER.

Because of the uncertainty in these codes, there is a question about the dose consequences reported in the PSID for the accidents discussed. The codes do not need to be verified and validated for the PDA review, but must be verified and validated for the FDA and design certification review, because these codes will be used to provide the normal operation and accident dose consequences needed to determine if the final MHTGR design is acceptable. DOE should address the problems noted in the TER in the discussion on fission-products transport and the computer codes used in the application for design approval.

The staff requested that DOE review the TER and comment on the evaluations made in the document. This is discussed in Section 6.4.1 of this report. The contractor reports discussed in Sections 6.3.5, 6.3.6, 6.4.2, 6.5.3, 7.2.1, and 7.3.1 of this report indicated that insufficient technical basis for modeling certain fission-product transport pathways for the MHTGR may exist at this time.

4.2.3 Source Term

The expression "source term" refers to the specific radionuclide composition, activity, and chemical and physical form of the radioactivity available for release to the environment. The source terms are composed of two parts: (1) the radionuclides in the coolant during normal operation which are released from the plant to the environment through the radwaste treatment system and (2) the radionuclides which are released from the fuel during accidents and available for leakage from the containment to the environment. For accidents, the source term is a function of time and will involve fission-product transport from the core through the containment, including the removal of the fission products by plant features or by natural removal processes.

The source term is separate from the fuel performance which is concerned with fuel failures during normal operation and accidents. However, the fuel performance, the source term, and the containment performance (i.e., leak rate) will determine the dose consequences for accident event sequences and whether the MHTGR design is acceptable. Therefore, these three issues are interrelated. Fuel performance and containment performance are also licensability issues for the MHTGR; they are discussed in Sections 4.2.1 and 4.3.5 of this chapter. The fission-product transport codes used for the MHTGR are also a licensability issue; they are discussed in Section 4.2.2 above.

In its decision on source terms for the advanced reactors policy issues (discussed in Section 5.2.8 of this report), the Commission approved the use of mechanistic source terms for the MHTGR; however, the Commission criteria for use of mechanistic source terms is that the source terms had to be based on the fuel performance being well understood, fission-product transport being adequately modeled, and events considered in the development of source terms bounding severe accidents and design-dependent uncertainties.

DOE has not adequately addressed these criteria in its justification of the proposed MHTGR source terms; however, the relationship between and among the fuel design, fuel manufacture, fuel performance tests, and core fluence and temperatures, which is discussed in Section 4.2.1 above and which DOE must explain to the staff for the staff to evaluate the fuel performance, should address these criteria.

DOE submitted information on the radionuclide composition and magnitude of the source term in the PSID in responses to staff comments in PSID Chapter R. DOE has not provided the radionuclide chemical and physical form. This information and the deposition of radionuclides within the containment are contained within the General Atomics fission-product transport codes used to calculate radioactivity releases from the plant; the staff has neither reviewed nor approved that data. The staff did not evaluate these codes for the MHTGR in draft NUREG-1338. They are a licensability issue for the MHTGR and are discussed in Section 4.2.2 above.

As discussed in the PSID, DOE proposed a mechanistic source term for the MHTGR design based on the low failure rate of the multicoated fuel particles. The normal operation source term, as discussed in PSID Section 11.1, is that the radionuclide inventory initially in the primary system comes from a very small amount of initially defective fuel. The defective fuel comprises fuel particles with either manufacturing defects (e.g., missing or too-thin coatings) or heavy metal contamination. The steady-state radioactivity in the coolant plates out on the surfaces within the reactor coolant pressure boundary and would be removed from the coolant by the helium purification system. The plant Technical Specifications on the maximum amount of radionuclides in the coolant would limit the potential dose consequences from this radioactivity.

As discussed in PSID Section 15.1, the accident source term includes a delayed release of fission products from the delayed heatup of the core after the core temperatures exceed the fuel performance limit for failure. The fission-product transport from the fuel was calculated using computer codes. The accident source term includes plateout and deposition in the containment, but the basis for the assumed values is not explained in the PSID. With the large negative Doppler coefficient and thermal mass of the core, the peak core temperature during an accident would not reach its maximum value until days after the accident has occurred.

In draft NUREG-1338, the staff did not discuss the DOE-proposed source terms in any detail and did not identify any concerns with the source terms. The normal operation and accident source terms are discussed in draft NUREG-1338 Sections 11.1 and 15.1.2, respectively.

The staff discussed HTGR source terms in Chapter V of NUREG-0111, during the staff reviews of the safety analysis reports for the Fulton, Summit, and General Atomics standard HTGR plant (GA-A13200). The activity in the failed fuel was assumed in NUREG-0111 to be instantaneously released to the coolant with the fuel failure occurring at the time the core region reached the threshold temperature, based on the fuel temperature calculations for the accident. The fractional release to the containment of radioiodines, noble gases, and particulates from the failed fuel particles in the core was assumed to be the same fractions specified in Regulatory Guides (RGs) 1.3 and 1.4 for LWRs. These are the fractions in the LWR source term described in TID-14844, which was used by the staff to evaluate the potential dose consequences of the loss-of-coolant accident (LOCA) in the licensing of the currently operating LWRs and is referenced in the note at the end of 10 CFR 100.11. Therefore, NUREG-0111 does not provide an absolute HTGR source term because the TID-14844 radionuclide fractions were still used.

The calculation of the number of particle failures was based on the reactor being an HTGR in that the delay-heatup of the core was considered in the core temperatures reaching the 1600 °C (2900 °F) fuel failure threshold. The form of the radioiodine and the aerosol deposition rates were not discussed in NUREG-0111 because the HTGRs had no containment spray and the staff used the source term in TID-14844, which did not include the form of the radioiodine and aerosol deposition rates.

The source term that was used by the staff in licensing Fort St. Vrain was the TID-14844 source term used in licensing LWRs for the radionuclide magnitude and composition. This Fort St. Vrain source term was not the HTGR source term discussed in NUREG-0111 or the source term proposed by DOE for the MHTGR in the PSID.

The staff has been developing a mechanistic source term for the design certification reviews of the evolutionary LWRs and passive advanced LWRs. It issued NUREG-1465 to define the accident source term for use in the design certification review of these LWRS. The source term described in NUREG-1465 replaces that specified in RGs 1.3 and 1.4 for currently operating LWRs.

In NUREG-1465, the staff described the release from (1) the reactor coolant, (2) the fuel rod gap, and (3) the fuel itself, which does not occur instantaneously with the LOCA design basis accident, as assumed in TID-14844. The ACRS has commented on draft NUREG-1465 (ACRS, September 20, 1944); also, NUREG-1465 is discussed in Appendix 15A of the System 80+ FSER (NUREG-1462). The staff discussed NUREG-1465 and its application to evolutionary LWRs (System 80+) and passive advanced LWRs (AP600 and SBWR) in SECY-94-302. The discussion on removal of fission products within containment by plant features and natural removal processes would apply directly to the MHTGR.

The staff is not developing a source term for HTGRs. Therefore, taking into account the differences between HTGRs and LWRs, the discussion in NUREG-1465, SECY-94-302, and the System 80+ FSER (NUREG-1462) provides guidance to the MHTGR designers on how the staff would approach a mechanistic source term for HTGRs. The MHTGR source term provided by DOE does not describe the chemical and physical form of the radionuclides and the aerosol deposition in containment as that described in NUREG-1465.

Since draft NUREG-1338 was issued, DOE has discussed the holdup and deposition of radioactivity in the containment in its responses to Comments 15-12 and 15-16, respectively, of PSID Chapter R.

DOE needs to describe the MHTGR source terms for normal operation and for accidents, and should address how these source terms conform to the Commission criteria for use of a mechanistic source term, which is described above. DOE should submit the radionuclide composition, magnitude, and chemical and physical form for the MHTGR source terms. The fission-product transport codes should be explained in detail and DOE should show how they are consistent with the source terms. DOE should also consider the work, discussed in Section 6.3.6 of this report, on diffusion of fission products through unfailed fuel particle layers during normal operation and accidents (UofT, 1994). Such diffusion could be a significant contribution to activity released from the fuel because so few of the fuel particles in the MHTGR are assumed to fail and release radioactivity.

In the recently (October 17, 1994) proposed changes to 10 CFR Parts 50 and 100 (59 FR 52255) which will allow the use of mechanistic source terms in siting nuclear power plants, the 2-hour exposure at the site boundary is retained as part of the acceptance criteria; however, the conservative meteorology used and the 2-hour exposure interval (at the site exclusion boundary) is assumed to occur at the time during the accident which would maximize the potential dose consequences of the accident. DOE should include this shifting 2-hour exposure window in its accident dose consequence calculations.

4.2.4 Unconventional Containment

The containment for the MHTGR design will be the reactor building below grade and the containment isolation valves to isolate the secondary side, outside containment, from the steam generator. It will be a vented, high-leakage structure containing the reactor and steam generator vessels with dampers that will open to relieve the pressure pulse following a depressurization of the reactor coolant pressure boundary (RCPB) or a steamline break.

The MHTGR containment will not be a conventional leaktight, LWR containment in that the MHTGR containment will immediately vent and not retain the gases released from a rapid RCPB depressurization, and is designed to have a high leak rate of not greater than one building volume per day after this depressurization.

DOE addressed the expected leakage rate from the reactor building during accidents involving primary coolant system leakage, compared to the assumed one building volume per day in its response to Comment 15-18 in PSID Chapter R. DOE stated that the expected rate is an initial puff of gas from the building falling to a leak rate of 0.3 building volume per day.

In its decision on containment performance for the policy issues involving the advanced reactors, discussed in Section 5.2.2 of this report, the Commission decided that a conventional LWR, leaktight containment should not be required for advanced reactor designs. It approved the use of containment functional design criteria for evaluating the acceptability of proposed containment designs rather than the use of prescriptive design criteria. The approved containment design criteria follow:

- Containment designs must be adequate to meet the onsite and offsite radionuclide release limits for the event categories developed for accident selection and evaluation, discussed in Section 5.2.1 of this report.

- For approximately 24 hours following the onset of core damage, the specified containment challenge event results in no greater than the limiting containment leak rate used in evaluation of the event categories, and structural stresses are maintained within acceptable limits (i.e., American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code) Service Level C or D requirements or equivalent). After this period, the containment must prevent uncontrolled releases of radioactivity..

These criteria were policy issues submitted to the Commission and are discussed in Section 5.2.3 of this report.

The MHTGR containment must perform within these functional criteria for the staff to approve the design. DOE has addressed the first criterion in its report DOE-HTGR-90321 which was submitted in a letter dated April 13, 1993. The report addressed containment leak rate alternatives for different performance levels for the MHTGR fuel. The staff will review this document in detail in the design approval review of the MHTGR. Because the containment leak rate during accidents must result in acceptable dose consequences for the accident event categories discussed in Section 5.2.1 of this report, this leak rate will depend on the fuel performance, the transport computer codes, and the source term for the MHTGR design. Fuel performance, transport computer codes, and source term are discussed in Sections 4.2.1, 4.2.2, and 4.2.3 (above), respectively.

DOE has not addressed the second of the two criteria for the containment. Criterion 2 may affect the safety classification of the dampers used to depressurize the containment which is part of the containment isolation design. The dampers are currently classified by DOE as non-safety-grade. The issue of safety classification is discussed in Section 4.2.5 of this chapter.

The staff did not address the MHTGR containment design-basis accident in draft NUREG-1338. The proposed design-basis event for the Power Innovative Small Module (PRISM) design, as discussed in NUREG-1368 Sections 6.1.4 and 15.6.8, was a deterministic, large, primary coolant boundary breach from an undefined initiating event accompanied by a sodium pool fire. For the MHTGR, the staff believes that a similar design-basis accident would be the rapid depressurization of the coolant boundary after the failure of the fuel from the high temperatures during an event. This type of event is discussed in Section 3.4.3.6 of this report. Contractor reports on the potential rapid depressurization of the reactor coolant pressure boundary are discussed in Sections 6.3.1 and 6.4.4 of this report.

Neither the staff nor DOE has addressed containment leakage testing and containment isolation for the MHTGR. As explained in Section 6.2.6 in both the System 80+ and ABWR FSERs (NUREG-1462 and NUREG-1503), leakage testing in accordance with Appendix J of 10 CFR Part 50 is the accepted method to demonstrate the containment leak rate assumed for accident dose consequences. This testing would have to be done for the MHTGR to ensure that the actual containment leak rate was not greater than that assumed for accident dose consequences. DOE stated in Comment G.3-1 of PSID Chapter R that the LWR general design criteria (GDC) (10 CFR Part 50, Appendix A) 50 through 57 on containment testing and isolation do not apply to the MHTGR. Based on this, DOE does not require containment leak testing or isolation (except for the secondary side of the steam generator system to prevent water, from outside the reactor building, being added during an accident) for the MHTGR.

The Commission also instructed the staff to address the potential loss of the RCPB which could result in air ingress from the chimney effect and a graphite fire in the core, failure of the fuel particles, and release of radioactivity from containment to the environment. This event is discussed in Section 3.6 of this report. Also, as explained in Section 3.4.3.6 of this report, DOE has not sufficiently addressed the potential for a core depressurization event following high temperatures in the core and significant fuel damage. This accident would change the safety classification of the containment (i.e., the containment design, testing, and isolation would change) using DOE's proposed safety classification criteria.

DOE should address the Commission's containment design criteria, the containment design-basis accident and leak rate testing (including containment isolation to meet the required leak rate for accidents), the potential for a fire following a prompt criticality event discussed in Section 3.6 of this report, and the potential for a depressurization of the RCPB at high fuel temperatures with significant fuel failures, in the design approval application for the MHTGR.

4.2.5 Safety Classification and Regulatory Treatment of Non-Safety-Grade Systems

In draft NUREG-1338, the staff identified a significant number of systems which it believed should be classified as safety-related systems. The staff stated that the safety classification criteria used for the MHTGR was inconsistent with the safety classification for the current generation of LWRs and may not be correct for the MHTGR. The specific MHTGR structures, systems, and components (SSCs) identified by the staff in draft NUREG-1338 as possibly needing to be classified as safety related are indicated in DOE's Preapplication Safety Evaluation Report (PSER) tracking system for draft NUREG-1338 issues, in Appendix C of this report.

Safety classification, a policy issue for the advanced reactors, is discussed in Section 5.2.7 of this report. The staff proposed, for defense in depth, that the safety classification criteria for advanced reactors should be the same as for the current generation of LWRs. These criteria require that SSCs be classified as safety related if they were needed to (1) maintain the RCPB integrity, (2) shut down the reactor and maintain it in a safe condition, and (3) prevent the dose consequences from exceeding 10 CFR Part 100 guidelines. They protect the three fundamental barriers to release of radioactivity from the fuel: the fuel, the RCPB, and the containment. DOE proposed in the PSID ([DOE]-HTGR-86-024) that the only SSCs that should be classified as "safety related" should be those needed to mitigate the potential dose consequences for accidents to within the guidelines of 10 CFR Part 100.

The Commission stated that the staff should apply the current LWR criteria for safety classification to the advanced reactors at the preapplication review stage. This is discussed in Section 5.2.7 of this report. The staff, however, was also to consider further justification from DOE for reducing the design, installation, and maintenance requirements of the staff-identified safety-related SSCs for the MHTGR design.

The Commission also stated that the resolution of the safety classification issue must await future design developments because the MHTGR design is still at an early stage. The staff should first classify the SSCs for the passive advanced LWRs and then consider classification for the MHTGR, taking into account whether current LWR classification criteria can be applied to the MHTGR design. The staff, however, has not completed this for the passive advanced LWRs.

Draft NUREG-1338 was based on applying the current LWR safety classification criteria to the MHTGR design; the staff's conclusions in draft NUREG-1338 about safety classification of MHTGR SSCs remain valid for the application of those criteria.

The RTNSS review process, discussed in Section 5.3.14 of this report, is still being developed for the passive advanced LWRs and has not been applied to the MHTGR design. It will be applied to the MHTGR at the design approval review stage.

Since draft NUREG-1338 was issued, DOE submitted additional information on safety classification in its responses to staff comments G-28, G-31, G-32, 5-46, and 7.2-14 in Chapter R of the PSID. DOE, however, did not provide any justification for reducing the design, installation, and maintenance requirements of the staff-identified safety-related SSCs. These responses do not change the staff's position discussed in draft NUREG-1338 and in SECY-93-092. Therefore, DOE needs to address the differences between its position and the staff's position on safety classification, the effect of these differences on the specific SSCs classified as safety related, and the RTNSS review process.

The staff will resolve this issue at the design approval review stage. It will consider justification for reducing the design, installation, and maintenance requirements of the identified safety-related SSCs for the MHTGR design and will apply the RTNSS requirements for non-safety-related systems approved for the passive advanced LWRs. The fact that the MHTGR core, as designed, can not be brought down to the refueling conditions using only the safety-grade control rods will be considered in the evaluation of the MHTGR in the design approval review.

4.2.6 Completely Passive System for Ultimate Heat Sink

The ultimate heat sink for the MHTGR is the reactor cavity cooling system (RCCS), which is described in Section 5.5 of the PSID ([DOE]-HTGR-86-024) and discussed in Section 5.5 of draft NUREG-1338. The RCCS panels will surround the reactor vessel inside the reactor building, below grade, and will be connected through a ring header to inlet and outlet ports, above grade. The inlet and outlet ports will connect to the environment. The heat will be transferred from the reactor core to the environment by heat conducted to the vessel, radiated from the reactor vessel to the panels, and transferred by convection from the panels to the air inside the panels. The heat will be discharged to the environment when the hotter air rising in the outlet panels draws in cooler air from the outside.

The heat transport system (HTS) is the cooling system for normal reactor operations, startup, shutdown, and refueling. It will use the steam generator and the non-safety-grade feedwater system and condenser. The shutdown cooling system (SCS) will be a backup to the HTS if the HTS should become inoperable or if the steam generator system is not available; it uses an alternative helium circulator and heat sink for core cooling from the HTS.

DOE has stated that redundancy will be provided for the RCCS by the four separate ports and the ring header around the reactor vessel (i.e., any panel can be fed from any inlet and can discharge to any outlet) and, because the reactor vessel is underground, the decay-heat removal function could also be provided by rejection of the heat through the RCCS panels to the ground.

The completely passive RCCS is unique to the nuclear industry. It is discussed in Sections 3.2.3.2, and 4.2.6 of this report and in Section 5.5 of draft NUREG-1338. Figures 4.3 and 4.4 are drawings of the RCCS from Section 5.5 of the PSID, showing a vertical and a horizontal view, respectively, of the system. The inlet/outlet structures are above grade and both are along one side of the reactor building.

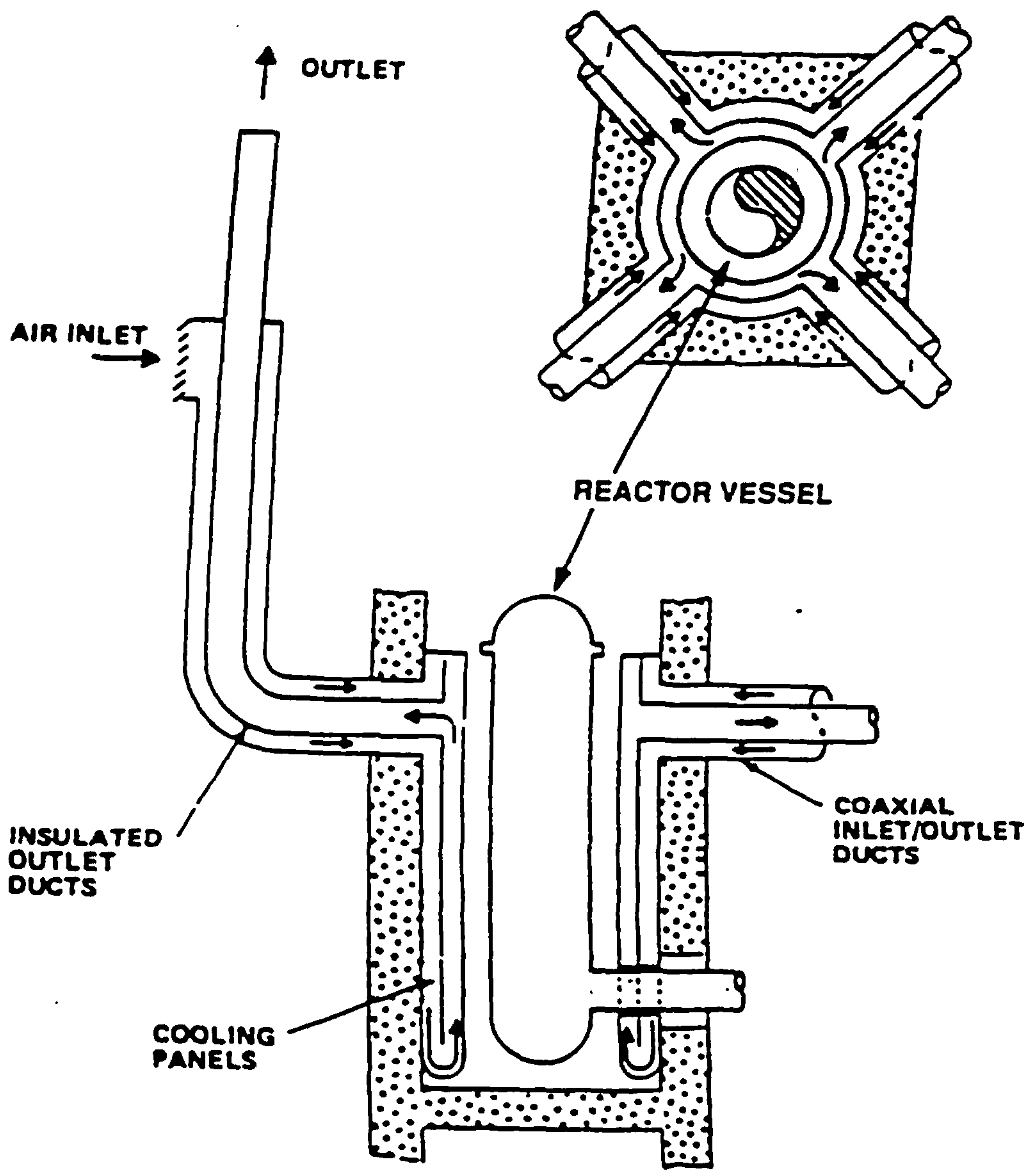
On Page 3-8 of Section 3.2 of draft NUREG-1338, the staff stated that reliance on a single system or plant feature to accomplish decay heat removal (even a highly reliable passive system) is not justifiable in a unique design, in light of the importance of this function to the protection of the public health and safety, and in view of the difficulty of predicting the failure-mode possibilities. This statement was directed toward the RCCS because it is the only safety-grade system for decay heat removal, is unique to the nuclear industry, and has a header joining four intakes and exhausts to provide redundancy in operation. This statement sets forth a determination regarding the RCCS that is too definite because the staff has not completed its evaluation of the system. The statement should say that the RCCS must be justified, because of the importance of the decay heat removal function for the safety of the core and the protection of the public health and safety, and the lack of operating experience in the industry with this design.

The Commission stated in its response to the policy issue on a single decay heat removal system for advanced reactors that reliance on a single, completely passive, safety-related residual heat removal system may be acceptable. This is discussed in Section 5.2.6 of this report. This acceptance will be in terms of the following staff concerns about the RCCS: the system reliability, the potential need for backup systems and the quality of these systems, and the ability of the RCCS to cool down the core in a reasonable time.

The staff has not developed a position on how the applicant should demonstrate the reliability of passive safety-grade systems. In developing a position, the staff has met with Westinghouse on the passive safety-grade systems in the AP600 passive advanced LWR. In the meetings of March 30 and April 20, 1995, on passive safety system performance reliability analysis, the staff presented its views on the reliability of the passive systems having two components: a system hardware performance reliability and a system thermohydraulic performance reliability. This is discussed in the two meeting summaries issued by the staff on these meetings (May 9 and 17, 1995). Although the staff discussed the reliability of the AP600 passive containment cooling system and passive core cooling system in Sections 6.2.1.1 and 6.3.2.8 of the AP600 Draft Safety Evaluation Report (DSER), NUREG-1512, passive system reliability was an unresolved item in the DSER. Passive system reliability is also discussed in SECY-95-172 for the AP600 passive advanced LWR; however, the staff concluded in that paper that the evaluation of this issue for AP600 has

FIGURE 4.3 OF THE RCCS

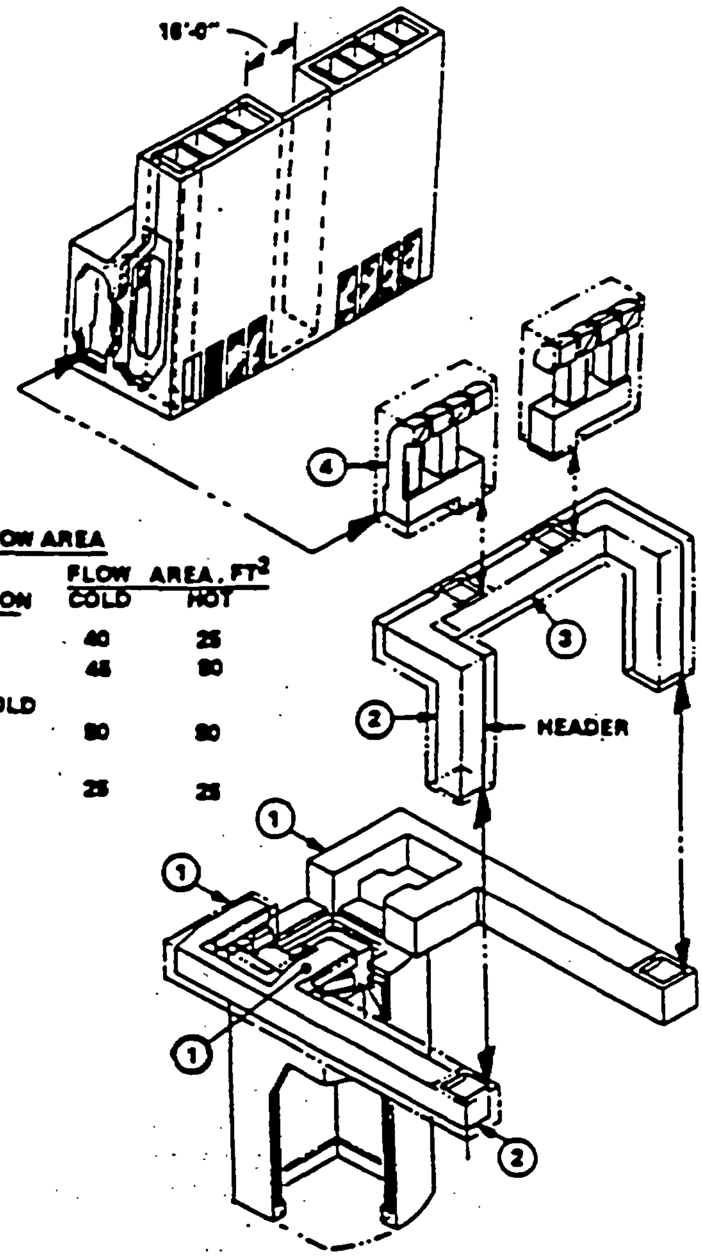
PASSIVE AIR-COOLED RCCS



1. COOLS THROUGH PANEL WALL.
2. OPERATES UNDER ALL MODES OF REACTOR OPERATION.

FIGURE 4.4 OF THE RCCS

RCCS DUCTWORK



DUCT FLOW AREA

DUCT LOCATION	DUCT DESIGNATION	FLOW AREA, FT^2 COLD	FLOW AREA, FT^2 HOT
1	DUCT	40	25
2	HEADER	45	80
3	I/O MANIFOLD PLENUM	80	80
4	INLETS/ OUTLETS	25	25

not reached a stage where a final position can be taken and did not recommend a position to the Commission.

However, in performing its detailed design evaluation, the staff is to ensure that the regulatory treatment of non-safety-related systems (RTNSS) is consistent with the Commission decisions on the design requirements for the passive advanced LWRs. RTNSS is discussed in Section 5.3.14, as a policy issue. The issue involves how to analyze a design in which a non-safety system can function as a backup to the passive safety-related system.

It should be pointed out that the "passive" systems in the passive advanced LWRs being reviewed by the staff for design certification are significantly different from the RCCS passive system for the MHTGR. The advanced LWR "passive" systems have active moving components (i.e., check valves) and the advanced LWR non-safety-grade backup systems prevent challenges to the "passive" systems.

For the MHTGR, the RCCS will be a completely passive system. The RCCS will have no moving parts and could not be started up or shut down (i.e., challenged by a startup because of a reactor protection system signal) because it would operate at all times, even during shutdown. The RCCS will work without having to be turned on and could not be turned off. Also, there will be no active non-safety systems in the MHTGR which would be required to be operational within a period of time following an event to allow the RCCS to continue operating, as is required for the passive advanced LWRs.

The RCCS instrumentation will offer the unique ability to continuously monitor the performance of the RCCS and measure any degradation of the system. Extrapolating measured RCCS performance could indicate RCCS failure. DOE provided, in its responses to staff Comments 5-5 and 5-47 in Chapter R of the PSID, values of measurable RCCS parameters for reactor shutdown, 100-percent power operation, and selected accidents. DOE also responded to staff questions concerning the sensitivity of calculated core temperatures to uncertainties in RCCS performance in its responses to staff Comments 5-4 and 5-40.

The RCCS will work more effectively at higher-than-normal HTGR reactor vessel temperatures which increase the radiant heat transfer to the RCCS panels. The use of the non-safety-grade HTS and SCS heat removal systems will reduce the frequency, magnitude, and duration of high-temperature challenges to the reactor vessel. The slow time scale (days) for MHTGR core heatup events will allow time to bring these systems back into service. The core could also discharge heat to the earth around the below-grade reactor building (see draft NUREG-1338 Section 6.2.5). The ASME Code case for higher allowed vessel temperatures is discussed in Section 4.2.8 of this report.

Current LWR criteria (i.e., General Design Criterion 34 of 10 CFR Part 50 Appendix A) require the residual heat removal system to function with only safety-grade equipment assuming a single failure within the safety systems (including power sources). Also, it is stated in RG 1.139 that an acceptable system would bring the plant down in temperature to a safe shutdown condition within 36 hours after reactor shutdown; however, it is stated in Branch

Technical Position (BTP) RES 5-1 (NUREG-0800, April 1984) that this should be performed in a reasonable period of time. For the MHTGR, which would operate and refuel at high temperatures compared to LWRs, the staff has not defined the core temperatures for such a safe shutdown condition.

For the PRISM design discussed in NUREG-1368, there is a reactor vessel auxiliary cooling system (RVACS) which is essentially the same passive decay heat removal safety system as the RCCS for the MHTGR. This system is discussed in NUREG-1368 Sections 3.1.2.8 and 5.7. The design-basis RVACS transient was the Bounding Event (BE)-3 discussed in NUREG-1368 Table 15.1:

- Loss-of-heat-sink events, from full-power conditions, assuming all cooling via the normal cooling system and auxiliary air cooling system is lost. A scram is assumed to occur when the protection system detects off-normal conditions. This event is analyzed for two cases:

  A. All airflow pathways in RVACS are assumed fully blocked for 12 hours, and sabotage is assumed on one module.

  B. Assume 75 percent blockage of the RVACS airflow pathways for an indefinite period of time, and an earthquake that affects all modules.

This event is similar to BE-3 for the MHTGR which is listed in draft NUREG-1338 Table 3.7 and Section 15.2.3.3.

Although the staff stated that 80 days were required for RVACS to handle the design-basis event and bring the reactor down to hot standby (550 °F, 290 °C), which is considerably longer than that required for LWRs, the staff did not conclude that this was an unacceptable time period. The staff also concluded that the RVACS could be unblocked within 12 hours or an alternative cooling system could be brought into operation. Neither DOE or the staff has calculated how long it may take for the MHTGR reactor to be brought down to safe shutdown conditions using only the RCCS.

In its application for design approval, DOE should address the reliability of the RCCS and its ability to cool down the reactor core in a reasonable time. It should also address the implications of RTNSS on the systems which could back up the RCCS, the use of a prototype to demonstrate the operation and redundancy of the RCCS, and the use of instrumentation to continuously monitor the performance of the RCCS. Instrumentation on the inlet and outlet of the RCCS would potentially show the performance and degradation of the RCCS. DOE discussed this matter in its response to Staff Comment 5-47 of PSID Chapter R.

DOE should also address the potential for extensive damage to the RCCS, possibly complete system failure, from a rupture of the reactor vessel or cross duct vessel, which is discussed in Sections 6.3.1 and 6.4.4 of this report. This was discussed to some extent in Appendix G.4 of the probabilistic risk assessment (PRA) report (DOE-HTGR-86011) for the MHTGR; however, this discussion should be expanded in the design approval application to describe the expected and potential RCCS damage. DOE has stated that the panels will be designed to withstand differential pressures up to 10 psi for postulated over-pressures in the reactor vessel cavity from depressurization events, or feedwater or steamline breaks.

The licensability issue of safety classification and RTNSS is discussed above, in Section 4.2.5 of this chapter.

4.2.7 Reactor Vessel Neutron Fluence Embrittlement

The reactor vessel for Fort St. Vrain was a steel-lined, prestressed concrete reactor vessel (PCRV), which served as the pressure vessel for the RCPB and the containment for fission-product retention in an accident. The MHTGR will have a steel reactor vessel housing the core and core-support structure, and the reactor vessel, along with the steam generator vessel, will be below grade within the concrete reactor building which is the containment.

The MHTGR reactor vessel will be irradiated at a lower temperature (about 400 °F or 200 °C) with a higher neutron energy spectrum than is true for LWR reactor vessels. In response to staff Comment 5-15 in Chapter R of the PSID ([DOE]-HTGR-86-024), DOE stated that the predicted shift in the nil-ductility transition temperature (NDTT) of the MHTGR steel reactor vessel caused by neutron irradiation should be less than for the current generation of pressurized LWR steel vessels because of the expected lower neutron fluence for the MHTGR vessel. The planned technology development program (DOE-HTGR-86-064) for the MHTGR design is to confirm this lower NDTT shift.

For the vessel to be approved by the staff, the NDTT shift for the MHTGR steel reactor vessel has to be within acceptable values, and the steel vessel is needed, as discussed in Section 4.2.8 below, for the RCCS to be effective. If the steel reactor vessel should not be acceptable, the MHTGR design may be significantly changed to provide an alternate safety-grade decay heat removal system to the RCCS.

The staff is concerned about:

- the effect of neutron damage on the vessel's long-term integrity and probability of failure
- the pneumatically pressurized vessel could potentially fail by catastrophic rupture rather than by a stable tearing mode characteristic of hydrostatically pressurized, gross vessel failures

As discussed in Sections 4.3.5.E, 5.2.5.A, 5.2.5.B, and 5.2.5.E of draft NUREG-1338, the staff stated that the issue of neutron damage with respect to the reactor vessel's long-term integrity, its effects on the probability of failure, and the potential for catastrophic pneumatic rupture of the vessel will remain open until the technology development program is completed. This concern is reinforced, as stated in draft NUREG-1338, by the reported neutron damage to the steel reactor vessel of a test reactor at Oak Ridge National Laboratory, which operates at low neutron fluence and low vessel temperatures. The NRC has published a status report on radiation embrittlement of LWR reactor pressure vessels, and regulations and guidelines in NUREG-1511.

DOE should address the staff's concerns about neutron fluence embrittlement of the MHTGR reactor vessel in its design approval application.

4.2.8 Reactor Vessel Elevated Temperature Service

The MHTGR reactor vessel will be an uninsulated steel pressure vessel to allow passive decay heat removal from the core fuel blocks to the atmosphere, during an LOCA. Decay heat will be removed by conduction through the core graphite to the reactor vessel, radiation from the reactor vessel to the RCCS panels in the reactor vessel cavity around the vessel, convection within the RCCS panels to the air inside the panels, and discharge of the decay heat to the environment when hot air rising within the panels draws cooler air from the outside into the panels.

The RCCS will be the only safety-grade heat removal system for the MHTGR. As discussed in Section 4.2.6 (above), there is a licensability issue about the RCCS being the single safety-grade heat removal system.

Because the decay heat from the reactor vessel will be transmitted to the RCCS panels by radiation heat transfer, the decay heat will be removed by a rise in the temperature of the reactor vessel. Certain loss-of-forced-coolant flow, or conduction-cooldown, events in the core would result in a temperature of the reactor vessel in excess of 370 °C (700 °F), which is the maximum vessel temperature currently allowed by Appendix I, Division 1, Section III of the ASME Code for the MHTGR reactor vessel materials (i.e., SA 533 Grade B, Class 1 steel plates, SA 508 Class 3 steel forgings, and their weldments). The tables in ASME Code Appendix I give the allowable stresses as a function of temperature (up to a maximum temperature) for different materials and ASME Code Service Levels A through D have stress multipliers for the allowable stresses in the ASME Code Appendix I tables.

The maximum temperature in the tables of the ASME Code Appendix I did not allow the elevated service temperature required for the reactor pressure vessel if decay heat was to be removed solely by the RCCS. And, if elevated temperature service for the reactor pressure vessel is not allowed, the MHTGR design must be changed to have some other safety-grade heat removal system in place of the RCCS (e.g., active heat removal systems in place of the passive RCCS). This would be a significant departure form the MHTGR design.

An inquiry was submitted to the ASME Code Committee requesting a special ASME Code case which would provide allowable stresses and design rules for limited elevated-temperature service of the MHTGR reactor vessel materials during Service Level C and D events. The inquiry included allowable stress values for the MHTGR reactor vessel materials for elevated-temperature service between 700 and 1000 °F (370 and 540 °C) for up to 1000 hours in ASME Code Case N-499 on elevated-temperature service of components. In its letter dated November 7, 1991, DOE explained the code inquiry and stated that the inquiry has been approved by the ASME Code main committee. The inquiry was approved for the ASME Code on December 16, 1991, and issued in "1992 Code Cases: Nuclear Components" (ASME Code, 1992).

The code case and the frequency of Service Level C and D events must be approved by the staff for the passive cooling of the reactor vessel by the RCCS to be acceptable. The staff has the following concerns:

- The code case inquiry has not been reviewed and approved by the staff.

- The service levels assigned to the conduction cooldown events resulting in the vessel's elevated temperatures have a higher probability than allowed by Table I of Standard Review Plan (SRP) 3.9.3 (NUREG-0800, July 1981).

These two concerns are discussed in Sections 5.2.5.C and 5.2.5.D of draft NUREG-1338. The staff will review Code Case N-499 during the design approval review.

The DOE proposal, in its response to staff Comment 5-45.D in Chapter R of the PSID ([DOE]-HTGR-86-024) on the expected frequency of conduction cooldown events is not acceptable to the staff. It is the staff's position that to ensure that the margins of integrity of the MHTGR steel reactor vessel are at a level comparable to that for LWR steel reactor vessels, some combination of plant systems design and additional safety analyses must be pursued to lower the expected frequency of Service Level C and D occurrences to values consistent with LWRs (i.e., Table I of SRP 3.9.3). This reference to plant system design involves the questions of safety classification and RTNSS discussed in Section 4.2.5 above.

DOE should address the staff's position on the frequency of Level C and D events for the MHTGR in its design approval application.

4.2.9 Applied Technology Designation

DOE has designated most of the information it has submitted to the staff on the MHTGR as "Applied Technology" information. For this type of information, DOE states that any distribution of such information:

> to third parties representing foreign interests, foreign governments, foreign companies and foreign subsidiaries or foreign divisions of U.S. companies shall be approved by the Associate Deputy Assistant Secretary for Reactor Systems, Development and Technology, U.S. Department of Energy. Further release may require DOE approval pursuant to Federal Regulation 10 CFR Part 810, and/or may be subject to Section 127 of the Atomic Energy Act.

This designation has been applied extensively to the MHTGR information submitted by DOE. Of the nine major documents on the MHTGR in Section 2.4 of this report, all but the containment study were designated "Applied Technology." This included the PSID and the PRA report for the MHTGR design up to 1995 as discussed below. DOE assigned the designation to entire documents and did not submit an undesignated version of the document which did not contain Applied Technology information. Also, DOE has not provided any basis for withholding design information from the public.

For an applicant other than DOE (DOE is not a person under 10 CFR 2.790), this designation would be determined to not meet the requirements of Section 2.790 of 10 CFR Part 2 which governs nondisclosure of information by NRC. It requires that without compelling reason for nondisclosure, the final NRC records and documents, including but not limited to correspondence to and from the NRC, will be made available for inspection and copying in the NRC Public Document Room. Section 2.790 does not specifically address "Applied Technology" information; however, it does allow a "balancing of the interests of the person or agency urging nondisclosure and the public interest in disclosure" and lists exceptions to this disclosure requirement in Paragraphs 2.790(a) to (e). Paragraph 2.790(a)(3) permits the nondisclosure of information specifically exempted from disclosure by statute if such statute "(i) requires that the matters be withheld from the public in such a matter as to leave no discretion on the issue, or (ii) establishes particular criteria for withholding or refers to particular types or matters to be withheld." Paragraph 2.790(a)(4) permits the nondisclosure of information that is privileged or confidential.

The DOE regulation 10 CFR Part 810, "Assistance to Foreign Atomic Energy Activities," which is being quoted to justify the nondisclosure of this information, governs the legal activities of U.S. citizens to assist foreign atomic energy operations. It implements Section 57b of the Atomic Energy Act which empowers the Secretary of Energy to authorize U.S. citizens to engage either directly or indirectly in the production of special nuclear material (i.e., plutonium, uranium-133, or uranium enriched above 0.711 percent by weight in uranium-235) outside the United States. The regulation states (1) which activities have been authorized and which require no further authorization by DOE and (2) which activities require authorization by DOE.

In complying with the DOE Applied Technology designation, the staff has not placed this information in the NRC Public Document Room and, therefore, has not released this information on the MHTGR to the public. Since 1985, when the review of this design began, this information has not been placed in public document rooms as part of the distribution of a DOE submittal or an NRC meeting summary on the MHTGR.

The Applied Technology designation of information by DOE was discussed in Section 1.8 of this report as it affected the preapplication review of the MHTGR. The staff concluded that it could complete its review of the MHTGR despite this designation on MHTGR information because none of the MHTGR design was being approved in the preapplication review.

In a letter to DOE of April 29, 1993, the staff stated its concern about complying with the Commission's objective of public disclosure of advanced reactor designs if the staff based most of its PSER on Applied Technology documents and stated that the Applied Technology designation in itself did not appear sufficient to justify withholding in their entirety these documents. In its responses dated May 26, 1993, and February 8, 1995, DOE removed the Applied Technology designation from (1) the information submitted by DOE on the PRISM design and (2) the PSID and PRA report for the MHTGR ([DOE]-HTGR-86-024 and DOE-HTGR-86011, respectively).

Of the nine DOE documents on the MHTGR listed in Section 2.4 of this report, the Applied Technology designation still remains on seven. Therefore, extensive areas of information on the MHTGR are still being withheld from the public. For example, little of the contractor report, discussed in Section 6.4.1 of this report on problems with the fission-product transport codes used for the MHTGR, is not designated as Applied Technology information.

Most of the information submitted by DOE on the PRISM design was once designated as Applied Technology information; DOE has removed the Applied Technology designation from all PRISM documents except for information on the fuel for the design (DOE letter dated May 26, 1993). Keeping the designation on the fuel may be consistent with 10 CFR Part 810. It would also be consistent with 10 CFR 2.790 because Paragraph 2.790(a)(3) specifically allows withholding information that is exempted from disclosure by statute.

This issue was not considered an obstacle to the preapplication review; nonetheless, the submittal of a design approval application with important or essential material withheld from public disclosure raises significant legal and policy issues for NRC. For design certification, there would be at least a technical violation of a statutory requirement to publish the design certification rule, because the rule would ordinarily include all essential parts of the application, and a policy issue as to the desirability of a rule, with access granted only to selected persons.

In view of the foregoing, DOE should provide in its design approval application the basis for designating design information as being required to be withheld from the public including an explanation as to how any information designated as Applied Technology falls within the scope of the Atomic Energy Act.

4.3 Conclusions

The licensability issues were identified in Chapter 3 of this report and discussed in detail in Sections 4.2.1 through 4.2.9 of this chapter. The first eight licensability issues are technical concerns with the MHTGR design. The last issue involves an administrative concern on the non-disclosure of information submitted by DOE on the MHTGR design.

The two most important licensability issues are the fuel performance and the Applied Technology designation. The issues of the fission-product transport computer codes, source term, unconventional containment, and safety classification are related to the issue of fuel performance. If the proposed fuel performance can be demonstrated, the other four issues should be able to be satisfactorily addressed for the MHTGR design.

The issue involving the passive RCCS is expected to be resolved through the demonstration of the reliability of the RCCS and the RTNSS treatment of the non-safety systems which support the RCCS. The RCSS could be demonstrated in an MHTGR prototype test for a wide range of conditions.

Although questions of neutron embrittlement of the reactor vessel still must be addressed, the Arbeitsgemeinschaft Versuchs Reaktor, Peach Bottom, and

Dragon were HTGRs that operated satisfactorily with steel reactor vessels, and it is expected that this issue will be satisfactorily addressed for the MHTGR.

Although the code case inquiry for reactor vessel elevated-temperature service has been approved by the ASME Code main committee, the staff has not reviewed the code inquiry for the MHTGR and DOE has not addressed the frequency of Service Level C and D events for the MHTGR reactor vessel.

In the matter of the Applied Technology designation, DOE should provide in its design approval application the basis for withholding design information from the public.

5. POLICY ISSUES

5.1 Introduction

In this chapter, the staff discusses the policy issues that were submitted to the Commission for the advanced reactors, and the evolutionary and advanced light-water reactors (LWRs) that are applicable to the Modular High Temperature Gas-cooled Reactor (MHTGR) design. These issues were documented in Commission papers and apply to the preapplication review and the design certification review stages. Because the advanced reactors are required to provide at least the same level of protection as the current generation of LWRs, the staff also considered the issues submitted to the Commission for the evolutionary and the advanced LWRs for applicability to the MHTGR design. As stated in Section 5.1 of NUREG-1226, the evolutionary LWRs are considered to be the current-generation LWRs.

The evolutionary designs (i.e., ABWR and System 80+) fall under 10 CFR 52.45 (a)(1) and all other designs differing significantly from them, or which use simplified, inherent, passive, or innovative means to accomplish safety functions (i.e., PRISM, MHTGR, PIUS, AP600, and SBWR) fall under 10 CFR 52.45(a)(2). For purposes of this report, these designs have been classified as evolutionary designs, advanced LWRs, and advanced reactors, as set forth below.

The Commission papers to be discussed are SECY-93-092 for the following advanced reactors:

- PRISM (Power Reactor Innovative Small Module) Reactor
- MHTGR (Modular High Temperature Gas-cooled Reactor)
- PIUS (Process Inherent Ultimate Safety) Reactor

and SECY-90-016, SECY-93-087, SECY-94-084, and SECY-95-132 for the following evolutionary and advanced LWRs:

• ABWR (Advanced Boiling Water Reactor)	Evolutionary LWR
• System 80+	Evolutionary LWR
• AP600	Advanced LWR
• SBWR (Simplified Boiling Water Reactor)	Advanced LWR

The AP600 and SBWR are classified as passive advanced LWRs because the designs rely on passive systems to perform safety functions. The MHTGR design is similar to the passive advanced LWRs in that it also relies on passive safety-related systems to perform safety functions. The passive systems are the simplified, inherent, or other innovative means to accomplish safety functions stated in the definition of advanced reactors in the Commission's Advance Reactor Policy Statement (51 FR 24643).

The five SECY papers are reproduced in Appendices E through I, respectively, of this report. Along with the SECY paper in each appendix is (1) the Commission's staff requirements memorandum (SRM), (2) the letter from the

Advisory Committee on Reactor Safeguards (ACRS), and (3) the staff's response to the ACRS letter.

Table 5.1 lists the issues alphabetically that the staff considers applicable to the MHTGR design and the Commission papers that discussed the issue (i.e., "93-092" in the table means "SECY-93-092"). The issues for the advanced reactors are listed first and separately from the issues for the evolutionary and advanced LWRs because these issues were applied to the MHTGR in SECY-93-092 and are for the preapplication review stage.

The issues for the evolutionary and advanced LWRs are issues which the staff has concluded in the preapplication review for the MHTGR (1) provide guidance to the MHTGR designers, even though the staff in the SECY paper and the Commission in the SRM did not apply the issue to the MHTGR, and (2) the Commission's decisions in the SRM were for the design certification review.

The sections listed in the table note the sections in which the issues are discussed. The discussion on the issues in this chapter involve the conclusions of the Commission and any relevant details of the MHTGR design for that issue.

SECY-93-092 is discussed in more detail than the other four papers because the issues in that paper were specific to the MHTGR, the Commission's conclusions on these issues applied to the MHTGR, and information on the issue comes from other documents on the MHTGR, such as draft NUREG-1338. The staff's recommendations in SECY-93-092 applied only to the preapplication review and do not preclude changes in the criteria for these issues in the design approval review (i.e., preliminary design approval, final design approval, or standard plant design certification under 10 CFR Part 52) for the advanced reactors.

The SECY papers for the evolutionary and advanced LWRs contain issues that the staff has concluded offer guidance to the MHTGR designers, although the Commission has not concluded that the guidance in its responses to these issues should be applied to the MHTGR. These SECY papers provide staff recommendations for issues in the design approval review process and should provide insights as to how these issues will be reviewed by the staff in the design approval review for the MHTGR design.

Also in SECY-95-172, the staff discussed key issues to date pertaining to the AP600 design including issues (Sections 5.2.1, 5.2.6, and 5.3.14 below) applicable to the MHTGR. However, because the staff discussions on these applicable issues in this SECY paper do not include staff positions and recommendations to the Commission, this SECY paper will not be discussed in detail in this report.

5.2 Advanced Reactor Issues

Of the ten issues in SECY-93-092, the following eight, listed alphabetically, apply to the MHTGR design:

- Accident Selection and Evaluation

TABLE 5.1 POLICY ISSUES APPLICABLE TO THE MHTGR

Policy Issues	SECY Papers				
	93-092	90-016	93-087	94-084	95-132
Advanced Reactor Issues					
Accident Selection and Evaluation	Section 5.2.1				
Containment Performance	Section 5.2.2				
Control Room and Remote Shutdown Area Design	Section 5.2.3				
Emergency Planning	Section 5.2.4		Section 5.2.4		
Operator Staffing and Function	Section 5.2.5				
Residual Heat Removal	Section 5.2.6				
Safety Classification	Section 5.2.7				
Source Term	Section 5.2.8				
Evolutionary and Advanced LWR Issues					
Anticipated Transients Without Scram		Section 5.3.1	Section 5.3.1		
Control Room Alarm Reliability			Section 5.3.2		
Control Room Habitability			Section 5.3.3.2	Section 5.3.3	Section 5.3.3
Defense Against Common-Mode Failures in Digital I&C Systems			Section 5.3.4		
Definition of Passive Failures				Section 5.3.5	Section 5.3.5
Electric Distribution				Section 5.3.6	Section 5.3.6

TABLE 5.1 POLICY ISSUES APPLICABLE TO THE MHTGR (Continued)

Policy Issues	SECY Papers				
	93-092	90-016	93-087	94-084	95-132
Evolutionary and Advanced LWR Issues (Continued)					
Equipment Survivability		Section 5.3.7	Section 5.3.7		
Fire Protection		Section 5.3.8	Section 5.3.8		
Industry Codes and Standards			Section 5.3.9		
Level of Detail			Section 5.3.10		
Elimination of Operating Basis Earthquake		Section 5.3.11	Section 5.3.11		
Prototype			Section 5.3.12		
Radionuclide Attenuation			Section 5.3.13		
Regulatory Treatment of Non-Safety Systems			Section 5.3.14	Section 5.3.14	Section 5.3.14
Reliability Assurance Program			Section 5.3.15	Section 5.3.15	Section 5.3.15
Role of the Passive Plant Control Room Operator			Section 5.3.16		
Safe Shutdown Requirements			Section 5.3.17	Section 5.3.17	Section 5.3.17
Severe-Accident Design Alternatives			Section 5.3.18		
Site-specific PRAs and Analysis of External Events			Section 5.3.19		
Station Blackout		Section 5.3.20	Section 5.3.20	Section 5.3.20	Section 5.3.20
Tornado Design Basis			Section 5.3.21		

- Containment Performance
- Control Room and Remote Shutdown Area Design
- Emergency Planning
- Operator Staffing and Function
- Residual Heat Removal
- Safety Classification
- Source Term

The other two issues of reactivity control system and positive void reactivity coefficient in SECY-93-092 did not apply to the MHTGR design. These issues are discussed in Attachment 1 to SECY-93-092.

The SECY-93-092 paper, the Commission SRM, and the ACRS letter (SECY-93-092 and February 19, 1993) are reproduced in Appendix E of this report. The staff response to the ACRS letter is Enclosure 5 to SECY-93-092.

The first four issues are related because the dose consequences, identified in the discussion below on the accident selection and evaluation policy issue, and the source term provide a basis for decisions on acceptable containment performance and emergency planning.

The staff requested the Department of Energy (DOE) to comment on the draft SECY-93-092 (December 16, 1992) and DOE replied on January 25, 1993. These comments are attached to SECY-93-092 in Appendix E of this report. In addition, Appendix D of this report has the list of DOE submittals on issues in draft NUREG-1338, which includes submittals on the advanced reactor issues in SECY-93-092.

5.2.1 Accident Selection and Evaluation

This issue involves the appropriate event categories, associated frequency ranges, and evaluation criteria for events that will be used to assess the safety of an advanced reactor design during the preapplication review. For current-generation LWR requirements, General Design Criterion (GDC) 4 requires that accidents be considered in the design basis (10 CFR Part 50 Appendix A).

The standards for the contents of Part 52 applications in 10 CFR 52.47 and the Commission's severe accident policy statement (50 FR 32138) require that both severe accidents and design-basis accidents (DBAs) be considered for such designs as the MHTGR that differ significantly from the current-generation LWRs or that use passive or other innovative means to accomplish safety functions. The potential dose consequences calculated for these accidents are part of this consideration.

DOE discussed event categories, frequency ranges, and criteria for the MHTGR in Chapter 15 of the Preliminary Safety Information Document (PSID) for the MHTGR ([DOE]-HTGR-86-024). DOE selected and evaluated accidents for the MHTGR design down to 10^{-8} per reactor-year, significantly less probable than the DBAs for current-generation LWRs (DOE-HTGR-86-024, -86011, and -87-001). To ensure that these accidents will have acceptable dose consequences, the dose consequences for the MHTGR are shown to be within the dose guidelines of Appendix I to 10 CFR Part 50, or at the lower limit of the Environmental

Protection Agency's (EPA's) Protective Action Guides (PAGs) (i.e., 1 rem whole body and 5 rem thyroid) (EPA-520/1-75-001).

The following event categories, associated frequency ranges, and evaluation criteria proposed by DOE for the MHTGR are discussed in Sections 15.1 and 15.2 of draft NUREG-1338:

Event Categories	Frequency Range (per plant year)	Evaluation Criteria	Ref.
Anticipated Operational Occurrences (AOOs)	$> 2.5 \times 10^{-2}$	Appendix I to 10 CFR Part 50	*
Design Basis Events (DBEs)	$< 2.5 \times 10^{-2}$ $> 1.0 \times 10^{-4}$	Lower limit PAG values**	*
Safety-Related Design Conditions (SRDCs)	$< 2.5 \times 10^{-2}$ $> 1.0 \times 10^{-4}$	Lower limit PAG values**	*
Emergency Preparedness Bounding Events (EPBEs)	$< 1.0 \times 10^{-4}$ $> 5.0 \times 10^{-7}$	Lower limit PAG values**	+
Beyond Licensing Basis Events (BLBEs)	$< 5.0 \times 10^{-7}$	Lower limit PAG values**	++

* [DOE]-HTGR-86-034
** EPA-520/1-75-001
\+ DOE-HTGR-87-001
++ DOE-HTGR-86011

The lower-limit PAG values are less than ten percent of the 10 CFR Part 100 dose guidelines (i.e., 25 rem whole body and 300 rem thyroid). The staff used Part 100 guidelines in the design certification reviews of the evolutionary and advanced LWRs.

The Commission approved, in the SRM for SECY-93-092, the staff recommendation for applying a single approach for accident evaluation to all advanced reactors during the preapplication review. The approved approach would have the following characteristics:

- Events and sequences will be selected deterministically and will be supplemented with the insights from probabilistic risk assessments (PRAs) of the specific designs.

- Categories of events will be established according to expected frequency of occurrence. One category of events that will be examined is accident sequences of a lower likelihood than traditional DBAs for LWRs. These accident sequences would be analyzed without applying the conservatisms used for DBAs. Events within a category equivalent to the current DBA category will require conservative analysis.

- Consequence acceptance limits for core damage and onsite/offsite releases will be established for each category to be consistent with

Commission policy guidance.

- Methodologies and evaluation assumptions will be developed for analyzing each category of events consistent with existing LWR practices.
- Source terms will be determined as approved by the Commission, as discussed in Section 5.2.2 of this report.
- A set of events will be selected deterministically to assess the safety margins of the proposed designs, to determine scenarios to mechanistically determine a source term, and to identify a containment challenge scenario.
- External events will be chosen deterministically as DBAs on a basis consistent with that used for LWRs.
- Evaluations of multimodule reactor designs will be considered as to whether specific events apply to some or all reactors on site for the given scenarios for all operations permitted by proposed operating practices.

The details on event categories, frequency ranges, and evaluation criteria for this single approach were not given in SECY-93-092. The details were provided, however, in Chapter 15 of NUREG-1368 which documents the preapplication review of accidents for the PRISM design. Because of timing and resource limitations, this review was not done for the MHTGR.

In NUREG-1368, the staff developed a spectrum of accidents for PRISM that was beyond the traditional DBA envelope for LWRs. This was done to (1) ensure that the advanced reactor design complied with the Commission Safety Goals and Severe Accident Policies (51 FR 28044 and 50 FR 32138, respectively), (2) sufficiently test the capability of the design to allow use of the mechanistic source term for siting determinations and for decisions involving containment design and emergency preparedness plans, and (3) ensure that the shift in emphasis in defense-in-depth from accident mitigation to accident prevention (i.e., fewer active safety-related systems), as compared to LWRs, does produce a design with safety at least equivalent to that of current-generation LWRs.

Therefore, a set of event categories corresponding to events that must be used for design, siting, containment performance, and emergency planning purposes was defined. Events in these categories were selected deterministically, supplemented by insights gained from the PRA for the design. The event categories (ECs) for accidents specified in NUREG-1368 were the following:

- EC-I: Anticipated operational occurrences with a frequency of occurrence equal to or greater than 10^{-2} per reactor-year. These accidents would be analyzed similarly to the analysis for LWRs to demonstrate compliance with Appendix I to 10 CFR Part 50 and 40 CFR 190.
- EC-II: DBAs selected similarly to that for LWRs and would include internal events down to a frequency of occurrence of 10^{-5}

per reactor-year for preapplication reviews. These accidents include a traditional selection of external DBAs and traditional conservatism presently done for LWRs to demonstrate compliance with 10 percent of the guidelines of 10 CFR Part 100.

- EC-III: Severe accidents beyond the traditional DBA envelope and chosen by engineering judgement and PRA results. The accidents include internal events down to a frequency of occurrence of 10^{-7} per reactor-year for preapplication reviews. The inclusion of external events beyond those in EC-II is consistent with their application to LWRs in the implementation of the Commission's Severe Accident Policy (50 FR 32138). The events would be analyzed on a best-estimate basis with conservative meteorology to demonstrate compliance with the guidelines of 10 CFR Part 100.

In addition, there were the bounding events that were chosen by the staff to establish confidence in the ability of the design to prevent accidents that could result in significant core damage or offsite release of radioactive material. These events were important for PRISM because the applicant was proposing a design with containment and emergency planning features significantly different from those applied to conventional LWRs. The MHTGR design also has containment and emergency planning features significantly different from those applied to conventional LWRs.

The bounding events were not rigorously qualified in terms of probability and the major assumptions were the following:

- Select worst-case plant states as initial conditions.
- Assume non-safety-grade equipment fails.
- Assume failure of safety-grade equipment for a period of time.
- Allow a reasonable time (consistent with emergency planning) to recover safety-grade equipment where no plant damage has occurred.
- Assume multiple human errors or other initiating events consistent with events that have occurred.
- Assure at least an equivalent challenge to that applied to LWRs.

Table 15.3 of NUREG-1368 showed the accident review criteria used to assess the PRISM design during the preapplication review and discussed above. In this table, it is stated in Note 3 that for relaxation of emergency planning requirements lower offsite doses must be met, but these lower doses are not specified. The PAG values proposed by DOE for the MHTGR and PRISM designs are significantly lower than 10 percent of 10 CFR Part 100 dose guidelines.

The methodology of accident selection and evaluation for preapplication reviews, discussed in SECY-93-092 and documented in NUREG-1368, is the same as that used for the LWRs in design certification reviews, except that the design certification reviews only included EC-III events down to 10^{-6} per reactor-year. The lower level for preapplication reviews ensures that significant accidents between 10^{-6} and 10^{-7} per reactor-year would be investigated in the preapplication review to determine their potential consequences. The design certification reviews of ABWR and System 80+ (NUREG-1503 and NUREG-1462, respectively) give guidance on how the spectrum of transients and accidents, and the radiological consequences of accidents, were applied to LWRs and might be applied to the advanced reactors in a design approval review. Also, the discussion on severe accidents for the ABWR and System 80+ provides guidance about how this might be applied to the advanced reactors in a design approval review.

In Chapter 15 of draft NUREG-1338 on the MHTGR design, the staff discussed the evaluation of transients and accidents for the design. The staff's conclusions in Section 15.6 are not changed by the conclusions of the Commission for this policy issue and, as discussed in Section 3.3.2.6 of this report, remain valid. Severe accidents for the MHTGR were considered in draft NUREG-1338 Sections 15.2.3 and 5.2.4 on events of lower frequency than licensing basis events and residual risks. This includes emergency-planning-basis and PRA events below 10^{-7} per reactor-year, and the bounding events postulated by the staff. As shown for the evaluation criteria on page 5-6 above, DOE has stated that the dose consequences for the MHTGR for all accidents, even down to less than 10^{-7} per reactor-year, did not exceed the PAG's at the site boundary. The staff's conclusions are in draft NUREG-1338 Section 15.6.

In accordance with the criteria used in NUREG-1368, the information submitted on the MHTGR should conform to the Commission's guidance on accident selection and evaluation for preapplication reviews, although DOE did not present a DBA for the MHTGR containment.

The staff is considering changing its dose calculation methodology and its acceptance criteria for accidents which it uses to license nuclear reactor plants. This is discussed in the section on DBA and long-term severe-accident radiological consequences in the draft Commission paper attached to a letter of May 18, 1995, from NRC to Westinghouse for the AP600 design (NRC letter May 18, 1995) and in Section X of SECY-95-172. DOE should review the information in this section in preparing the design approval application.

5.2.2 Containment Performance

This issue involves whether an advanced reactor design should be allowed to employ alternative approaches to the traditional "essentially leak-tight" containment structures for the current generation of LWRs to provide for the control of fission product releases to the environment. If the overall safety of a plant design is improved (i.e., smaller accident dose consequences outside the containment) by reducing the requirements on the containment and increasing the integrity of fuel on an advanced reactor design, then there is an incentive to improve the fuel and there is a basis for accepting a different containment design.

The current LWR containment leakage requirements are in GDC 16 and Appendix J of 10 CFR Part 50. GDC 16 requires that LWR containments provide an essentially leak-tight barrier against the uncontrolled release of radioactivity into the environment and that containment-associated systems ensure that containment design conditions important to safety are not exceeded during accidents. The requirements for LWR containment leakage testing are in Appendix J to 10 CFR Part 50.

As discussed in PSID Section 6.1.1 and draft NUREG-1338 Section 6.2, the MHTGR was not designed with a leak-tight containment barrier. Instead, the design relies upon high-integrity fuel particles to minimize radionuclide release, and on a below-grade, safety-related concrete reactor building to retain and contain any radioactive releases, and to protect against external hazards. The reactor pressure vessel and steam generator vessel will be in separate cavities within the reactor building. Non-safety-grade louvers in the building would allow immediate passage of coolant gases to the environment upon vessel rupture or safety valve release, preventing building overpressure. The building design does not include containment isolation valves for the ventilation line from the building cavities and has an open path to the environment through the drain line in the reactor cavity cooling system panels around the reactor vessel. For accident dose calculations, DOE assumed a one building volume per day containment leak rate and took credit for plateout on the building walls and surfaces.

In the SRM for SECY-93-092, the Commission approved the use of a standard based upon containment functional performance to evaluate the acceptability of proposed designs rather than to rely on prescriptive containment design criteria. Containment performance will be compared with the following accident evaluation criteria:

- Containment designs must be adequate to meet the onsite and offsite radionuclide release limits for the event categories developed for accident selection and evaluation, discussed in Section 5.2.1 above.

- For approximately 24 hours following the onset of core damage, the specified containment challenge event results in no greater than the limiting containment leak rate used in evaluation of the event categories, and structural stresses are maintained within acceptable limits (i.e., ASME Code Service Level C or D requirements or equivalent). After this period, the containment must prevent uncontrolled releases of radioactivity.

This position for containment allows the acceptance of containments with leak rates that are not "essentially leak tight" as required in GDC 16 for LWRs.

The Commission also instructed the staff to address the potential loss of the reactor coolant pressure boundary which could result in air ingress from the chimney effect and a graphite fire in the core, failure of the fuel particles, and release of radioactivity from containment to the environment. This event and the potential depressurization of the coolant boundary at high fuel temperatures, and thus after significant fuel failures, discussed in Section 3.4.3.6 of this report, would be addressed as part of the first item listed above. DOE must address these events in its design approval application.

The onset of core damage discussed in the second item listed above occurs days after the event for the MHTGR because of the high negative Doppler coefficient to shut down the reactor with increasing core temperatures and the large thermal mass of the core. Therefore, DOE must also address the structural stress in the containment and the control of releases from the containment following the onset of fuel damage during the design approval review.

The MHTGR containment design is a licensability issue. It is involved with the source term, accident selection, and fuel performance because all these issues are involved in calculating the accident dose consequences for the MHTGR design. Safety classification may also affect the containment design in that the criteria to control the release of radioactivity after core damage may require the containment louvers (i.e., containment testing and isolation) to be classified as safety related. There are GDC 50 through 57 for LWRs (10 CFR Part 50, Appendix A) on containment testing and isolation that DOE stated, in Response G.3-1 in PSID Chapter R, are not applicable to the MHTGR. This is discussed in Section 4.2.4 of this report.

5.2.3 Control Room and Remote Shutdown Area Design

This issue involves whether an advanced reactor should have a remote shutdown area and a non-seismic Category I, non-Class 1E control room in place of the current LWR requirements for a seismic Category I, Class 1E control room and an alternate shutdown panel.

The current requirements for LWRs for the control room and remote shutdown area designs are in GDCs 2, 17, and 19 of 10 CFR Part 50 and in 10 CFR Part 100. The GDCs and Part 100 require the following for the control room to operate the plant safely: (1) adequate radiation protection, (2) electrical systems meeting Class 1E requirements, (3) structures, systems, and components meeting quality standards commensurate with their importance to safety, and (4) adequate seismic plant design. For the remote shutdown area, Standard Review Plan (SRP) Section 7.4 (NUREG-0800, July 1981) states that the area should be separate from the control room, be in communication with the control room, and have Class 1E instrumentation and controls capable of bringing the reactor down to cold shutdown.

As discussed in PSID Section 6.2.7, the MHTGR plant with four reactor modules will have a non-safety-grade central control room to operate the plant and a seismic Category I remote shutdown area from which to respond to accidents. Neither the equipment in the control room nor the remote shutdown area will be Class 1E. The remote shutdown area would not contain safety-related equipment, a ventilation system for operator habitability, or a safety-grade manual scram. This is based on DOE's position that accidents in a MHTGR do not require operator response to keep the plant safe and within the specified dose consequences. The manual scrams will be non-safety-grade and will be located in both the control room and remote shutdown area. The plant will have a reactor protection system vault in each reactor module, separate from the control room area and the remote shutdown area, which will be seismic Category I and will have Class 1E instrumentation and controls.

The Commission approved the staff position that the advanced reactor designers have not justified at this time departure from current requirements for this issue. The operators remain a critical element in ensuring reactor plant safety, and the control room is the space in the plant where operators are most familiar with the surroundings and normally manage plant activities. The staff is reluctant to approve any design that would

- increase the burden on operators managing off-normal operations
- increase the frequency of evacuation of the control room during design-basis accident conditions
- possibly hamper the control or monitoring of upset conditions as an event sequence progresses

The staff believes that human performance will still play a significant role in the safety of the advanced reactor plants and that the quality of support provided by a safety-related, seismic Category I, Class 1E control room will be appropriate.

The staff also believes that any remote shutdown area should be designed to complement the main control room. Sufficient Class 1E instrumentation and controls should be available to effectively manage anticipated accidents that would cause a loss of the control room functions. The location and qualification of this area should ensure protection of the remote shutdown functions to the greatest extent possible.

Therefore, the current LWR requirements for the control room and remote shutdown area will be applied at the preapplication review stage; however, the staff will consider the preapplicant's justification for a departure from these requirements. The preapplication review was to be used to evaluate the preapplicant's design to determine whether or not a different approach to designing the control room and remote shutdown area would be acceptable. For the MHTGR design, DOE stated in its response to draft SECY-93-092 that it had submitted additional information since draft NUREG-1338 was issued on the control room design requirements. These responses were Comments G-29 and G-30 of PSID Chapter R where DOE provided an explanation for its position in the PSID on the control room and remote shutdown area, although it stated that the habitability, seismic, and fire protection criteria for the remote shutdown area was being given additional consideration for investment protection and regulatory requirements. Because of timing and resource limitations, the staff did not perform any further evaluation of the MHTGR control room and remote shutdown area other than what was presented in draft NUREG-1338 and in SECY-93-092.

The control room is not considered a licensability issue for the MHTGR because the potential changes to the MHTGR control room to meet the staff's concerns would not fundamentally change the MHTGR design. The discussion on the control room for PRISM in Section 13.2.3 of NUREG-1368 provides guidance for the MHTGR design. DOE should consider the Commission-approved position for the preapplication submittal and the discussion in NUREG-1368 in preparing its design approval application.

5.2.4 Emergency Planning

This issue involves whether advanced reactors with passive safety features should have reduced emergency planning zones and requirements. Although emergency plans are not required for design certification under 10 CFR Part 52, they are necessary for issuance of an operating license. 10 CFR 50.47 requires that no operating license can be issued unless the NRC finds that there is reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency and 10 CFR 52.79(d) requires that an application for a combined operating license must contain emergency plans which provide adequate protective measures.

DOE has proposed reduced offsite emergency planning for the MHTGR design. An MHTGR emergency plan, described in DOE-HTGR-87-001, would include any agency that would be involved in the response to a radiological emergency (i.e., sheltering and evacuating the public, and controlling the food supply) for an MHTGR plant. DOE proposed the following differences and reductions from a typical emergency plan for LWRs:

- The exclusion area boundary (EAB) of 10 CFR Part 100 may also function as the boundary of the emergency planning zone (EPZ), as may be allowed by Appendix E of 10 CFR Part 50 for gas-cooled reactors.

- There would be no rapid notification (e.g., local sirens) or annual drills for the public and offsite agencies.

DOE based these proposed differences and reductions on the following reasoning:

- predicted dose consequences estimated for the EAB/EPZ are below the lower-limit EPA sheltering PAGs, and the public can be excluded from being within the EAB

- a significantly long time is calculated for the core to return to criticality after shutting down in an accident from the Doppler coefficient, without either of the two safety-grade reactor protection systems functioning (i.e., about 37 hours)

- a significantly long time is calculated for the fuel and reactor vessel to reach maximum temperatures (i.e., about 100 hours) during an accident with only the reactor cavity cooling system functioning

DOE asserted that the public around the plant would always be outside that area where exposure could be above the PAGs, and in which members of the public may need to be sheltered or evacuated, and that there would always be ample time to notify the public and move people out if it should be necessary, considering experience with such relatively common events as hurricanes.

The DOE-proposed emergency planning for the MHTGR does not mean that there would be no offsite emergency plan developed, but rather that such a plan could have fewer details concerning movement of people, need not contain provisions for early notification of the public, and need not require periodic exercises of the offsite plan. The plans used to move people out of areas for such events as hurricanes may serve as examples because the time periods for people to respond to hurricanes are similar to the time periods for the public to respond to MHTGR core heatup during an accident.

The Commission has not approved any changes to the existing regulations governing emergency preparedness for the MHTGR design or any other advanced reactor design. The Commission stated that it was premature to reach a conclusion on emergency planning for advanced reactors and that the staff should remain open to proposals to simplify emergency planning requirements for reactors that are designed with greater safety margins. It also stated that emergency planning requirements should be correlated with the work being done on accident evaluation and source term, which are discussed in the previous two sections, to avoid unnecessary conservatism, and with the work being done on emergency planning for advanced LWRs. The staff will provide its recommendation on this issue at or before the start of the design certification review phase so that any implications on the design can be addressed.

Consistent with the current LWR regulatory approach, the staff views the inclusion of emergency preparedness by advanced reactor designers as an essential element in NRC's "defense-in-depth" philosophy. Briefly stated, this philosophy (1) requires high quality in the design, construction, and operation of nuclear plants to reduce the likelihood of malfunctions, (2) recognizes that equipment can fail and operators can make mistakes, thus requiring safety systems to reduce the chances that malfunctions or mistakes will lead to accidents that release fission products from the fuel, and (3) recognizes that, in spite of these precautions, serious fuel damage accidents can happen, thus requiring containment structures and other safety features to prevent the release of fission products off site. The added feature of emergency planning to this philosophy provides that, even in the unlikely event of an offsite fission product release, there is reasonable assurance that emergency protective actions can be taken to protect the population around nuclear power plants.

Information obtained from accident evaluations conducted, as outlined in Section 5.2.1 above, will be factored into the emergency planning requirements for advanced reactors. Based in part upon these accident evaluations, the staff will consider whether some relaxation from current requirements may be appropriate for advanced reactor offsite emergency plans. The relaxations to be considered will include, but will not be limited to, notification requirements, size of the EPZ, and frequency of the exercises, and will take into account the Commission policy decisions on passive LWR emergency planning.

In Section 13.1 of draft NUREG-1338, the staff discussed the evaluation of emergency planning for the design. Except for Section 13.1.6, the conclusions of the staff in Section 13.1 are not changed by the conclusions of the Commission for this policy issue and, as discussed in Section 3.4.3.5 of this report, remain valid.

In Section III.G of SECY-93-087, the staff addressed simplification of emergency planning for the passive advanced LWRs. The staff discussed the proposals made by Electric Power Research Institute (EPRI) to reduce emergency planning requirements on early public notification, detailed emergency planning, and provisions for offsite emergency planning drills. These proposals are similar to what DOE has proposed for the MHTGR. The staff has concluded that its resolution on these proposals should be presented in a separate SECY paper which will also discuss issues related to source term.

There are two current staff endeavors involving emergency planning. The first is SECY-95-090 giving the staff's views on how emergency planning requirements should be addressed at each phase of nuclear power plant licensing under 10 CFR Part 52. The staff briefed the Advisory Committee for Reactor Safeguards on August 5, 1993 and a notice of the availability of a draft of this paper for public comment was published in the *Federal Register* on May 20, 1994 (59 FR 26530). SECY-95-090 addressed the public comments.

The second endeavor is a progress report (February 27, 1995) to the Commission on the efforts of the staff to develop recommendations for possible simplification of emergency planning requirements for reactor designs with greater safety margins. This report addressed the Commission request in the SECY-93-092 SRM (discussed above) that the staff submit recommendations for proposed criteria and methods to justify simplifying existing emergency planning requirements. The staff stated in the report that it is concentrating on the evolutionary and passive advanced LWRs and described the parametric studies being conducted to assess industry-proposed initiatives. The staff stated that the contract work should be completed by the end of 1995; industry representatives have stated that documents for the emergency preparedness initiatives will be submitted during 1995 and 1996.

Emergency planning is also discussed for the PRISM advanced reactor in Section 13.1 of NUREG-1368, and this discussion should provide additional guidance for the MHTGR design.

DOE should reflect in its design approval application the work the staff is doing on the passive advanced LWRs in response to the Commission's SRM on SECY-93-092.

5.2.5 Operator Staffing and Function

This issue involves whether an advanced reactor design should operate with a staffing complement that is less than that required by 10 CFR 50.54(m). The current LWR requirements on staffing are in 10 CFR 50.54(m)(2). Paragraph II.C in SRP Section 13.1.2 (NUREG-0800, July 1981) states that the minimum unit shift crew for modes other than cold shutdown is two licensed senior reactor operators (one is shift supervisor), two licensed reactor operators, and two unlicensed auxiliary operators.

As discussed in PSID Section 13.2, the MHTGR design will have four reactor modules, with two modules for each of the two turbine-generator systems. The design has a shift staffing level of eight persons dedicated to plant operations: a senior licensed shift supervisor, two licensed reactor operators in the control room, and five roving non-licensed operators. This is three licensed and five non-licensed operators for four reactor modules. There will be a safety-grade computer-based plant protection and instrument system for each reactor module to indicate module status and control safety systems, and a non-safety-grade plant control, data, and instrumentation system to control the startup and operation of the four modules.

The Commission approved the staff recommendation that operator staffing may be design dependent and the staff should review the justification for a smaller crew size for the advanced reactor designs by evaluating the function and task analyses for normal operation and accident management. The analyses must demonstrate and confirm the following:

- Smaller operating crews can respond effectively to a worst-case array of power maneuvers, refueling and maintenance activities, and accident conditions.

- An accident at a single unit can be mitigated with the proposed number of licensed operators, less one, while all other units could be taken to a cold-shutdown condition from a variety of potential operating conditions, including a fire in one unit.

- The units can be safely shut down with eventual progression to a safe shutdown condition under each of the following conditions: (1) a complete loss of computer control capability, (2) a complete station blackout, or (3) a design-basis seismic event.

- The adequacy of these analyses shall be tested and demonstrated. The staff is currently recommending that an "actual control room prototype" be used for test and demonstration purposes.

DOE has not submitted such analyses for the MHTGR. In Section 13.2 of draft NUREG-1338, the staff discussed the role of the operators for the design. The conclusions of the staff are not changed by the conclusions of the Commission for this policy issue and, as discussed in Section 3.3.2.5 of this report, remain valid. DOE submitted additional information since draft NUREG-1338 was issued on the role of the operator in R 13-16, 13-17, and G-30 of the PSID. These responses do not change the staff's position discussed in draft NUREG-1338, in Section 3.3 of this report, and in SECY-93-092 above. DOE should address these analyses in its design approval application.

The role of the operator is also discussed for the PRISM advanced reactor in Section 13.2.4 of NUREG-1368 and in Section 5.3.16 below for the passive advanced LWRs. These discussions provide guidance for the MHTGR design for the design approval application.

5.2.6 Residual Heat Removal

This issue involves whether an advanced reactor should rely on a single, completely passive, safety-related residual heat removal system. GDC 34 requires that the residual heat removal function be performed using a reliable safety-grade system assuming loss of either on site or off site and a single failure in the system. Branch Technical Position (BTP) RSB 5-1 (NUREG-0800, April 1984) and Regulatory Guide (RG) 1.139, explaining the GDC, state that this safety system should be capable of bringing the plant down to a safe shutdown condition within in a reasonable period of time and within 36 hours of reactor shutdown.

The reactor cavity cooling system (RCCS), discussed in PSID Section 5.5 and draft NUREG-1338 Section 5.5, will be the only MHTGR safety-grade system for removing heat from the core. The RCCS has no active components and is always in operation; it cannot be turned off and does not need to be turned on to operate. Instruments can monitor the performance of the RCCS continuously. Redundancy is provided by the four separate ports and the ring header around the reactor vessel (i.e., any panel can be fed from any inlet and can discharge to any outlet). The RCCS is relied upon should the non-safety-grade heat transport system (HTS) and shutdown cooling system (SCS) be inoperable; however, the RCCS is always working and removing heat from the reactor vessel, even at normal power operation, depending on the temperature of the reactor vessel and the resulting radiant heat transfer.

In the SRM for SECY-93-092, the Commission stated that reliance on a single, completely passive, safety-related, residual heat removal system may be acceptable. This acceptance would be based on the demonstrated reliability and heat removal performance of the system. In evaluating the reliability of the system, the staff is to ensure that the regulatory treatment of non-safety-related backup systems (RTNSSs) for the passive, safety-related system is consistent with the Commission's decisions on the passive advanced LWR designs. The staff's evaluation to date of passive system reliability for the AP600 passive advanced LWR, which also relies on a passive safety-related residual heat removal system, is discussed in Section 4.2.6 of this report, and in Section V of SECY-95-172.

Because not approving the RCCS would result in significant changes to the MHTGR design, the RCCS is a licensability issue for the design; it is discussed in Section 4.2.6 of this report.

DOE should address the reliability of the RCCS, the time for the RCCS to bring the MHTGR core down in temperature, and RTNSS for the possible RCCS backup systems in its design approval application. RTNSS is discussed in Section 5.3.14 below for the passive advanced LWRs.

5.2.7 Safety Classification of Structures, Systems, and Components

This issue involves what criteria should be applied to the advanced reactor designs to identify the safety-related plant structures, systems, and components (SSCs). The current LWR criteria in Appendix A.VI(a)(1) of 10 CFR Part 100 state that safety-related SSCs are those required to perform the following safety functions:

- maintain the integrity of the reactor coolant pressure boundary
- shut down the reactor and maintain it in a safe condition
- prevent or mitigate the consequences of accidents to within the 10 CFR Part 100 guidelines

As shown in the PSID, the MHTGR design does not use the current LWR criteria for safety classification. For the MHTGR, the only criterion (the third criterion of the three listed above) is that the SSCs are needed to mitigate the accident dose consequences at the site boundary to below the guidelines in 10 CFR Part 100. DOE's application of this criterion to the MHTGR has resulted in the inclusion in the plant of only one barrier to the potential release of fission-product radioactivity to the public: the multicoated fuel particles. DOE has not classified the entire RCPB and containment as safety related.

Because the MHTGR design has two safety-grade reactor protection systems, the design may also meet the second criterion listed above. The only question is what is the safe shutdown temperature for the MHTGR because the design would not be able to reach cold shutdown temperatures with only safety-related control rods. This is discussed further in Section 5.3.17 on safe shutdown requirements. The design does not meet the first criterion listed above because the reactor coolant pressure boundary would not be classified safety related.

The NRC LWR safety classification criteria are based on the fundamental regulatory standard to require defense-in-depth for a reactor design and to require safety-related SSCs to protect three separate barriers to the potential release of radioactivity to the public: the fuel, the reactor coolant pressure boundary, and the containment.

The advanced reactor designs contain high-quality, non-safety-related active systems to provide defense-in-depth capabilities for reactor coolant makeup and decay heat removal. These would be the first line of defense should transients or other plant upsets occur. According to the designers, the non-safety-related systems are not required for mitigation of DBAs, but do provide alternative mitigation capability.

The Commission approved the position that the staff should apply the current LWR criteria to the advanced reactors at the preapplication review stage. However, the staff was to consider further justification from DOE for reducing the design, installation, and maintenance requirements of the identified safety-related SSCs for the MHTGR design.

The Commission further stated that the resolution of the safety classification issue must await future design developments because the MHTGR design is still at an early stage. The staff should first classify the SSCs for the passive advanced LWRs and then consider classification for the MHTGR, taking into account whether current LWR classification criteria can be applied to the MHTGR design.

In draft NUREG-1338, the staff questioned how safety classification was applied by DOE to the MHTGR and identified several systems the staff believed should be classified as safety-related. Safety classification is a licensability issue for the MHTGR design; it is discussed in Section 4.2.5 of this report.

The LWR safety classification criteria within the Commission's proposed changes to 10 CFR Parts 50 and 100 (59 FR 52255 remains the current LWR criteria. It is expected that if there are any changes to the current LWR criteria in these proposed changes it will apply to the plant designs submitted for design approval under 10 CFR Part 52.

5.2.8 Source Term

This issue involves whether mechanistic source terms should be developed in order to evaluate the advanced reactor designs. A mechanistic source term is an analysis of fission-product release based on the amount of cladding damage, fuel damage, and core damage resulting from the specific accident scenarios being evaluated. It is developed using best-estimate models of the transport of fission products from the fuel, through the coolant, through intervening holdup volume and barriers, taking into account mitigation features, and finally into the environment.

The current LWR requirements are in Appendix I to 10 CFR Part 50 (as low as is reasonable achievable, or ALARA), 10 CFR Part 100 (which references the source term in Technical Information Document (TID) 14844), and 10 CFR Part 20. They all have limitations on releases related to nuclear power plant source terms.

The source term for the MHTGR is based on the coated microspheres which contain the fuel in the kernel. Except for the failed particles that have (1) broken coatings from the manufacturing process and (2) failed coatings from the neutron fluence and high temperatures during accidents, these microspheres are designed to contain the fission products within the coatings during normal operation and transients. The source terms are the radioactivity in the coolant from the broken-coating particles and the radioactivity released from the failed-coating particles during the accident. During the accident, the radioactivity would be released immediately from the particles to the coolant when the fuel exceeded the critical fuel failure temperature. This is discussed in draft NUREG-1338 Section 11.1 and PSID Section 15.1. DOE has not submitted sufficient data to demonstrate that the MHTGR fuel performance will meet DOE's expectations of very low fuel failures from manufacturing and transients.

The Commission approved the staff's position that the source terms for advanced reactors should be based upon a mechanistic analysis if

- The performance of the reactor and fuel under normal and off-normal conditions is sufficiently well understood to permit a mechanistic analysis. Sufficient data should exist on reactor and fuel performance through research, development, and testing programs to provide adequate confidence in the approach.

- The transport of fission products can be adequately modeled for all barriers and pathways to the environs, including specific consideration of containment design. The calculations should be as realistic as possible so that the values and limitations of any mechanism or barrier are not obscured.

- The events considered in the analysis to develop the set of source terms for the design are selected to bound severe accidents and design-dependent uncertainties.

The staff does not believe that there is sufficient information at this time to accept the source term proposed by DOE for the MHTGR. Because source term is a licensability issue for the MHTGR, it is discussed in Section 4.2.3 of this report.

In draft NUREG-1338, the staff did not discuss the accident source term proposed by DOE. The steady-state source term is discussed in draft NUREG-1338 Section 11.1, and fuel failures and problems with demonstrating fuel performance are discussed in Section 4.2 of draft NUREG-1338. The conclusions of the staff in draft NUREG-1338 are not changed by the conclusions of the Commission for this policy issue and remain valid.

5.3 Evolutionary LWR and Passive Advanced LWR Issues

The evolutionary LWR and passive advanced LWR issues are discussed in the following SECY papers: SECY-90-016, SECY-93-087 (Attachment 1), SECY-94-084, and SECY-95-132 (Attachment 2). The Commission SRM, ACRS letter, and staff response to the ACRS letter for these SECY papers are the following:

Document	SECY-90-016	SECY-93-087	SECY-94-084	SECY-95-132
Commission SRM	June 26, 1990	July 21, 1993	June 30, 1994	June 28, 1995
ACRS Letter	April 26, 1990	*	Nov. 10, 1993	None
Staff Response to ACRS Letter	April 27, 1990	+	Feb. 2, 1994	None

* four letters: May 13, August 17, and September 16, 1992 (2 letters)
\+ four letters: June 12, October 22, October 23, and October 29, 1992

The issues in these SECY papers on the evolutionary LWRs and passive advanced LWRs that are applicable to the MHTGR and that should provide guidance to the MHTGR designers are discussed below. The issues are discussed in the order listed in Table 5.1 of this report, with the SECY papers that the issue was discussed in.

The Commission's decisions discussed below on the staff positions on the issues in the SECY papers are from the Commission SRM (listed above) for that SECY paper. In Attachments 1 and 2 to SECY-95-132, the staff (1) responds to the Commission SRM for SECY-94-084 and (2) presents the corresponding revisions to SECY-94-084, respectively.

5.3.1 Anticipated Transients Without Scram

Anticipated transients without scram (ATWSs) are reactor transients during which the reactor is not scrammed by the reactor protection system. For the currently operating LWRs, 10 CFR 50.63 provides the requirements to reduce the probability of an ATWS and to enhance mitigation capability if an ATWS should occur.

In the SRM for SECY-90-016, the Commission accepted the staff's position for the evolutionary LWRs that diverse scram systems should be provided for the reactor; however, the Commission stated also that acceptable ATWS consequences are a permissible alternative to diverse scram systems.

For the passive advanced LWRs, the vendors have stated that their designs will comply with EPRI design requirements on ATWSs, which are consistent with the Commission's decisions on ATWS for the evolutionary LWRs. The staff considered this policy issue resolved for the passive advanced LWRs. (This determination does not, however, approve the ATWS system for the MHTGR.)

ATWS is addressed for the MHTGR in the accident sequence SRDC-2 of Chapters 7 and 15 of the PSID. To prevent an ATWS, the safety-related reactor protection system will actuate the safety-related reserve shutdown control equipment if the safety-related reactor control rods have failed to operate. The staff, however, did not evaluate the MHTGR design for ATWS in draft NUREG-1338. DOE should consider the Commission-approved requirements discussed above for the evolutionary LWRs and passive advanced LWRs in its design approval application.

5.3.2 Control Room Alarm Reliability

The annunciator system in a nuclear power plant serves as a "first alert" to the control room operators of an abnormal state in the plant. The system focuses the operators' attention on the location and nature of the abnormality or malfunction. The extent to which this is achieved depends on the design features of the system.

For the evolutionary LWRs and passive advanced LWRs, the Commission approved the staff's recommendation that the alarm system should meet the EPRI design requirements for redundancy, independence, and separation. In addition, the alarms that are provided for manually controlled functions, for which no automatic control is provided and that are required for the safety systems to accomplish their safety functions, shall meet the applicable requirements for Class 1E equipment and circuits.

The control room alarms for the MHTGR were not described in the PSID. DOE should consider these Commission-approved requirements in designing the control room annunciator system for the MHTGR.

5.3.3 Control Room Habitability

Control room habitability implements GDC 19, which requires that adequate radiation protection be provided to the control room operators.

For the passive advanced LWRs, the staff discussed the issue in SECY-93-087, SECY-94-084, and SECY-95-132. The staff recommended the following for pressurizing and filtering the control room in SECY-94-084 and SECY-95-132:

- The concept of a passive, safety-grade control room pressurization system using bottled air to keep the operator exposure within the GDC 19 dose limits for the first 72 hours and safety-grade connections for offsite, portable air supplies if pressurization is needed after 72 hours to minimize operator exposure for the remainder of the accident is acceptable.

- Part 52 Combined license (COL) applicants must demonstrate, through performance of the applicable inspections, tests, analysis, and acceptance criteria (ITAACs), the feasibility and capability of a pressurization system, and the capability and availability of the backup air supplies.

- The regulatory treatment of the portable air supply and the non-safety-related ventilation should be in accordance with the RTNSS process discussed in SECY-94-084 and SECY-95-132.

In its SRM responding to SECY-94-084, the Commission deferred any decisions on this issue until the staff could discuss this issue further with the passive advanced LWR applicants to resolve whether to require leak-tight testing of the control room at every reload outage. The staff addressed the Commission's comments in Item D of SECY-95-132 and restated the staff recommendations given above. The Commission has approved the staff recommendation in SECY-95-132.

Control room radiation habitability during accidents for the MHTGR was not addressed in the PSID. DOE should consider the staff's recommendations for the passive advanced LWRs in its review of radiation protection of control room operators when it considers control room habitability for the MHTGR.

5.3.4 Defense Against Digital I&C Common-Mode Failures

Instrumentation and control (I&C) systems help ensure that the plant operates safely and reliably by monitoring, controlling, and protecting critical plant equipment and processes. The digital I&C systems differ significantly from the analog systems used in operating nuclear power plants in that they share more data transmission functions and more process equipment than their analog counterparts.

The Commission approved the following for digital I&C systems:

- The applicant shall assess the defense in depth and diversity of the proposed I&C system to demonstrate that any vulnerabilities to common-mode failures have been adequately addressed.

- In performing the assessment, the applicant shall analyze each postulated common-mode failure for each event that is evaluated in the accident analysis section of the safety analysis report (SAR) for the design using best-estimate methods. The applicant shall demonstrate adequate diversity within the design for each of these events.

- If a postulated common-mode failure could disable a safety function, then a diverse means, with a documented basis that the diverse means is unlikely to be subject to the same common-mode failure, shall be required to perform either the same function or a different function to achieve an acceptable consequence. The diverse or different function may be performed by a non-safety system if the system is of sufficient quality to perform the necessary function under the associated event conditions.

- A set of displays and controls located in the main control room shall be provided for manual, system-level actuation of critical safety functions and monitoring of parameters that support the safety functions. The displays and controls shall be independent and diverse from the safety computer system identified in either of the first two items above.

DOE described the digital I&C systems for the MHTGR in Chapter 7 of the PSID; however, it did not address common mode failures in these systems. DOE should consider these Commission-approved requirements in designing the digital I&C systems for the MHTGR.

5.3.5 Definition of Passive Failures

In discussing the single-failure criterion, a distinction is made between active and passive failures of the system or component. In the SRM for SECY-94-084, the Commission approved the staff's recommendation to continue using the current LWR licensing practice for passive component failures on the passive advanced LWR designs, and, except for check valves whose proper function can be demonstrated and documented, to redefine check valves in the passive safety systems as active components subject to single-failure consideration.

In SECY-95-132, the staff addressed the comments of one Commissioner in the SRM for SECY-94-084. This did not change the recommendations approved by the Commission for this issue. The Commission approved the staff recommendation in SECY-95-132.

DOE did not discuss the definition of passive failures for the MHTGR in the PSID; DOE should include this in its considerations of passive component failures and single-failure reviews for the MHTGR design.

5.3.6 Electric Distribution

This issue concerns the requirements on offsite power sources for non-safety system loads which may be relied upon as some form of backup to the safety systems. For the passive advanced LWRs, the plants rely on only passive systems which do not require electric power. The process by which the staff reviews these non-safety systems and decides if they are sufficiently important to the safety of the plant for electric power to be provided during accidents is described in RTNSS, which is discussed in Section 5.3.14 below.

In the SRM for SECY-94-084, the Commission approved the staff's recommendation to resolve the electrical distribution concerns on the passive advanced LWRs using the RTNSS process defined in SECY-94-084. Because the staff addressed the Commission's comments in the SRM for SECY-94-084 on RTNSS, the correct discussion of the RTNSS process is in SECY-95-132. The Commission approved the staff recommendation for RTNSS in SECY-95-132.

Because the MHTGR also relies solely on passive safety systems to respond to accidents, the MHTGR would be reviewed in the same manner as the passive advanced reactors. This type of review was not done in the preapplication review and DOE should consider the staff's discussion in SECY-95-132 on RTNSS in considering the non-safety systems which may be backups to the passive safety systems in the MHTGR design. These considerations should be included in the design approval application for the MHTGR.

5.3.7 Equipment Survivability

This issue involved systems and equipment required only to mitigate severe accidents and their ability to perform their intended function (e.g., environmental qualification). In the SRMs for SECY-90-016 and SECY-93-087, the Commission approved the staff's recommendation that these features for the evolutionary LWRs and passive advanced LWRs are not subject to the following requirements:

- environmental qualification of 10 CFR 50.49
- quality assurance of Appendix B to 10 CFR Part 50
- redundancy and diversity in the general design criteria of Appendix A to 10 CFR Part 50

DOE did not address equipment survivability during severe accidents in the PSID. DOE should consider this issue in its design approval application for the MHTGR.

5.3.8 Fire Protection

To minimize fire as a significant contributor to the likelihood of severe accidents for advanced reactors, the staff proposed enhanced fire protection requirements for the evolutionary LWRs in Section II.D of SECY-90-016. The Commission approved the staff's enhanced fire protection proposal, as supplemented by the staff's response to the ACRS letter on the SECY paper. These enhanced fire protection requirements with the ACRS recommendations were proposed by the staff for the passive advanced LWRs in Section I.E of SECY-93-087 and approved by the Commission in its SRM for SECY-93-087.

The staff discussed fire protection for the MHTGR in Section 9.6 of draft NUREG-1338. This review did not include the fire protection enhancements approved by the Commission for the evolutionary LWRs and passive advanced LWRs. Also, the fire protection program for the PRISM advanced reactor is discussed in Section 9.8 of NUREG-1368, although the fire protection enhancements discussed above were not applied to the design.

The discussion in NUREG-1368 and the Commission-approved enhanced fire protection in SECY-93-087 provides guidance for the MHTGR design for a design approval application.

5.3.9 Industry Codes and Standards

For the evolutionary LWRs and passive advanced LWRs, the Commission approved the staff position in Section II.A of SECY-93-087 that the staff would review the plant designs using the newest codes and standards that have been endorsed by the NRC; unapproved revisions to codes and standards would be reviewed on a case-by-case basis.

DOE should use the newest codes and standards that have been endorsed by the NRC for the MHTGR design. DOE has requested and has been issued an ASME Code approved code inquiry on elevated temperature service for the reactor pressure vessel which has not been approved by NRC. This is a licensability issue for the MHTGR, and it is discussed in Section 4.2.8 of this report.

5.3.10 Level of Detail

This issue involves the level of detail of information required for the staff to determine the adequacy of design approval applications under 10 CFR Part 52. Although the staff did not request guidance from the Commission in Section II.J of SECY-93-087, it listed the SECY papers which have addressed issues related to level of detail.

The staff concluded that this issue is applicable to all design approval applications and, therefore, DOE should review the discussion on level of detail for its application for the MHTGR.

5.3.11 Elimination of Operating Basis Earthquake

Industry experience has been that the 10 CFR Part 100 requirement that the operating basis earthquake (OBE) is at least one-half the safe shutdown earthquake (SSE) leads to designs that are governed by the OBE requirements and produce unnecessary and inconsistent margins for the SSE loading. In the SRM for SECY-90-016, the Commission approved the recommendation that the staff can consider requests to decouple the OBE from the SSE.

In Section I.M of SECY-93-087, the staff proposed criteria for an analysis using only the SSE. The Commission approved the staff's recommendation to eliminate the OBE from the design of structures, systems, and components for both evolutionary LWRs and passive advanced LWRs.

The seismic loads that DOE considered in the MHTGR design were for both the OBE and SSE. DOE should consider eliminating the OBE in the design approval application for the MHTGR.

5.3.12 Prototype

This issue concerns the need for a prototype in certifying a standard plant design under 10 CFR Part 52. In SECY-91-074, the staff described the process that it will use to assess the need for a prototype or some other demonstration facility for the advanced reactor designs. The staff will also follow the guidance in SECY-91-074 to identify the various types of testing, up to and including testing of a prototype facility, that may be needed to demonstrate that the designs are sufficiently mature to be certified.

The staff briefly discussed the need for a prototype in Section II.K of SECY-93-087; however, it did not request any guidance from the Commission.

DOE did not discuss a prototype or a safety test program for the MHTGR for design certification under 10 CFR Part 52, although it proposed a demonstration plant at an unspecified utility site. The staff discussed the prototype and the safety test program for the PRISM in Chapter 14 of NUREG-1368. This discussion and SECY-91-074 provide guidance as to whether a prototype is needed for the MHTGR and how the prototype can fit into the safety test program for the MHTGR.

5.3.13 Radionuclide Attenuation

This issue involves fission-product removal inside containment by such natural effects as plateout and by decay through holdup. The staff discussion in Section III.F of SECY-93-087 concerned the uncertainty of natural removal effects and holdup in the secondary building for the passive advanced LWRs because these designs do not have containment sprays. These attenuation methods are similar to the natural removal effects and holdup assumed inside containment for the MHTGR as discussed in Section 15.1 of the PSID.

The staff stated in SECY-93-087 that it will present its proposed resolution of this issue in a SECY paper on source term related issues. Recommendations in this paper and the Commission's decisions will serve as guidance on the staff's views about the reduction in fission products inside containments during accidents.

5.3.14 Regulatory Treatment of Non-Safety Systems (RTNSS)

The passive advanced LWRs, in contrast with the currently operating LWRs and the evolutionary LWRs, rely on safety systems that make use of the passive driving forces of buoyancy, gravity, and stored energy sources. These passive safety systems for LWRs include core and containment cooling during accidents. The systems have no pumps and the valves are powered by air, batteries, or differential pressure across the valve, and the vendors contend the systems do not need safety-grade alternating current (ac) power.

The passive advanced LWRs rely solely on the passive systems to demonstrate compliance with the acceptance criteria for DBAs; however, in addition to the passive systems, these LWRs include non-safety-grade active systems to provide defense-in-depth capabilities for the passive systems.

The RTNSS process, discussed in SECY-93-087, SECY-94-084, and SECY-95-132, is the staff's method of dealing with the uncertainties and reliability of the passive systems which are unique to the industry and have little or no proven operational performance history. Therefore, the staff's review of the passive systems must include an evaluation of whether the functional capability and availability of the active non-safety systems are needed to complement the passive systems. The specific systems and the specific reasons for uncertainties given in the discussion for RTNSS may differ between the passive advanced LWR and the MHTGR design, but how the staff deals with the non-safety-related systems does apply to the MHTGR design.

In SECY-94-084, the staff proposed a RTNSS process for maintaining appropriate regulatory oversight of these non-safety-related active systems in the passive advanced LWR designs. In the SRM, the Commission approved the staff's recommendations in SECY-94-084 concerning RTNSS and directed the staff to consider Westinghouse's comments on RTNSS for the AP600 design in its letter of May 24, 1994. In its letter of October 24, 1994, the staff issued its proposed review criteria for the systems identified in the AP600 design review to be important non-safety-related systems through this RTNSS process.

In SECY-95-132, the staff provided its revised discussion on RTNSS and addressed Westinghouse's comments on RTNSS for AP600 (letter dated May 24, 1994) in Section A of Attachment 2 to SECY-95-132. The Westinghouse letter of May 24, 1994, and the staff letter of October 24, 1994, are included with SECY-95-132 in Appendix I of this report. The Commission approved the staff recommendation on RTNSS in SECY-95-132.

For the AP600 passive advanced LWR design, the non-safety-related systems for core and containment cooling are discussed in NUREG-1512. These systems are not relied upon to provide safety functions required to mitigate DBAs; however, these non-safety systems provide defense-in-depth for the design in that they are the first line of defense to reduce or mitigate challenges to the passive systems, and that they may be required to replenish the passive systems or to perform core and containment heat removal duties after 72 hours into the event. Also, the passive safety systems have some inherent uncertainties: for example, the low differential pressures from natural circulation or gravity injection may not create sufficient force to fully open a stuck check valve, whereas the pumps in active emergency core cooling systems will develop sufficient pressure to open these valves (NUREG-1512, Chapter 1.2.2.7).

The passive safety systems in the AP600 are significantly different from the RCCS passive safety system in the MHTGR; however, how the staff may deal with the non-safety-related backup systems does apply to the MHTGR design. The active non-safety systems in the MHTGR do not prevent challenges to the RCCS and, because it operates continuously, the RCCS cannot be challenged by transients. The RCCS is truly passive having no components with moving parts, unlike the "passive" systems for the AP600. Also the active non-safety systems in the MHTGR are not required to be operational within any period of time following an event to allow the passive safety system to continue operating. Although the staff has not considered the passive safety-related systems in the MHTGR and their differences from these systems in passive advanced LWRs, the discussion on RTNSS in SECY-95-132 should provide guidance to the MHTGR designers.

The MHTGR has a passive safety-grade residual heat removal system, which also does not contain any valves; the MHTGR also has non-safety-grade systems which can be considered backups to the safety-grade system. The staff has not applied RTNSS to the MHTGR. DOE should consider RTNSS in its decisions on safety classification for SSCs for the MHTGR in its design approval application. Safety classification is discussed in Section 5.2.7 above.

5.3.15 Reliability Assurance Program

The reliability assurance program (RAP) is required for design certification to ensure that the design reliability of safety-significant SSCs is maintained throughout the life of the plant. In Section II.M of SECY-93-087, the staff discussed RAP and stated there would be a D-RAP for the design phase of the plant life cycle and an O-RAP for the construction and operation phases. The staff presented its interim position for the evolutionary LWRs and passive advanced LWRs in SECY-93-087; however, the staff did not request guidance from the Commission.

In Section E of SECY-94-084, the staff provided its position on D-RAP and O-RAP, and recommended that the position be required for evolutionary LWR and passive advanced LWR designs. In its SRM for SECY-94-084, the Commission stated the following:

- The requirement for the D-RAP should be monitored within the bounds of the Commission's Safety Goals Policy (51 FR 28044).
- D-RAP has been approved subject to the resolution of implementing D-RAP using the ITAAC process.
- O-RAP will not be required for the life of the COL, but the objectives of O-RAP should be incorporated into existing programs for maintenance or quality assurance.

The Commission did not approve the staff's position regarding a separate O-RAP. The Commission also modified the staff's statement of purpose of the RAP, on the top of page 18 of SECY-94-08, to state that the purposes are to give reasonable assurance that (1) the plant is designed, constructed, and operated in a manner that is consistent with the assumptions and risk insights for those risk-significant SSCs, (2) the risk-significant SSCs do not degrade to an unacceptable level during plant operations, (3) the frequency of transients that challenge SSCs are minimized, and (4) these SSCs function reliably when challenged. The Commission's modification to the statement of purpose is to have the RAP address the assumptions, risk insights, and degradation of risk-significant SSCs and not only their reliability.

In SECY-95-132, the staff has submitted a revised position on RAP for design certification which incorporates the Commission's comments in its SRM for SECY-94-084, and has provided a revised discussion on D-RAP in Section E of the second attachment to SECY-95-132. The Commission approved the staff recommendation on RTNSS in SECY-95-132.

The D-RAP program is discussed in Chapter 17 of the final safety evaluation reports (FSERs) for the final design approval for the ABWR and System 80+ designs (NUREG-1503 and NUREG-1462, respectively).

DOE should discuss the RAP for the MHTGR in its design certification application.

5.3.16 Role of Passive Plant Control Room Operator

In SECY-91-272, the staff discussed the role of the control room operators in a plant with passive safety systems. Specifically, these operators may use non-safety-related systems and active "investment protection" systems as the primary means to mitigate transients and accidents because the safety-related systems are passive and do not require operator action. The design of safety-related systems in the passive plants differs significantly from the design of such systems in the currently operating and the evolutionary LWRs.

In Section III.H of Attachment 1 to SECY-93-087, The Commission approved the staff's recommendation that sufficient "man-in-the-loop" testing and evaluation must be performed to focus on the control operator and the man/machine interface in the control room. In addition, the Commission stated that a fully functional integrated control room prototype is likely to be necessary for passive plant control room designs to demonstrate that function and tasks are properly integrated into the man/machine interface.

This Commission decision is consistent with the role of the control room operator set forth in SECY-93-092 and is discussed in Section 5.2.5 above. DOE should consider this guidance in its design approval application for the MHTGR.

5.3.17 Safe Shutdown Requirements

GDC 34 requires that a heat removal system be provided to remove residual heat from the reactor core so that acceptable fuel design limits are not exceeded. RG 1.139 and BTP 5-1 provide guidance for implementing this requirement and set forth conditions for cold shutdown (i.e., 93.3 °C (200 °F) for a pressurized LWR and 100 °C (212 °F) for a boiling LWR). Also, GDC 26 requires that one of the reactivity control systems shall be capable of holding the reactor core subcritical under cold conditions.

Because the passive advanced LWRs use passive removal systems for decay heat removal, they are limited by the inherent abilities of the passive heat removal processes (i.e., heat transfer into water) and cannot reduce the temperature of the reactor coolant system below the boiling point of water for the heat to be removed from the core. Even though active shutdown cooling systems are available to bring the reactor to cold shutdown or refueling conditions, these active systems are not classified as safety related. It is also true that the AP600 design, for example, cannot reach cold shutdown with safety-related control rods.

In Section III.D of SECY-93-087, the staff discussed safe shutdown requirements for passive advanced LWRs, but did not propose a maximum temperature for a safe and stable shutdown condition. The staff stated that it may not be necessary for the passive advanced LWRs to be capable of reaching cold shutdown with safety-related systems. The staff did not provide a position on this issue in SECY-93-087.

In Section C of SECY-94-084, the staff proposed that a safe stable condition for the passive advanced LWRs should be 215.6 °C (420 °F), instead of cold shutdown. This proposal was based on an acceptable passive system performance and an acceptable resolution of RTNSS issues. The staff was concerned with the 72-hour passive design basis (i.e., 72-hour capability) for the passive advanced LWRs in that the passive safety systems would be designed to have sufficient water for only 72 hours after a scram. The staff stated that a long-term, safe, stable condition can be maintained if a reliable non-safety support system or equipment is available to replenish the water needed to sustain long-term operation of the safety-related passive residual heat removal systems after the 72 hours.

The Commission approved the staff's recommendation in SECY-94-084 and stated, with respect to the 72-hour capability, that the requirements for replenishing the water for the passive safety system should be based on design-specific attributes.

Safe shutdown requirements were briefly mentioned in SECY-95-132; however, the staff did not propose any new requirements for this issue.

For the MHTGR, the passive RCCS, which relies on heat transfer to air, will not have the same lower temperature limit for performance as does the above LWR passive heat removal systems because the RCCS does not rely on heat transfer to water. Nevertheless, the RCCS does rely on radiant heat transfer from the MHTGR reactor vessel and, therefore, there will be some lower vessel temperature where the heat transfer to the RCCS becomes insignificant. This temperature will be above cold shutdown, and is also discussed in Section 4.2.6 of this report. Also, as discussed in Section 3.4.3.2, 4.2.5, and 5.2.7 of this report, the reactor is not designed to reach cold shutdown solely on the safety-related control rods. The MHTGR, however, does not have the 72-hour capability limit, after the reactor is scrammed, discussed above for the LWR passive safety systems.

A long-term, safe, stable condition of 215 °C (420 °F) may address the questions for the MHTGR regarding the slow time for the RCCS to bring the reactor down to a safe condition (in Section 4.2.6 of this report) and the fact the design would not be able to reach a cold shutdown with the safety-related control rods (in Section 5.2.7 of this report). DOE should consider this in preparing its application for design approval for the MHTGR design.

5.3.18 Severe Accident Design Alternatives

Severe-accident design alternatives for mitigating and preventing severe accidents in nuclear plant designs are severe-accident mitigation design alternatives (SAMDAs) required by Section 102(2)(c)(iii) of the National Environmental Policy Act of 1969 (NEPA) and design alternatives required by 10 CFR 50.34(f)(1)(i). These design alternatives are to assess potential improvements in a design that significantly reduce risk for severe accidents, are practical, and do not impact excessively on the plant. SAMDAs are addressed in the environmental assessment of the design and design alternatives are addressed in the safety evaluation of the design.

The staff addressed severe-accident design alternatives in SECY-91-229 and the Commission stated the following in the SRM for SECY-91-229 (October 25, 1991):

- The staff's approach in SECY-91-229 for considering the costs and benefits of design alternatives was approved.

- Applicants for final design approval and design certification must address severe-accident design alternatives and provide the rationale supporting their decisions.

In SECY-93-087, the staff further addressed SAMDAs for the evolutionary LWRs and passive advanced LWRs, but did not request any further guidance from the Commission.

In the design certification review of the evolutionary LWRs and passive advanced LWR designs, the staff requested the vendors to assess severe-accident design alternatives for their designs. The staff's evaluation of design alternatives for the evolutionary LWRs is documented in FSER Section 20.5.1.3 for the ABWR (NUREG-1503) and FSER Section 19.4 of the System 80+ (NUREG-1462). The staff's assessment of SAMDAs for evolutionary LWRs is documented in its assessment of the Technical Support Documents for the ABWR and System 80+ (60 FR 17921 and 17944, respectively, Section VI).

DOE should assess severe-accident design alternatives for the MHTGR and provide their respective costs and benefits in its final design approval and design certification applications.

5.3.19 Site-Specific PRAs and Analysis of External Events

In the SRM for SECY-93-087, the Commission stated the following for the site-specific PRA and analysis of external events in the plant design approval application:

- PRA analyses submitted in accordance with 10 CFR 52.47 should include an assessment of external and internal events.

- The use of 1.67 times the design basis safe shutdown earthquake (SSE) for a margin-type assessment of seismic events was approved.

- PRA insights will be used to support margins-type assessment of seismic events, which will consider sequence-level high confidence, low probability of failure, and fragilities for all sequences leading to core damage or containment failures up to approximately 1.67 times the ground motion acceleration of the design basis SSE.

- Fires will be evaluated by simplified probabilistic methods, such as but not limited to the Electric Power Research Institute's (EPRI's) FIVE methodology.

- Traditional probabilistic techniques should be used to evaluate internal floods.

- Bounding analyses should be performed of site-specific external events likely to be a challenge to the plant.

- The characteristics of a site should be compared to those assumed in the bounding analyses to ensure the site is enveloped.

- If the site characteristics are enveloped, the COL applicant need not perform further PRA evaluations for these external events, but should perform site-specific PRA evaluations to address any site-specific hazards for which a bounding analysis was not performed or which was not enveloped by the bounding analyses.

DOE should include the above information in the PRA for the MHTGR in its design approval application.

5.3.20 Station Blackout

The station blackout rule in 10 CFR 50.63 allows utilities several design alternatives to ensure that an operating plant can safely shut down should all offsite and onsite ac power be lost. The alternatives call for either an alternate ac power source to withstand station blackout or the capacity for coping with a station blackout.

In the SRM for SECY-90-016, the Commission approved for the evolutionary LWRs the staff position that the method to demonstrate compliance with 10 CFR 50.63 is through installation of a spare (i.e., full capacity) alternate ac power source of diverse design that is consistent with the guidance in Regulatory Guide (RG) 1.155 and is capable of powering at least one complete set of normal shutdown loads. This approval is more restrictive than the current LWR regulations on station blackout because coping with station blackout, which is permitted in 10 CFR 50.63, will not be allowed for the evolutionary LWRs.

In Section I.D of SECY-93-087, in a discussion on station blackout for the passive advanced LWRs, the staff stated that these designs do not rely on active systems for safe shutdown following an accident and the applicants have stated that safety-related diesel generators and an alternate ac power source for station blackout should not be required. The staff concluded that it was still evaluating this issue for these plants and would discuss station blackout in a later SECY paper in the context of RTNSS for the passive advanced LWRs. The staff did not request guidance from the Commission.

In Section F of SECY-94-084, the staff again discussed station blackout for the passive advanced LWRs and stated that each ac power system for the plant must be viewed both as a non-safety system and as an impact on station blackout for these designs. The staff recommended that station blackout for these designs must be addressed by evaluating the ac power sources through the RTNSS process. RTNSS is discussed in Section 5.3.14 of this report.

In the SRM for SECY-94-084, the Commission approved the staff recommendations on station blackout. Station blackout is referred to, but not discussed, in SECY-95-132 to state that the Commission approved the staff's recommendations in SECY-94-084.

DOE should consider the staff's position on station blackout in its design approval application for the MHTGR.

5.3.21 Tornado Design Basis

The staff has reevaluated the regulatory positions on the design basis tornados in RG 1.76 using the data on tornados that has become available since the RG was issued.

In the SRM for SECY-93-087, the Commission approved the staff's position that a maximum wind speed of 482 km per hour (300 mph) would be used as the design-basis tornado for the evolutionary LWRs and passive advanced LWRs. Compared to the maximum wind speeds in Regions I, II, and III given in RG 1.76, the 482 km per hour (300 mph) figure is less than that for Region I, equal to that for Region II, and greater than that for Region III.

The staff further stated in SECY-93-087 that EPRI had agreed to this maximum tornado wind speed for these LWR designs and that the COL applicant for the reactor design will also have to demonstrate that this maximum tornado windspeed design requirement is sufficient for the site-specific tornado hazards. The COL applicant would also have the option of performing a site-specific analysis to show the plant design is acceptable for the specific site.

In Section 2.4.1 of the PSID ([DOE]-HTGR-86-024), DOE stated that the design-basis wind speed for the MHTGR is 579 km per hour (360 mph), the maximum windspeed for Region I in RG 1.76. Therefore, the MHTGR conforms to the staff position on the tornado design basis.

5.4 Conclusions

In this chapter, Commission guidance on evolutionary and advanced reactors, in Commission papers SECY-90-016, SECY-93-087, SECY-93-092, SECY-94-084, and SECY-95-132, has been discussed as it applies to the MHTGR design. The issues in SECY-93-092 provided direct guidance to the MHTGR and are discussed in Section 5.2 of this chapter.

Some of the issues in SECY-90-016, SECY-93-087, SECY-94-084, and SECY-95-132 provide indirect guidance to the MHTGR designers. Although these SECY papers were written specifically for the evolutionary and passive advanced LWRs, it is the judgment of the staff that certain issues in these documents apply to the MHTGR. These issues are listed in Table 5.1 and discussed in Section 5.3 of this chapter. The Commission, however, has not applied these requirements to the MHTGR.

6. NRC CONTRACTOR REPORTS

6.1 Introduction

During the preapplication review of the Modular High Temperature Gas-cooled Reactor (MHTGR) design, the Nuclear Regulatory Commission (NRC) staff has obtained technical assistance on the review of the design from the University of Tennessee and from two of the Department of Energy's (DOE's) national laboratories: Brookhaven National Laboratory (BNL) and the Oak Ridge National Laboratory (ORNL). The work of the two laboratories for draft NUREG-1338, the staff's draft preapplication safety evaluation report (PSER) for the MHTGR design, is discussed in Section 1.9 of draft NUREG-1338.

The technical assistance work performed by these contractors for the staff, since draft NUREG-1338 was issued in March 1989, is reported in (1) 6 letter reports (LRs), (2) 6 technical evaluation reports (TERs), and (3) 8 reports in the NRC NUREG/CR series. These 20 reports are listed in Table 6.1, as well as the section in this chapter in which the contractor report is discussed. Certain of the contractor reports are reproduced in Appendix J of this report.

The staff reviewed the contractor reports to determine if they contained discussions and conclusions which

- identified a licensability issue for the MHTGR
- supported a licensability issue identified by the staff
- contradicted a licensability issue identified by the staff

In the sections that follow, conclusions are drawn about whether the recommendations in the contractor reports identified a new licensability issue or changed a licensability issue raised in Section 4.2 of this report. The statements in the contractor reports on the technology development programs needed for the MHTGR design are addressed in Chapter 7 of this report.

Any reference to "design approval" in this chapter are to preliminary design approval (PDA), final design approval (FDA), or standard plant design certification, in accordance with 10 CFR Part 52, in which the staff approves a reactor plant design.

6.2 Applied Technology Information

The Applied Technology designation which DOE has applied to most of the information submitted on the MHTGR prevents the disclosure of this information to the public. This designation is discussed in Sections 1.8 and 4.2.9 of this report.

Because the first three LRs and first five TERs in Table 6.1 were based on documents designated by DOE as having Applied Technology information, the eight contractor reports were also designated at first as containing Applied Technology information. The contractor submitted these reports to NRC with the Applied Technology label on them. Because only DOE may determine if a

TABLE 6.1 CONTRACTOR REPORTS

Contractor Report	Section
A. Letter Reports	6.3
Oak Ridge National Laboratory, Letter Report 1-20-93A, "Estimate of Air Shock Pressures Induced in the MHTGR Reactor Cavity by a Range of Vessel Failures," January 20, 1993.	6.3.1
Oak Ridge National Laboratory, Letter Report 10-27-92, "Comparison of NPR vs MHTGR Fuel Designs," October 27, 1992.	6.3.2
Oak Ridge National Laboratory, Letter Report 11-3-92, "Report on Safety Classification Differences Between the Nuclear Energy MHTGR and the NPR MHTGR," November 3, 1992.	6.3.3
Oak Ridge National Laboratory, Letter Report 11-13-92, "Hydrolysis Effects on MHTGR Fuel," November 10, 1992.	6.3.4
Brookhaven National Laboratory, Letter Report L-2213 11/93, "Initial Assessment of the Data Base for Modelling of Modular High Temperature Gas-Cooled Reactors," November 1993 (Applied Technology).	6.3.5
University of Tennessee, letter report, "Final Report, Research on Fuel Performance in Modular High-Temperature Gas-Cooled Reactors," letter dated October 7, 1994, from Paul Kasten.	6.3.6
B. Technical Evaluation Reports	6.4
Oak Ridge National Laboratory, TER 2-2-93, "Evaluation of Computer Codes Used to Calculate MHTGR Accident Dose Consequences," February 2, 1993 (Applied Technology)	6.4.1
Oak Ridge National Laboratory, TER 2-10-93, "Review and Evaluation of Recent Publications Bearing on the Fuels Sections of the Draft PSER," February 10, 1993 (Applied Technology).	6.4.2
Oak Ridge National Laboratory, TER 12-3-92, "Update of Independent Analyses Section 15.4, Preapplication Safety Evaluation Report for the MHTGR, NUREG-1338," December 3, 1993 (Applied Technology).	6.4.3
Oak Ridge National Laboratory, TER, "An Assessment of MHTGR Cavity Overpressure Accidents that May Impair Functionality of the Reactor Cavity Cooling System," June 22, 1992.	6.4.4
Oak Ridge National Laboratory, TER 12-1-92, "Factors Affecting the Relative Failure Probabilities of the MHTGR Vessels," December 1, 1992.	6.4.5

TABLE 6.1 CONTRACTOR REPORTS (Continued)

Contractor Report	Section
B. Technical Evaluation Reports (continued)	6.4
Oak Ridge National Laboratory, TER 12-16-92, "Evaluation of the DOE Standard MHTGR Containment Design Alternatives," December 11, 1992.	6.4.6
C. NUREG/CR Reports	6.5
Brookhaven National Laboratory, NUREG/CR-5261, BNL-NUREG-52174, "Safety Evaluation of MHTGR Licensing Basis Accident Scenarios," April 1989.	6.5.1
Oak Ridge National Laboratory, NUREG/CR-5514, ORNL/TM-11451, "Modelling and Performance of the MHTGR Reactor Cavity Cooling System," April 1990.	6.5.2
Oak Ridge National Laboratory, NUREG/CR-5647, ORNL/TM-11685, "Fission Product Plateout and Liftoff in the MHTGR Primary System: A Review," April 1991.	6.5.3
Oak Ridge National Laboratory, NUREG/CR-5810, ORNL/TM-12014, "Evaluation of MHTGR Fuel Reliability," July 1992.	6.5.4
Oak Ridge National Laboratory, NUREG/CR-5947, ORNL/TM-12237, "Magnitude and reactivity Consequences of Moisture Ingress into the MHTGR Core," December 1992.	6.5.5
Oak Ridge National Laboratory, NUREG/CR-5922, ORNL/TM-12179, "Modular High Temperature Gas-Cooled Reactor Short-Term Thermal Response to Flow and Reactivity Transients," February 1993.	6.5.6
Brookhaven National Laboratory, NUREG/CR-5983, BNL-NUREG-52356, "Safety Aspects of Forced Flow Cooldown Transients in Modular High Temperature Gas-Cooled Reactors," March 1993.	6.5.7
Stevenson and Associates, NUREG/CR-6358, "Assessment of United States Industry Design Codes and Standards for Application to Evolutionary and Advanced Nuclear Power Reactors," In Draft.	6.5.8

report contains or does not contain Applied Technology information, these eight contractor reports were submitted to DOE for an Applied Technology review. The fifth LR in Table 6.1 was also based on documents designated as having Applied Technology information and was originally designated by the contractor as containing Applied Technology information. By letter dated June 26, 1995, DOE was requested to review the fifth LR for Applied Technology information; in the letter of July 17, 1995, DOE stated that this report did not contain such information.

The staff submitted the eight contractor reports in the letters listed in Table 6.2 and requested that DOE identify what information in the eight reports was Applied Technology. At that time, the contractor reports were given only to DOE because of the Applied Technology designation on the reports.

After DOE responded to this request, the contractor reports were modified to remove the Applied Technology information and the modified reports were then submitted to DOE for its technical comments, and placed in the NRC Public Document Room.

Table 6.2 lists the NRC letters which separately requested that DOE identify the Applied Technology information in the eight contractor reports (discussed above) and submit technical comments on them. There were no NRC letters sent to DOE on the remaining contractor reports. Table 6.2 also lists the DOE responses to these requests.

6.3 Letter Reports

6.3.1 Letter Report 1-20-93A, "Estimate of Air Shock Pressures Induced in the MHTGR Reactor Cavity by a Range of Vessel Failures"

This letter report has no Applied Technology information.

The contractor prepared LR 1-20-93A to estimate the air shock pressures in the reactor vessel cavity of the reactor building caused by a range of failures of the MHTGR reactor pressure vessel. The TER, discussed in Section 6.4.4 of this report, investigated the general pressure rise in the reactor vessel cavity from the failure of the pressure vessel. The conclusion of that TER was that the localized damage to the reactor cavity cooling system (RCCS) panels from the shock wave, directly opposite from the site of the vessel failure, was less extensive than the damage to the RCCS panels from the general pressure rise in the reactor cavity. The RCCS is important to the design because it is the only safety-grade decay heat removal system for the MHTGR.

DOE addressed this accident in Section G.4 of Appendix G to the probabilistic risk assessment for the MHTGR and concluded that the RCCS would be damaged and may have reduced effectiveness in removing heat (DOE-HTGR-86011). DOE has stated that the calculations discussed in Section G.4 were for the general pressure rise in the reactor vessel cavity and did not take into account the shock wave from the vessel failure. The contractor's TER supports DOE's

TABLE 6.2 LISTING OF NRC AND DOE LETTERS ON CONTRACTOR REPORTS

Contractor Report*	Applied Technology Review		Technical Review	
	NRC Letter	DOE Response	NRC letter	DOE Response
TER 2-2-93	July 8, 1993	August 26, 1993	October 5, 1993	February 24, 1995
TER 2-10-93	July 23, 1993	August 26, 1993	October 12, 1993	January 18, 1995
TER 12-3-92	July 28, 1993	August 30, 1993	October 5, 1993	October 12, 1994
TER [unnumbered]	October 11, 1994	December 13, 1994	None	May 10, 1995
LR 1-20-93A	October 11, 1994	December 13, 1994	None	May 10, 1995
TER 12-1-92	October 11, 1994	December 13, 1994	None	May 10, 1995
LR 10-27-92	October 11, 1994	December 13, 1994	None	May 10, 1995
LR 11-3-92	October 11, 1994	December 13, 1994	None	May 10, 1995

* = TER = technical evaluation report; LR = letter report.

approach of disregarding the effects of the shock wave on the RCCS; however, neither DOE nor the NRC contractor calculated the extent of the damage.

This potential damage to the RCCS will need to be considered in the resolution of the licensability issue on the RCCS in Section 4.2.6 of this report and addressed at the design approval review stage.

In its letter of May 10, 1995, DOE stated that the letter report declared that an analysis (referring to the TER discussed in Section 6.4.4 below) "showed that failure sizes exceeding 0.5 ft^2 in area would cause extensive damage to the RCCS"; however, the referenced TER did not include a failure analysis of the RCCS. This statement is correct because the referenced TER used the statement in Appendix G.4 of DOE-HTGR-86011 that the overpressure in the reactor vessel cavity following the rupture of the crossduct in Figure G-22 would exceed the design pressure of the RCCS (i.e., 10 psig) and the contractor did not perform an independent analysis of the calculated overpressure on the RCCS.

6.3.2 Letter Report 10-27-92, "Comparison of NPR vs MHTGR Fuel Designs"

This letter report has no Applied Technology information.

The contractor prepared LR 10-27-92 to compare the fuels proposed for the MHTGR design and for the DOE MHTGR-New Production Reactor (MHTGR-NPR) which would produce tritium using the MHTGR concept. DOE cancelled the MHTGR-NPR program in late 1992. The fuel for the two reactor programs was similar (both were the TRISO multicoated microspheres) but not identical, and there were two separate experimental programs to demonstrate the performance of the two fuels. DOE submitted information on the MHTGR-NPR fuel in its letter of June 24, 1992.

The contractor's conclusions in the letter report follow:

- DOE has not provided technical bases for either fuel design in that the report referred to by DOE only describes the fabrication procedures.
- The MHTGR fuel is likely the more stressed fuel, because of the greater likelihood for chemical attack from fission products on the silicon carbide layer.

These conclusions do not add a new licensability issue and do not change the licensability issue on fuel performance discussed in Section 4.2.1 of this report.

Although this letter report is also included in the DOE letter of May 10, 1995, DOE did not comment on it.

6.3.3 Letter Report 11-3-92, "Report on Safety Classification Differences Between the Nuclear Energy MHTGR and the NPR MHTGR"

This letter report has no Applied Technology information.

The contractor prepared LR 11-3-92 to evaluate the differences in safety classification between the MHTGR design and the MHTGR-NPR design discussed in Section 6.3.2 above. The staff knew that DOE had approached safety classification differently in the two designs and wanted to understand the differences in order to gain a better understanding of the approach taken for the MHTGR design. DOE sent information on safety classification for the MHTGR-NPR in its letter to the NRC of June 24, 1992.

Because DOE cancelled the MHTGR-NPR program, the differences between the two MHTGR designs are no longer important. There are no significant conclusions in the letter report for the MHTGR design and the letter report does not change the licensability issue on safety classification discussed in Section 4.2.5 of this report.

In the letter of May 10, 1995, DOE identified the following statements in the letter report as incorrect:

- For the MHTGR, there is no commitment to meet the single failure criterion in the PSID and the RCCS is not considered safety-related.

- For the NPR, the shutdown cooling system, heat treatment system, production assurance protection system, and steam and water dump system are safety related.

The staff agrees that these statements in the letter report are incorrect. The DOE comments about the NPR do not affect the review of the MHTGR described herein. For the MHTGR, there is a stated commitment by DOE in the PSID to meet the single-failure criteria and the RCCS is classified as safety related.

6.3.4 Letter Report 11-13-92, "Hydrolysis Effects on MHTGR Fuel"

The contractor prepared this letter report to provide a state-of-the-art summary of the hydrolysis effects on MHTGR fuel for the range of conditions expected in a water ingress event. Activities toward this goal were only partially completed when the work was changed to only provide documentation of the current status of work in this area.

The letter report contains a description of the potential moisture effects on fuel materials (Section 2), a brief review of the most recent in-pile experiments at the Oak Ridge and Petten reactors (Section 3), and a list of hydrolysis models (Section 5).

6.3.5 Letter Report L-2213 11/93, "Initial Assessment of the Data Base for Modelling of Modular High Temperature Gas-Cooled Reactors"

This letter report contains no Applied Technology information. It documented the contractor's review of the currently available data base for modeling the MHTGR accident transients. The MHTGR accidents were identified in the report and grouped into four event categories based on the PSID and draft NUREG-1338. The existing data base was reviewed, including the technology development program (TDP) for the MHTGR, and the contractor identified areas of missing or partial data. Several current DOE technology development reports are reviewed in this report, including the four reports submitted to NRC in 1993 and listed in Section 7.1 of this report. The report also listed references from the open literature.

The contractor concluded in the letter report that a significant number of models and data do not exist for analyzing MHTGR accident transients; however, the completion of the TDP and the addition of the items listed in Table 2-1 of the letter report would provide a strong data base for the MHTGR to support design approval reviews. The following significant technical areas of missing or partial technical data or needing model development are discussed in Chapter 8 and Tables 2-1, 7-1, 7-2, and 7-5 of the letter report:

- reactor heat transfer and fluid flow
- heat transfer to the RCCS
- graphite and fuel chemical reactions with coolant
- upper plenum natural circulation

The area of compressible flow transients was also identified as needing data; however, the author concluded that such models are only of interest for very few specialized scenarios and, therefore, are not needed at this time.

DOE should address the above areas of missing or partial data at the design approval review stage.

The contractor concluded that the following technical areas had no significant data needs:

- neutron kinetics
- heat transport system and shutdown cooling system
- plena (enclosures) multidimensional radiation
- pressure tracking
- upper head cooling

Fission-product transport was also an area considered not to have significant data needs. The contractor anticipated that no significant areas of missing or partial data will be identified because the TDP for the MHTGR (DOE-HTGR-86-064) addressed fission-product retention and transport through all barriers; however, the author also stated that the individual technology plans for fission-product transport lacked sufficient detail to determine if the successful completion of the plan will produce the needed data. The author stated that based on past experience it is assumed that the DOE planned efforts will be sufficient; however, the discussion in Sections 7.2.1 and 7.3.1 of this report on fission-product transport data would indicate that this assumption may be incorrect.

In the discussion on fission-product transport, the contractor also stated that "it appears at this time that fuel integrity for normal and accident conditions up to 1600 °C can be assured, but that the fractions of initially defective fuel particles may have to be increased." This does not change the discussion of these computer codes in Section 4.2.2 of this report.

The TDPs for the MHTGR design are discussed in Chapter 7 of this report.

6.3.6 University of Tennessee, Letter Report, "Final Report, Research on Fuel Performance in Modular High-Temperature Gas-Cooled Reactors"

The contractor prepared this letter report to provide a technically sound, phenomenological basis for estimating the performance of high-temperature gas-cooled reactor (HTGR) coated-particle fuel under severe-accident conditions. This purpose was considered accomplished in that a new fuel performance model was developed which the letter report stated gave results in good agreement with the experimental data obtained under the German fuel test program and helped determine the requirements for the MHTGR fuel in order for the fuel to demonstrate satisfactory performance for the MHTGR design. Previous models had significant shortcomings which the new model is stated to overcome, although the report identifies additional experiments that are needed.

The new fuel performance model in the letter report consists of the following three parts:

- fission gas release from recoil due to heavy metal contamination of the matrix graphite material and the fuel particles
- pressure vessel failure of the fuel particle coatings, simplified diffusion through the coatings, and diffusion of fission gas release from contamination during heatup
- diffusion release of fission products from "intact" and "cracked" fuel particles, and from the fuel compacts

The contractor concluded the following:

- Substantially more research and development was need for the MHTGR fuel in order to develop and validate fuel-performance models for the fuel.
- The relatively low performance of the MHTGR fuel, compared to what DOE has proposed for the fuel, is due primarily to poor performance of the outer pyrolytic-carbon layers in the fuel caused by excessive stresses in the layers.
- There are "weak tails" in the statistical distribution of the strength of the silicon carbide layer in the MHTGR fuel compared to the German fuel.

The following research and development on the fuel was noted in the letter report to validate the proposed new model:

- Determine the nature of silicon carbide coating defects that occur during manufacture of the coated fuel particles.
- Determine the outer pyrolytic-carbon layer strength distribution in a large batch of coated fuel particles and the effect of neutron fluence on that strength.

- Perform irradiation testing and subsequent heating of unfueled particles to study the effects of natural contamination in the matrix graphite.

The letter report showed gas release fractions for the German fuel, used in the Arbeitsgemeinschaft Versuchs Reaktor (AVR), that demonstrates fuel failures up to 2100 °C (3800 °F) that may be sufficiently low for the proposed MHTGR high-leakage containment. The diffusion of fission products through intact fuel particles and the "weak tails" in the silicon carbide layer strength and thus "weak fuel particles" are discussed in Sections 3.4.3.2, 4.2.8, and 5.2.1 of this report. DOE should address this new model at the design approval review stage.

6.4 Technical Evaluation Reports

6.4.1 TER 2-2-93, "Evaluation of Computer Codes Used To Calculate MHTGR Accident Dose Consequences"

As stated in the staff request (NRC letter of October 5, 1993) to DOE for technical comments on the TER, the "Applied Technology" areas in TER 2-2-93 were so extensive that the staff did not develop a modified TER with the Applied Technology information removed. Therefore, this TER was not included in the staff's request to DOE and is not in Appendix J of this report.

TER 2-2-93 is an evaluation of the fission-product transport computer codes used for the MHTGR design. Because of the problems identified in these codes, the codes are considered a licensability issue for the MHTGR design and are discussed in Section 4.2.2 of this report. Problems with modeling of fission-product transport is also discussed in Section 6.5.3 of this report.

In its letter of February 24, 1995, DOE commented on TER 2-2-93. The fundamental concern with this TER is that the TER did not acknowledge the verification and validation (V&V) plan for the computer codes discussed in the TER. DOE stated that the TER does not acknowledge that the designers were aware of the current limitations of the computer codes and had proposed plans to upgrade the component models in the codes and to verify and validate the codes to the quality assurance standards of Part 2.7 of ANSI/ASME Standard NQA-2 (ANSI/ASME NQA-2). This, DOE stated, was part of the Technology Development Plan for the MHTGR (DOE-HTGR-86-064). The schedule is to complete the V&V plan before the completion of the final MHTGR design. For the staff, this would have to mean that the codes are verified and validated before staff completed its FDA review of the MHTGR design.

There were extensive other comments from DOE on the TER. These additional comments, however, are not important for this report because the staff is not approving the use of these computer codes for the MHTGR. The important point for this report is that the computer codes discussed in the TER have not been verified and validated, and DOE plans to complete this V&V before the final MHTGR design would be approved.

6.4.2 TER 2-10-93, "Review and Evaluation of Recent Publications Bearing on the Fuel Sections of the Draft PSER"

The contractor prepared TER 2-10-93 to recommend changes to Sections 4.1 and 4.2 of draft NUREG-1338, on the MHTGR fuel, based on 10 technical reports on the fuel which were completed since draft NUREG-1338 was issued. Draft NUREG-1338 documented the staff's preapplication review of the MHTGR design as of its issue date, March 1989. The 10 technical reports are listed on page 1 of TER 2-10-93 and some of the 10 technical reports were submitted by DOE in its letters of July 9 and 16, and October 2, 1991.

Among the technical reports, which are discussed in separate sections of the TER, are the following:

- DOE-HTGR-90257, "MHTGR Fuel Process and Quality Control Description," September 1991

- DOE-HTGR-86-027,"MHTGR Fuel/Fission Product Technology Development Program," April 1987

- NUREG/CR-5810, "Evaluation of MHTGR Fuel Reliability," 1992

- DOE-HTGR-85107, "U.S./FRG Accident Conditions Fuel Performance Models," 1989

The recommended revisions to Sections 4.1 and 4.2 of draft NUREG-1338 are in Chapters 4 and 5, respectively, of TER 2-10-93. The recommended revisions do not resolve problems addressed in draft NUREG-1338; however, they note that additional work since draft NUREG-1338 was issued must be addressed by both DOE and NRC, including new information on fuel failure and performance models, and fuel manufacturing. DOE should address this additional work at the design approval review stage.

The recommendations in the TER do not add a new licensability issue and do not change the licensability issue on fuel performance discussed in Section 4.2.1 of this report.

In its letter of January 18, 1995, DOE commented on the following technical reports in the TER:

- Meyers, B.F., "The Effect of Water Vapor on Fission Gas Release From MHTGR Compacts," DOE-HTGR-88486, August 1991

- Meyers, B.F., "Experiment HFRB1: A Preliminary Evaluation of Water-Vapor Injection into Capsule 3," ORNL/TM-11846, Oak Ridge National Laboratory, October 1991

- Richards, M.B., "Preliminary Evaluation of Petten Fuel Hydrolysis Data," DOE-HTGR-88506, September 1990

- U.S. Department of Energy, DOE-HTGR-86-027, "MHTGR Fuel/Fission Product Technology Development Plan," April 1987

- U.S. Department of Energy, DOE-HTGR-86-064, "Regulatory Technology Development Plan for the MHTGR," January 1987

- General Atomics (GA), "Technical Support Document of the MHTGR Fuel Product Specification, " Scheffel and Tang, GA Document 903728, Issue D, June 1989 (Applied Technology)

- U.S. Nuclear Regulatory Commission, "Evaluation of MHTGR Fuel Reliability," NUREG/CR-5810, July 1992 (this document is also discussed in Section 6.5.4 of this chapter.)

The main discussion in the DOE letter, in Section 2 of the enclosure, was on the concept of "weak fuel" which DOE stated was being extended in the TER from the concept stated by the staff in Section 4.2.5.D of draft NUREG-1338. DOE further stated that the TER extended this concept by suggesting that a non-mechanistic factor be applied to the dose consequence evaluations (i.e., accident dose consequences) for the MHTGR as a means to account for the unforeseen fuel failures. DOE explained that its position is that the credible mechanisms for "weak fuel" will be addressed by the completion of the MHTGR fuel development program which will ensure that all mechanisms for fuel failure are recognized and quantitatively accounted for in the fuel performance models. Therefore, DOE concluded that a non-mechanistic factor applied to the dose consequences evaluations should not be needed.

The weak fuel particles are discussed as "tails" in distribution curves in Section 4.2.5.d of draft NUREG-1338 and in Section 6.3.6 of this report in a contractor letter report. These "tails" could be taken into account by relating the following aspects of the fuel particles:

- the design of the fuel particle for the expected fuel temperatures, neutron fluence, and fuel failures

- the manufacturing process and tolerances

- the quality control process

- the fuel performance tests and the statistics of the number of particles tested and the number of particles in the core

which are discussed in Section 4.2.1, of this report, on the fuel performance licensability issue for the MHTGR design.

DOE has not demonstrated that the MHTGR fuel can perform at the low failure rate needed for the MHTGR design. This is discussed in Section 4.2.4, of this report, on the containment licensability issue for the design.

The MHTGR fuel development program should address the "tails" of the weak fuel concept. However, unless the staff obtains sufficient information regarding how the four aspects of the fuel particles listed above interact to ensure that the very low fuel failure fraction is not exceeded with 95-percent confidence, a non-mechanistic factor may have to be applied to the dose consequence calculations for the MHTGR accidents. This was what the staff did when Fort St. Vrain was licensed. The TID-14844 source term was applied to the multicoated fuel particles used in the Fort St. Vrain core.

6.4.3 TER 12-3-92, "Update of Independent Analyses Section 15.4, Preapplication Safety Evaluation Report for the MHTGR, NUREG-1338"

In its letter of August 30, 1993, DOE stated that this TER did not contain Applied Technology information except for two references listed in the TER. Applied Technology information also includes the references to documents designated as containing Applied Technology information.

The contractor prepared TER 12-3-92 to recommend changes to Section 15.4, on the independent MHTGR accident analyses, of draft NUREG-1338 based on Amendments 11 through 13 to the Preliminary Safety Information Document (PSID) for the standard MHTGR ([DOE]-HTGR-86-024) and the DOE containment study (DOE-HTGR-88311). Other documents were also considered by the contractor and the entire list of documents used by the contractor appears in the abstract of the TER.

The recommended revisions to Section 15.4 of draft NUREG-1338 are in Chapters 4 and 5 of the TER. The recommended revisions do not resolve problems addressed in draft NUREG-1338 and do not significantly affect the accident analysis results presented in Section 15.4 of draft NUREG-1338, although it is stated in the TER that the peak reactor vessel temperatures are not a safety problem but are significantly higher than the DOE predictions in the PSID. The recommendations do not add a new licensability issue (i.e., a licensability issue not discussed in Section 4.2 of this report) and do not change the licensability issue on reactor vessel elevated temperature service discussed in Section 4.2.8 of this report.

The TER did point out on page 2 of the attached revisions to Section 15.4 that the control room operator action to actuate the non-safety-related shutdown cooling system (SCS) during certain unscrammed transients could raise the fuel temperature in the core. It should be pointed out that an unscrammed transient for the MHTGR design would involve the failure of two diverse, safety-related reactor protection systems which is beyond an anticipated transient without scram (ATWS) in accordance with 10 CFR Part 50.62.

The operator action is also discussed in Section 6.5.7 of this report. DOE will need to address this issue at the design approval review stage in addressing the questions raised about the role of the control room operator in Section 13.2 of draft NUREG-1338. The role of the operator is discussed in Sections 3.4.3.5 and 5.2.5 of this report.

The DOE comments on TER 12-3-92 are in the DOE letter of October 12, 1994. The first DOE comment on page 2 of the abstract for the TER referred to the "1600 °C limit for onset of fuel failure;" however, the abstract of the TER does not contain this statement. DOE made the valid point that fuel failure is a time-at-temperature effect and the fuel can be at 1600 °C (2900 °F) for an extended period of time (DOE stated hundreds of hours) before significant failures occur, and, therefore, it was more accurate to refer to the 1600 °C fuel temperature limit as a "design goal for maximum fuel temperatures."

However, as discussed in Section 5.2.2 of this report on NUREG-0111, the staff has assumed, in evaluating the HTGR source term described in NUREG-0111, that there is a threshold fuel failure temperature and that the fuel immediately fails upon reaching that temperature.

For the attachment in the TER on suggested revisions to draft NUREG-1338 Section 15.4, DOE commented on the following statements in the TER:

- The TER stated the "... 1600 °C limit for onset of fuel failure."
- The reference to "several-hundred-degree variations in peak fuel temperatures ..." did not state Fahrenheit (F) or Celsius (C), and the range of uncertainties that was considered to represent "... reasonable variations in assumed conductivities" was not given.
- The "operator interaction" of starting up the SCS during later stages of an anticipated transient without scram and loss-of-forced-cooling (ATWS-LOFC) event resulted in higher fuel temperatures in parts of the core.
- The TER stated that a main steam line break in the steam generator cavity would rupture the blowout panel between the reactor and steam generator vessel cavities when the panel is designed to blow out in only one direction, from the reactor vessel to the steam generator cavity.

For the first comment, the staff assumes that fuel failure of HTGR fuel will occur the instant the design fuel temperature limit is reached. This is discussed in Section 4.2.1 of this report and in NUREG-0111.

For the second through fourth comment, DOE was correct in its comments. The sentences referred to in the TER for the second comment were deleted from the TER. "Operator interaction" was discussed in the TER to present the fact that an operator action in the aftermath of an accident could have an adverse effect on the core, that is, to increase the fuel temperatures in the core. The analysis by the contractor did not, at the preapplication review stage, include the status of the SCS, and the point being made by the contractor was that operator action can affect the MHTGR core. DOE should address this in its response to the role of the operator discussed in Section 3.4.3.5 of this report and Section 13.2 of draft NUREG-1338. For the fourth comment (on the design of the blowout panel between the steam generator and reactor vessel cavities of the reactor building), the information about the panel was not available to the contractor when the TER was being prepared; it will be taken into account during the design approval review.

6.4.4 TER [unnumbered], "An Assessment of MHTGR Cavity Overpressure Accidents That May Impair Functionality of the Reactor Cavity Cooling System"

The contractor did not assign a number to this TER. This TER contains no Applied Technology information.

The contractor prepared this TER to assess overpressure accidents in the reactor vessel cavity containing the reactor vessel and the RCCS that may impair the functionality of the RCCS. In the TER, the contractor concluded that any primary system break greater than 0.046 m^2 (0.5 ft^2) was found to cause an increased cavity pressure in excess of the 170 kPa (10 psig) stated in Section G.4 of Appendix G of DOE-HTGR-86011 for the resulting pressure of this event and the design limit for the RCCS.

This potential damage to the RCCS will need to be considered in the resolution of the licensability issue on the RCCS in Section 4.2.6 of this report and to be addressed at the design approval review stage.

In its letter of May 10, 1995, DOE stated that when the assumptions in the TER are similar to the design description in the PSID, the results in the TER are in reasonable agreement with the results in the PSID; however, DOE also stated that some of the TER assumptions go beyond the design descriptions in the PSID. DOE was correct to state that the TER assumption that the relief valve would be in the reactor cavity was inaccurate; however, the TER results on the RCCS panels are an indication of the effect of such a failure. DOE is also correct in pointing out that the effect of a break in the reactor coolant pressure boundary is discussed in Response 6-4 of Chapter R of the PSID.

DOE concluded its review of this TER by stating that because of the extreme nature of the assumptions in the TER, the results in the TER should not be used to represent credible, design-basis conditions for the MHTGR. However, as discussed in Section 4.2.4 of this report, the staff is not prepared to state at this time that the design basis accident for the MHTGR containment should not be such a depressurization event.

6.4.5 TER 12-1-92, "Factors Affecting the Relative Failure Probabilities of the MHTGR Vessels"

This TER has no Applied Technology information.

The contractor prepared TER 12-1-92 to examine the relative values of failure probabilities of the three vessels making up the MHTGR vessel system: the reactor pressure vessel, the steam generator vessel, and the crossduct vessel. The contractor recommended that the reliability of the snubber system at the base of the steam generator should be examined, the performance of sliding supports systems used for the steam generator should be reviewed, the neutron embrittlement effects on the reactor vessel should be reviewed, the validity of the assumptions used to develop reactor vessel wall temperatures under accident conditions should be examined, and more definitive vessel failure probabilities should be determined than those given in the TER.

The contractor concluded in the TER that the three vessels should exhibit only small differences in relative failure probabilities with the highest cause for concern being the malfunction of either of the two snubbers at the base of the steam generator.

The conclusions and recommendations of the TER do not add a new licensability issue and do not change the licensability issue on reactor vessel neutron fluence embrittlement discussed in Section 4.2.7 of this report.

In its letter of May 10, 1995, DOE stated that this TER was incorrect in several instances discussing the following: the three separate "vessels," the reference to ASME Code Case N-47, the SA 508 weldment, the three vessels supported at different elevations, the insulation on the outer crossduct vessel, current design of snubbers to prevent lockup, and the time the maximum fuel temperature occurs. The staff does not disagree with the comments made by DOE on the TER; however, these comments are qualifying what the TER stated and do not have a substantive effect on the discussion of relative failure probabilities in the TER.

DOE made further comments about a further design evolution in the MHTGR as the design is finalized, that the designer does, in fact, require a materials surveillance program for the vessels as recommended in the TER, and that the transition temperature approach to vessel fracture discussed in the TER must be viewed as being largely academic and not having gained industry and NRC acceptance. These comments will be considered at the design approval review stage when a more final MHTGR design is reviewed.

6.4.6 TER 12-16-92, "Evaluation of the DOE Standard MHTGR Containment Design Alternatives"

The contractor preparing TER 12-16-92 was to summarize the containment-related work done before 1993 and to assess the DOE Containment Study, DOE-HTGR-88311. By letter dated November 28, 1989, DOE submitted the containment study in response to an NRC request dated February 28, 1989 to provide engineering studies on containment alternatives for the MHTGR design, and address the differences between the containment systems proposed by DOE for the commercial MHTGR and the MHTGR-NPR designs.

Because DOE cancelled the MHTGR-NPR program in 1992, the differences between the two MHTGR designs are no longer important. The TER concluded that the cost estimates were reasonable; however, there was an uncertainty in the reduction of the source term for each building alternative. The current problems with the MHTGR reference fuel, discussed in Section 6.5.4 of this chapter, compounded this uncertainty.

6.5 NUREG/CR Reports

6.5.1 NUREG/CR-5261, BNL-NUREG-52174, "Safety Evaluation of MHTGR Licensing Basis Accident Scenarios"

The contractor prepared this NUREG to evaluate the safety potential of the MHTGR design by searching for potential accident scenarios that might lead to (1) fuel failures due to excessive core temperatures, (2) reactor pressure vessel damage due to excessive vessel temperatures, or (3) both. The design basis accident (DBA) leading to the highest vessel temperature is a depressurized core heatup without forced cooling and with decay heat rejection to RCCS. This scenario was evaluated in NUREG/CR-5261, including parametric variations of the input parameters, and it was determined that significant safety margins exist for the MHTGR. However, high confidence levels in the core effective thermal conductivity, the reactor vessel and RCCS thermal emissivities, and the decay heat function are required for this safety margin.

Severe-accident extensions of the depressurized core heatup scenario were evaluated in the NUREG. These extensions included the complete RCCS failure, massive air ingress, core heatup without scram, collapse of the core support structure, and degraded RCCS performance. Except for no-scram scenarios extending beyond 100 hours, the NUREG stated that the fuel never reached the limiting temperature of 1600 °C (2900 °F), below which measurable fuel failures from temperatures are not expected. In some of the scenarios, the NUREG stated that excessive vessel and concrete temperatures could lead to investment losses, but these scenarios are not expected to lead to any source term beyond that from the radioactivity in the circulating coolant.

In Chapter 10 of the NUREG, the contractor proposed additional accident scenarios for the MHTGR design. DOE should consider these in developing the preliminary safety analysis report for the design approval application.

This NUREG did not consider a rapid depressurization of the reactor coolant pressure boundary at high fuel temperatures, which is discussed in Section 3.4.3.6 of this report.

6.5.2 NUREG/CR-5514, ORNL/TM-11451, "Modelling and Performance of the MHTGR Reactor Cavity Cooling System"

The RCCS is the only safety-grade decay heat removal system for the core in the reactor pressure vessel and it is discussed in Section 4.2.6 of this report. This RCCS study was to model, in a computer, the RCCS independent of the reactor vessel and investigate the dynamic thermal performance of the system. Although the computer model predicted reasonable thermal performance of the RCCS, the NUREG stated the following about the heat transfer from the reactor vessel to the RCCS panels:

- The heat transfer rate is very dependent on the surface emissivities of the reactor vessel and the panels, and these emissivities should be checked periodically.

- The heat transfer rate is reduced, by up to 10 percent, by the presence of steam between the vessel and panels.

The RCCS as a completely passive heat removal system is a licensability issue for the MHTGR design and discussed in Section 4.2.6 of this report. The RCCS instrumentation discussed in Section 4.2.6 may provide the periodic checking of the emissivities discussed above.

6.5.3 NUREG/CR-5647, ORNL/TM-11685, "Fission Product Plateout and Liftoff in the MHTGR Primary System: A Review"

The contractor who performed this study was to evaluate the technical status of modeling the deposition mechanisms for plateout and the liftoff release of fission products in the primary reactor coolant system of the MHTGR. Consideration of liftoff in this NUREG was restricted to dry depressurization events with no involvement of the steam system. A major part of the review dealt with the expected dust levels, types, and transport.

The contractor concluded in the NUREG that the deposition mechanisms controlling the distribution of fission-product material in the reactor coolant system, and hence also controlling the degree of liftoff, depend strongly on the chemical nature of the individual elements. Therefore, both plateout and liftoff models should reflect these unique chemical and physical properties. The contractor concluded that a sufficient technical basis for plateout and liftoff modeling does not exist or has not been applied.

DOE should address this at the design approval review stage in the discussion of fission-product transport codes, as discussed in Section 4.2.2 of this report.

6.5.4 NUREG/CR-5810, ORNL/TM-12014, "Evaluation of MHTGR Fuel Reliability"

The contractor that prepared this study was to review the reliability of the MHTGR fuel to behave according to model predictions in normal service and under postulated accident conditions. Fuel manufacture, failure mechanisms, design requirements, and quality control are discussed. The "weak fuel" concept introduced by the staff in Section 4.2 of draft NUREG-1338 is also discussed in this report.

The contractor concluded that the "weak fuel" penalty should be continued for the MHTGR design without a sealed containment (i.e., the low-leakage light-water reactor containment) because the following have not been met:

- Results from a testing program of convincing scope on the MHTGR reference fuel, manufactured using prototypical methods with quality control, which demonstrate fuel performance in accordance with model predictions for both normal operation and accident conditions.

- Good comprehension of fuel failure mechanisms to provide unambiguous interpretation of capsule test and HTGR operational data.

- Identification of the range of possible fuel manufacturing defects and their mechanistic relation to fuel failures.

- Adoption of a carefully considered quality control program for fuel manufacture, based on identification and rejection of the most significant manufacturing defects.

This is the same relationship among fuel design, manufacture, and testing discussed in Section 4.2.1 of this report.

The contractor does not identify any new licensability issue and does not change the licensability issue of fuel performance discussed in Section 4.2.1 of this report.

6.5.5 NUREG/CR-5947, ORNL/TM-12237, "Magnitude and Reactivity Consequences of Moisture Ingress into the MHTGR Core"

The work described in NUREG/CR-5947 is an analytical methodology to quantify the pressure and reactivity consequences for the MHTGR core of steam generator tube rupture and other moisture-ingress-related events. Neutronic and thermohydraulic processes were coupled with reactivity feedback, and safety and control system responses. NUREG/CR-5987 stated that in ATWS events, the reactor protection system is partially defeated. An unscrammed transient for the MHTGR design would involve failures in two diverse, safety-related reactor protection systems which is beyond an ATWS in accordance with 10 CFR Part 50.62.

The rate and magnitude of water ingress were found to be dominated by such major system features as break size compared with safety valve capacity and reliability, and less sensitive to such factors as heat transfer and reactivity coefficients. The results reported in the NUREG indicated that ingress transients progress more slowly than previously predicted by bounding analyses, with milder power overshoots and more time for operator or automatic corrective actions. The comparatively slow buildup of moisture in the core allows the core power to be limited by the fuel temperature rise and reactivity feedback. There appears to be sufficient time for operator action if automatic trips fail.

The contractor does not identify any new licensability issue, but does support DOE's position on operator action discussed in Section 5.2.5 of this report.

6.5.6 NUREG/CR-5922, ORNL/TM-12179, "Modular High Temperature Gas-Cooled Reactor Short-Term Thermal Response to Flow and Reactivity Transients"

The contractor analyzed the short-term thermal response of the MHTGR for a range of flow and reactivity transients. The purpose was to compare the results from the MHTGR designers in the PSID ([DOE]-HTGR-86-024) and search for conditions that could lead to more severe transients than previously identified.

The contractor's overall conclusion in the NUREG report was that, for the events analyzed, the inherent features of the MHTGR have the potential for providing a high degree of safety. For the events analyzed in NUREG/CR-5922, the results agreed with those presented in the PSID except for the moisture-ingress events resulting from the rupture of one steam generator tube. The contractor stated that the reactivity transient for this event should be less severe than that reported in Chapter 15 of the PSID. The contractor also stated that no conditions were identified that could lead to transients that are significantly more severe than previously identified. There are also specific conclusions about the transients studied.

The contractor did not identify any new licensability issue for the MHTGR design.

6.5.7 NUREG/CR-5983, BNL-NUREG-52356, "Safety Aspects of Forced Flow Cooldown Transients in Modular High Temperature Gas-Cooled Reactors"

The contractor considered potential accident transients in the MHTGR with forced convection cooldown and the use of either one of the two available forced-flow heat-transport systems: the main circulator at the top of the steam generator vessel used for normal forced flow or the SCS at the bottom of the reactor pressure vessel which is used when the main circulator is not available.

The NUREG concluded that if the SCS were started during a pressurized conduction cooldown transient, it would not cause excessive temperatures in the fuel or the metallic components of the core support structure; however, if it were used during a low-probability core-heatup transient in a non-scrammed reactor, an already serious accident would be aggravated by increasing peak fuel temperatures above 1600 °C (2900 °F). This potential aggravation of an accident by an operator is the same situation discussed in Section 6.4.2 of this chapter and DOE will need to address it at the design approval review stage when it addresses the questions raised by the staff about the role of the control room operator in Section 13.2 of draft NUREG-1338. The role of the operator is discussed in Section 3.4.3.5 of this report.

The contractor does not identify any licensability issue for the MHTGR design.

6.5.8 NUREG/CR-6358, "Assessment of United States Industry Design Codes and Standards for Application to Evolutionary and Advanced Nuclear Power Reactors"

This NUREG provides an assessment of United States design codes and standards for evolutionary and advanced nuclear power plants, including the MHTGR design. The objective of the NUREG was to determine (1) the necessary changes to industry codes and standards to have them applicable to the design and construction of the evolutionary and advanced commercial nuclear power reactors and (2) the unique attributes and features associated with the seismic Category I, safety class structures.

The changes stated to address deficiencies in industry codes and standards were listed and described in Table 3.5.2.1 of the draft NUREG.

In Section 3.3.5 of the draft NUREG, the unique attributes associated with the seismic Category I, safety class structures were stated to be the following: the safety classification approach, the use of American Concrete Institute Standard ACI-349 for design of the confinement/containment structure, elimination of the operating-basis earthquake from the design basis, the high concrete temperatures, the high distribution support temperatures, and the deeply soil-embedded reactor and other buildings.

6.6 Conclusions

In this chapter, 20 reports from NRC contractors on the MHTGR design that were completed after draft NUREG-1338 was issued were reviewed for information on the MHTGR design. Where the original reports contained Applied Technology information, the reports in the appendices were modified to remove the Applied Technology information identified by DOE. The Applied Technology designation is a licensability issue for the MHTGR and is discussed in Section 4.2.9 of this report.

In the discussion on TER 2-2-93 in Section 6.4.1 of this chapter, the staff identified a licensability issue for the MHTGR design with the computer codes used to calculate the fission-product transport from the fuel. This is the only licensability issue identified in the contractor reports; it is discussed in Section 4.2.2 of this report. Because of the extensive areas of the TER that were identified as Applied Technology, this TER was not issued as a modified non-Applied Technology report and is not in Appendix J of this report.

The other contractor reports discussed in this chapter do not change any licensability issues for the MHTGR discussed in Chapter 5 of this report. The conclusions in the contractor reports on the consequences of accidents are not significant departures from what is presented in Chapter 15 of draft NUREG-1338.

Some of the contractor reports concluded that additional information was needed through the TDP for the MHTGR design. These are discussed in Sections 6.3.5, 6.3.6, and 6.5.3 of this chapter.

The letter report discussed in Section 6.3.6 of this chapter proposed a new model for fuel performance and supported the "weak fuel" concept proposed by the staff in Section 4.2.5, Item D, of draft NUREG-1338. The letter report also showed gas release fractions for German coated fuel that demonstrates fuel failures up to 2100 °C (3800 °F) that may be sufficiently low for the proposed MHTGR high-leakage containment. In the letter report, discussed in Section 6.3.5, the contractor stated that it appeared that fuel integrity for normal and accident conditions up to 1600 °C (2900 °F) can be assured.

DOE should address the contractor's reports discussed in Sections 6.3.1, 6.3.5, 6.3.6, 6.4.1 through 6.4.4, 6.5.1, and 6.5.7 of this chapter at the design approval review stage. These contractor reports are reproduced in Appendix J of this report except for TER 2-2-93 discussed in Section 6.4.1. As explained above, this TER has been designated as containing Applied Technology information by DOE and cannot be made available to individuals representing foreign interests. If Applied Technology information was identified in the other contractor reports, the copy in Appendix J of this report was reproduced without the Applied Technology information.

The contractor reports discussed in Sections 6.4.3 and 6.5.7 addressed the role of the operator where operator action increased fuel temperatures during certain MHTGR accidents.

7. TECHNOLOGY DEVELOPMENT PLAN

7.1 Introduction

One objective of the preapplication review was to assess the adequacy of the applicant's research and development program for the advanced reactor design. This chapter discusses the adequacy of the Department of Energy's (DOE's) research and development program for the MHTGR design based on the staff's conclusions in draft NUREG-1338, on the initial preapplication review of the MHTGR design, and in contractor reports completed since draft NUREG-1338 was issued.

By its letter of January 15, 1987, DOE submitted the Regulatory Technology Development Plan (RTDP) for the MHTGR design (DOE-HTGR-86-064). The RTDP described the research and development programs that would generate the technical information related to radionuclide control and retention for the MHTGR design. The RTDP programs were those judged by DOE to be needed to complete the data base that assures the MHTGR design will meet the broad (top-level) regulatory criteria, described in Section 1.5 of this report, and will perform as described in the Preliminary Information Safety Document (PSID) for the design ([DOE]-HTGR-86-024). The RTDP describes the technology development for the following technical areas:

- fuel and fission product transport
- graphite
- metals
- control materials
- system and component tests

The RTDP is not the entire research and development program for the MHTGR design; it is only that part concerned with safety-grade equipment and systems. There is also research and development for other technical areas, such as heat exchangers, circulators, and fuel handling equipment. These latter areas are not included in the RTDP.

Since the RTDP was submitted, DOE has also submitted, in two letters dated July 16, 1993, the following DOE reports on technology needs for the MHTGR design:

- DOE-HTGR-90352, "Integrated Technology Plan To Support MHTGR Source Term and Containment Concept," Revision 0, April 1993 (Applied Technology)
- DOE-HTGR-90348, "Reactor Physics Development Plan," Revision 0, December 1992 (Applied Technology)
- DOE-HTGR-90357, "450 MW(t) MHTGR Reactor Metals Development Plan," Revision 0, April 1993 (Applied Technology)
- DOE-HTGR-90358, "450 MW(t) MHTGR Reactor Graphite and Ceramics Development Plan," Revision 0, June 1993 (Applied Technology)

These four DOE reports are the only technology development plans for the MHTGR design that have been submitted since the RTDP was provided in 1987 and draft NUREG-1338 was issued. These reports were reviewed in the contractor report discussed in Section 7.3.2 of this chapter.

The staff discussed the RTDP in draft NUREG-1338 and some of the contractor reports reviewed in Chapter 6 of this report discussed the technology needs for the MHTGR design. The conclusions on the MHTGR technology needs in draft NUREG-1338 and in the contractor reports are discussed in Sections 7.2 and 7.3, respectively, in this chapter.

7.2 Draft NUREG-1338

Draft NUREG-1338 documents the preapplication review of the MHTGR design from 1986 through 1989. A discussion on conclusions by the staff in draft NUREG-1338 that remain valid for this report are in Sections 3.4.2 and 3.4.3 of this report. In each major section of most of the chapters in draft NUREG-1338, a discussion on the research and development plans identified in the RTDP for that technical area are discussed in the fourth subsection (that is, Section X.X.4). These subsections have been reviewed and the following conclusions stated therein remain valid.

7.2.1 MHTGR Fuel

In Sections 4.2.4 and 11.1.4 of draft NUREG-1338, the staff discussed the research and development identified for the MHTGR fuel in the RTDP. The performance of the fuel for the MHTGR design is the most important licensability issue for the design and is discussed in Section 4.2.1 of this report.

The staff listed the 19 technology development needs (TDNs), or separate research plans, for the MHTGR fuel which are described in Section 6 of the RTDP. It concluded that the adequacy of the fuel development in the RTDP is an essential requirement for the staff acceptance of the MHTGR concept of an unconventional, high-leakage containment if the staff, as proposed by DOE, is to forgo some of the traditional requirements for defense in depth and use a mechanistic MHTGR source term. The staff also concluded that the TDNs did not demonstrate that a proven correlation existed between the fuel design and all the possible and postulated conditions that the fuel may be exposed to in normal and accident operations. The TDN plans must demonstrate the correlations between the MHTGR fuel design and the response to postulated accidents. These statements in draft NUREG-1338 remain valid.

The staff further stated at the conclusion of draft NUREG-1338 Section 4.2.4 that the areas of fuel manufacture, quality control, and statistical uncertainty — the design requirement that only a very small number of particles in the core (i.e., 6.0×10^{-4}) can fail in an accident — need to be addressed in the technology development plans for the MHTGR fuel. The need for DOE to relate fuel manufacture and quality control to the performance of the fuel is discussed in Section 4.2.1 of this report.

The new integrated technology plans on the MHTGR source term and containment, listed in Section 7.1 of this chapter and submitted by DOE in 1993 (DOE letter dated July 16, 1993), do not address the areas of fuel manufacture, quality control, and statistical uncertainty. This DOE report recommends alternative technology and design options relating the source term to the containment leak rate that was discussed in DOE-HTGR-90321.

At the design approval review stage, DOE needs to address the staff's conclusions in draft NUREG-1338 discussed above and the research and development needed for the MHTGR fuel, and how the research and development programs will demonstrate the level of fuel performance required during normal plant operation and accidents.

7.2.2 Other Technical Areas in Draft NUREG-1338

In Section 4.3.4 of draft NUREG-1338, the staff concluded that a reactor physics development plan should be submitted to NRC and included in the RTDP. DOE submitted this plan in its letter of July 16, 1993, and it is listed in Section 7.1 of this chapter. This reactor physics development plan does not involve a licensability issue and will be reviewed at the design approval review stage.

In Section 5.2.4 of draft NUREG-1338, the staff discussed TDN 8-2 on the properties of the steel in the reactor vessel (i.e., material SA533B) at elevated temperatures and concluded that it would review this TDN at a later review stage after the American Society of Mechanical Engineers (ASME) Code committee had reviewed the data. The elevated temperature service of the reactor pressure vessel involves a code case inquiry which has been approved for the ASME Code, is a licensability issue for the MHTGR, and is discussed in Section 4.2.8 of this report.

In Section 5.5.4 of draft NUREG-1338, the staff discussed the research needed for the reactor cavity cooling system (RCCS), which is the only safety-grade decay heat removal system for the MHTGR. Since draft NUREG-1338 was issued, the staff has had a contractor review the performance of the RCCS (NUREG/CR-5514). The contractor's report is discussed in Section 6.5.2 of this report. The contractor's report does not change the staff's conclusions in draft NUREG-1338 about the following research needs for the RCCS:

- an integral test to demonstrate effectiveness and reliability of the RCCS
- additional data on the thermal conductivity of graphite
- establishment of the resistance of the RCCS to large seismic events and other potential external-event failure modes
- understanding long-term RCCS failure modes to aid in the development of an inservice inspection program and to address aging

The RCCS instrumentation discussed in Section 4.2.6 of this report may address the development of inservice inspection for and the aging (i.e., degradation in service) of the RCCS. DOE should address the instrumentation and technology development needed for the RCCS in its design approval application.

The staff did not address in draft NUREG-1338 whether any technology development was needed to demonstrate the helium-water heat transfer through the MHTGR steam generator because the staff did not review for the NUREG the areas where experience with earlier high temperature gas-cooled reactors was considered satisfactory (in this case, the experience with the steam generator at Fort St. Vrain); however, the MHTGR steam generator and its material are not the same as that at Fort St. Vrain, and DOE will need to demonstrate the required heat transfer performance of the MHTGR steam generator at the design approval review stage. The fact that the staff does not address a technology development area in draft NUREG-1338 and in this report does not mean that DOE will not have to demonstrate the required technology for that area of the MHTGR design.

7.2.3 Conclusions

The significant staff comments on the RTDP for the MHTGR design in draft NUREG-1338 were on the MHTGR fuel and RCCS. Consistent with the discussions on the licensability issues of the fuel and source term, DOE will need to address the technology needs for the fuel design. Addressing these technology needs must involve addressing the relationship of the performance of the fuel, its manufacture, the quality control, and the statistical concerns of testing only a few particles compared to the millions of particles contained in the core. DOE needs to update the RTDP at the design approval review stage and address the MHTGR fuel technology needs, when it addresses the fuel performance and source term licensability issues of Sections 4.2.1 and 4.2.3 of this report.

For the RCCS, DOE should address the research needs identified by the staff for this safety system and whether a prototype should be used to demonstrate the performance and reliability of the RCCS. The MHTGR prototype can also demonstrate the instrumentation for monitoring the RCCS and discussions on a prototype are presented in Sections 3.4.3 and 5.3.12 of this report. The RCCS is a licensability issue for the MHTGR design; it is discussed in Section 4.2.8 of this report.

7.3 Contractor Reports

The technology development program for the MHTGR design was discussed in the following contractor reports:

- Brookhaven National Laboratory, Letter Report L-2213 11/93, "Initial Assessment of the Data Base for Modelling of MHTGRs," November 1993 (Applied Technology)
- University of Tennessee, letter report, "Final Report, Research on Fuel Performance in Modular High-Temperature Gas-Cooled Reactors," letter dated October 7, 1994, from Paul Kasten, Research Professor, University of Tennessee

- Oak Ridge National Laboratory, TER 2-10-93, "Review and Evaluation of Recent Publications Bearing on the Fuels Sections of the Draft PSER," February 10, 1993 (Applied Technology)

These contractor reports are discussed in detail in Sections 6.3.5, 6.3.6, and 6.4.1, respectively, of this report; however, the conclusions concerning the RTDP for the MHTGR are discussed in the sections that follow.

7.3.1 TER 2-10-93, "Review and Evaluation of Recent Publications Bearing on the Fuels Sections of the Draft PSER"

In Sections 3.6 and 3.7 of the TER, the contractor stated that the RTDP needed the following:

- to be updated
- to consider fuel failure statistics
- to include an evaluation of the fuel manufacture and quality control processes
- to include the validation of fuel performance models

These statements in the TER are consistent with what the staff stated in Section 4.2.4 of draft NUREG-1338, which is also discussed in Section 7.2.1 of this chapter.

The contractor stated that DOE does not identify the documentation requirements for evaluation models and standards for acceptability in the RTDP, as does Appendix K of 10 CFR Part 50 for light-water-reactor fuel. The RTDP only made reference to the broad (top-level) regulatory criteria and did not address whether the designer's satisfaction with the technology to meet these top-level criteria also constituted compliance with all the appropriate regulations. Also, the RTDP did not discuss if the required MHTGR goal accuracies are consistent with the experimental uncertainties.

The validated fuel performance models need to be included in the computer programs for fission-product transport from the fuel to the containment which were discussed in Section 4.2.2 of this report.

The final conclusion by the contractor was that the RTDP plans for the MHTGR fuel are narrowly focused plans based on the belief that it is only necessary to test the fuel to show its proven performance. The TER stated that the RTDP was not a broad-based technology development plan. The results of the recent HRB-21 test of the MHTGR fuel at Oak Ridge National Laboratories did not demonstrate the proposed fuel performance needed for the MHTGR design. No further results have been submitted to NRC.

DOE should address the conclusions of this TER about the RTDP and the discussion in Section 7.2.1 of this chapter at the design approval review stage.

7.3.2 Letter Report L-2213 11/93, "Initial Assessment of the Data Base for Modelling of Modular High Temperature Gas-Cooled Reactors"

The contractor reviewed the currently available data base for modeling the MHTGR transients. The data base reviewed included the four DOE technology development reports that are listed in Section 7.1 of this chapter. The conclusions in this letter report on the technology needs for the MHTGR identified by DOE are discussed in detail in Section 6.3.5 of this report. In this section, only the technical data needed for the MHTGR design are discussed.

The technical areas identified by the contractor as having missing or partial technical data, or needing model development are the following:

- reactor heat transfer and fluid flow
- heat transfer to the RCCS
- graphite and fuel chemical reactions with helium coolant impurities
- upper plenum natural circulation

Although it was concluded in the letter report that data identified in the RTDP for fission-product transport were sufficient, this conclusion was based on the author's assumption that experience has indicated DOE's planned efforts will be sufficient. The discussion in Sections 7.2.1 and 7.3.1 of this chapter on fission-product transport data indicates that this assumption may be incorrect at this time. DOE should discuss the technology development needed for the MHTGR fuel, and should address the discussions in Sections 7.2.1 and 7.3.1 of this chapter, in its design approval application.

7.3.3 Letter Report, University of Tennessee, "Final Report, Research on Fuel Performance in Modular High-Temperature Gas-Cooled Reactors"

In this letter report, the contractor stated that the following fuel research and development was needed to validate the proposed new fuel model in the letter report:

- Determine the nature of silicon carbide coating defects that occur during manufacture of the coated fuel particles.

- Determine the outer pyrolytic-carbon layer strength distribution in a large batch of coated fuel particles and the effect of neutron fluence on that strength.

- Perform irradiation testing and subsequent heating of unfueled particles to study the effects of natural contamination in the matrix graphite.

The contractor stated that the RTDP did not contain technology plans that included these identified needs. Although this is to be expected because the RTDP was not developed to validate the new fuel model proposed in this letter report, it would seem that such data would be needed to validate the models being used by DOE. The fact that these data are not being collected may be the result of the RTDP being too narrowly focused, as discussed in Section 7.3.1 above.

The contractor also stated that substantially more research and development was needed for the MHTGR fuel in order to develop and validate performance models for the fuel. The most important factor was said to be the excessive breakage of the outer-pyrolytic-carbon layers because of the interactions between these layers and the matrix graphite of a fuel compact. The MHTGR fuel was also reported to have different coating properties than the German fuel, which has shown the low failure rate proposed and needed for the MHTGR design. For example, the strength of the silicon carbide layer in the MHTGR fuel was said to have "weaker tails" than the German fuel.

At the design approval review stage, DOE needs to address the research needs raised in this letter report.

7.3.4 Conclusions

The significant comments on the RTDP in contractor reports discussed above were on the MHTGR fuel and the following technical areas identified by the contractors as needing further data:

- reactor heat transfer and fluid flow
- heat transfer to the RCCS
- graphite and fuel chemical reactions with coolant
- upper plenum natural circulation
- fuel performance models

Contractors stated that the RTDP for the MHTGR needs to be updated with consideration of fuel failure statistics (i.e., the statistics of the few fuel particles tested in comparison to the many particles in the core), defects occurring in the fuel manufacturing process, the quality control for the fuel manufacturing process, and the validation of fuel performance models.

Contractors identified more technical data needs for the MHTGR design than were addressed in the DOE reports on the RTDP (listed in Section 7.1 of this chapter) that were submitted after draft NUREG-1338 was issued. The staff, therefore, concludes that additional information on the RTDP for the MHTGR will be needed for the design approval review of the design.

DOE should address all of the TDPs for the MHTGR and the RTDP issues raised in the contractor reports discussed in Section 7.3 of this chapter in its design approval application.

8. CONCLUSIONS

8.1 Introduction

This report documents the preapplication review by the staff of the Nuclear Regulatory Commission (NRC) of the standard Modular High Temperature Gas-Cooled Reactor (MHTGR) design. The design was submitted by the Department of Energy (DOE) and described in the Preliminary Safety Information Document (PSID) for the standard MHTGR ([DOE]-HTGR-86-024). The preapplication review stage is an early interaction with NRC for advanced reactor designs preceding the time when the applicant will submit its design for preliminary design approval (PDA), final design approval (FDA), or standard plant design certification under 10 CFR Part 52. The NRC requested this early interaction between the designer and the NRC staff in the Commission's Advanced Reactor Policy Statement (51 FR 24643).

The preapplication reviews of advanced reactor designs are conducted for the staff and the public to understand the design, and for the designers to learn about the licensability issues of the design before applying for NRC staff review under 10 CFR Part 52. The preapplication review is not to approve the design because it is completed while the design is still being developed and lacks the final details that would be required in an application for design approval (i.e., a PDA, FDA, or design certification application under 10 CFR Part 52).

The initial phase of this preapplication review for the MHTGR design was conducted from 1986 through 1989 by the NRC Office of Nuclear Regulatory Research (RES) and documented in the draft Preapplication Safety Evaluation Report (PSER), draft NUREG-1338. After 1989, RES continued technical assistance on the MHTGR, which is discussed in Chapter 6 of this report. Since 1991, the NRC Office of Nuclear Reactor Regulation (NRR) has been conducting the last phase of the preapplication review; that phase is being documented in this final PSER.

As discussed in Section 1.2 of this report, this final PSER on the MHTGR design provides the following information stated in NUREG-1226: (1) the major safety and policy issues associated with the design, (2) guidance on the licensing criteria applicable to the design, (3) potential impediments to approving the advanced reactor design, and (4) an assessment of the adequacy of the applicant's research and development program. These are discussed in Chapters 3, 4, 5, 7, and this chapter (Section 8.2) of this report. Chapter 6 discusses the contractor reports either not completed for the NRC staff before draft NUREG-1338 was issued or not discussed in draft NUREG-1338.

This report contains the licensability and policy issues, and the technology development plans for the MHTGR design; it does not review the MHTGR plant systems. This report discusses draft NUREG-1338 and builds upon the conclusions of the staff in that document to discuss the licensability and policy issues for the design. This review for the MHTGR is, therefore, different from the review done by the staff for the liquid metal reactor PRISM and documented in the staff's Preapplication Safety Evaluation Report (PSER) for PRISM, NUREG-1368. The review approach and review criteria applied to the MHTGR design in this report are also different from that applied to license conventional light-water reactors (LWRs) because this is a preapplication review preceding a 10 CFR Part 52 application.

This report does not constitute an approval of the MHTGR design and is not intended.to be a basis for NRC to approve any part of the design under 10 CFR Part 52. The object of this review was to provide guidance on licensability early in the design process to aid the designer in developing a design approval application. The Commission can only make a determination on the acceptability of the design after the design has been submitted for design approval under 10 CFR Part 52.

Therefore, the conclusions of the staff in this report, particularly in Chapter 4 on licensability issues and Chapter 5 on policy issues, are not intended to be complete discussions on these issues or to close out any reviews of the staff during a design approval review of the MHTGR in the future. Also, other licensability issues may appear during the design approval review. The expectation of the staff is that these new issues should be less important than those issues discussed in Chapter 4 of this report. The staff's conclusions are to provide insights on licensability problems of the design which the designer is expected to address in its design approval application.

8.2 Responses to DOE Questions

As discussed in Section 1.9 of this report, DOE asked the NRC staff to respond to seven questions listed in Section 1.1.5 of the PSID. The questions and the NRC responses are given below:

- Is the standard MHTGR design licensable?

  The answer is a qualified yes because there are licensability issues, discussed in Chapter 4 of this report, for the design that may fundamentally change the design during licensing. This is discussed in the conclusion sections of this report in Sections 8.3 and 8.4 of this chapter (below).

- Are the interfaces between the standard Nuclear Island (NI) and the Energy Conversion Area (ECA), and the site appropriately identified and characterized?

  The staff did not review the interfaces between the NI, ECA, and the site during the preapplication review.

  The MHTGR plant arrangement was evaluated in Chapter 6 of draft NUREG-1338. The staff stated in draft NUREG-1338 Section 6.1.3 that the overall plant layout and building designs were not reviewed, with the exception of the location of the control room and protection of reactor operators where the staff identified as a safety issue the fact that the control room was not located in the NI. This is discussed in Section

3.4.2.4 of this report. Therefore, the staff will complete its review of the MHTGR plant arrangement at the design approval review stage.

The site data in Chapter 2 and Section 3.7 of the PSID were evaluated in Chapter 2 of draft NUREG-1338. The staff concluded that the treatment of standard site characteristics in the PSID was consistent with NRC regulatory guidance.

- Are the top-level regulatory criteria acceptable and can they remain valid through final design approval?

The broad (top-level) regulatory criteria are listed in Section 1.5 of this report. These criteria are valid because they must be met for a nuclear power plant to be licensed and, in their application to the MHTGR design, the staff has concluded, in Section 2.7 of this report, that the MHTGR provides several safety enhancements and should produce at least the same level of protection as the current-generation LWRs. However, these criteria are not complete because, as discussed in Chapter 2 of this report, there are other criteria that also must be met for a license to be issued. These other criteria, which may have to be modified for a non-LWR, are the (1) LWR general design criteria (10 CFR Part 50 Appendix A) and NRC-approved industry codes and standards, and (2) criteria reflected in the Standard Review Plan (NUREG-0800) and the Commission Severe Accident Policy (50 FR 32138). These other criteria define the safety margins for the new designs and provide the assurance that the top-level regulatory criteria have been met.

Therefore, as discussed in Section 2.8 of this report, the staff does not agree with DOE that the use of the top-level regulatory criteria is sufficient assurance that the MHTGR design provides the same degree of protection to the public and the environment that is required for current-generation LWRs. DOE needs to address this issue at the design approval review stage.

The staff applied the General Design Criteria (GDCs) in 10 CFR Part 50 for LWRs to the non-LWR PRISM advanced reactor design in Section 3.2 of NUREG-1368 to develop plant-specific GDCs for the PRISM design. This NUREG-1368 section shows how the Part 50 GDCs could be applied to the MHTGR to develop plant-specific GDCs for the design. In NUREG-1368 Section 3.2, the staff concluded that only a few general design criterion were not applicable, directly or with revisions, to the PRISM design.

DOE discussed the applicability of the GDCs to the MHTGR design in Comment G.3-1 of PSID Chapter R and concluded that many of the GDC, including GDC 50 through 57 on containment design, did not apply to the MHTGR because of DOE's positions on the top-level criteria, containment design and isolation, protection provided by the fuel, and safety classification. Because of time and resource limitations, the staff did not evaluate DOE's positions on each general design criterion, but discussed the top-level criteria, containment, fuel, and safety classification for the MHTGR.

The GDCs for the LWRs were originally developed not to implement regulations, but to codify certain general design requirements that came out of the early experience with LWRs and which were considered necessary to prevent design problems which could cause unsafe plant operation. The regulations were written to solve specific reactor design problems. Therefore, applying only top-level criteria to designing a plant may miss general design requirements that could be important to the safe operation of the plant.

- Is the methodology for proceeding from the top-level regulatory criteria through risk assessments and other safety analyses to the licensing basis acceptable and can it remain through final design approval?

  No. Other criteria need to be included along with the broad (top-level) regulatory criteria for the MHTGR design, as discussed in the previous response.

- Is the approach for emergency planning acceptable?

  The staff has not made a determination. The Commission concluded that it was premature to reach a conclusion on emergency planning for advanced reactors and did not reach a decision on the proposed approach for emergency planning for the MHTGR. This is discussed in Section 5.2.4 of this report, where current staff endeavors involving possible simplification of emergency planning are addressed. The staff is to provide regulatory direction on this issue at or before the start of the design approval review stage and consider the proposed changes to emergency planning proposed by the applicants for the advanced LWRs.

- Is the Regulatory Technology Development Plan (RTDP) adequate for final design approval?

  No. As stated in Section 7.4 of this report, there are technical areas of the MHTGR design that have been identified as needing additional data and the conclusion is that the RTDP may be too narrowly focused. Therefore, the staff concludes that the RTDP needs enhancement for final design approval of the MHTGR design.

- Is the proposed application procedure in the licensing plan of HTGR-85-001 acceptable?

  The licensing plan for the MHTGR, in HTGR-85-001, was to identify the necessary activities needed for NRC to issue an FDA and publish a rule for certification of the standard MHTGR design. It outlined a preapplication review, a preliminary safety analysis report (PSAR) review leading to a preliminary design approval (PDA), a final safety analysis report (FSAR) review leading to a final design approval (FDA), a demonstration plant, and rulemaking. The demonstration of the plant would be completed before the FDA is issued.

  This licensing plan is consistent with 10 CFR Part 52 and is an acceptable plan, although it should be pointed out that it is not necessary to go through the review for a PDA. Although the PDA requires less information on the design, the PDA is not a prerequisite for the FDA or rulemaking. The FDA is a prerequisite for rulemaking.

8.3 MHTGR Licensability Issues

As discussed in Section 2.7 of this report, licensability issues are those concerns raised by the staff about a design that involve either (1) issues that are significant departures from past acceptance licensing practices of the NRC and which neither the Commission nor the staff has approved the departure, and (2) issues whose resolution may result in a fundamental change to the proposed design. For the MHTGR design, the following nine licensability issues are discussed in Sections 4.2.1 through 4.2.9, respectively, of this report:

- fuel performance
- fission product transport computer codes
- source term
- unconventional containment
- safety classification and regulatory treatment of non-safety-grade systems
- completely passive system for ultimate heat sink
- reactor vessel neutron fluence embrittlement
- reactor vessel elevated-temperature service
- Applied Technology designation

The two most important issues are the fuel performance and the Applied Technology designation, which are discussed below. The issues of the fission-product transport computer codes, source term, unconventional containment, and safety classification are related to the issue of fuel performance. If the proposed fuel performance can be demonstrated, the other four issues should be able to be satisfactorily addressed for the MHTGR design.

The completely passive system for the ultimate heat sink, the reactor cavity cooling system (RCCS), is the only safety-grade system in the MHTGR for decay heat removal. This issue is expected to be resolved through the demonstration of the reliability of the RCCS and the regulatory treatment of non-safety-systems (RTNSS) which support the RCCS, discussed in Sections 4.2.6 and 5.2.6 of this report, and perhaps demonstrated in an MHTGR prototype test for a wide range of conditions.

Although questions of neutron embrittlement of the reactor vessel still must be addressed, the Arbeitsgemeinschaft Versuchs Reaktor, Peach Bottom, and Dragon were HTGRs that operated satisfactorily with steel reactor vessels, and it is expected that this issue will be satisfactorily addressed for the MHTGR.

Although the code case inquiry for reactor vessel elevated-temperature service has been approved by the ASME Code main committee, the staff has not reviewed the code inquiry for the MHTGR and DOE has not addressed the frequency of Service Level C and D events for the MHTGR reactor vessel.

For the issues of fuel performance and Applied Technology designation, there is a question about how soon these issues could be satisfactorily resolved. These two issues are briefly discussed below.

8.3.1 MHTGR Fuel

The performance required from the MHTGR fuel and the lack of demonstration of this performance is discussed in Section 4.2.1 of this report. The proposed MHTGR fuel performance (i.e., very few fuel failures during normal operation and accidents) confines essentially all the fission products in the fuel particles, and could justify, using the Commission guidance discussed in Section 5.2 of this report, the proposed high-leakage containment feature of the design and the design's inclusion of few systems classified as safety related. The Commission has accepted the principle that onsite and offsite dose consequences can be used to determine the acceptable leakage from the containment. However, no reactor in the United States has been licensed with an HTGR source term, and DOE has not (1) demonstrated the proposed fuel performance and (2) explained the relationship among the fuel design, manufacture, quality assurance, and performance for the MHTGR.

Because only a very small fraction of the fuel in the core can be allowed to fail in accidents and because of the small sample of fuel particles tested compared to the number that would be in a core, there is a large statistical uncertainty in applying the results of the fuel performance tests to what will happen in the overall core. Therefore, DOE needs to consider the volume of the core that is above the MHTGR fuel failure design temperature limit during any accident. It is only the fuel in this region that can fail. The number of particles subject to failure in that part of the core may be sufficiently smaller than the total number in the core that the statistical uncertainty discussed previously becomes moot. DOE needs to address this at the design approval review stage.

Although the German fuel test data indicate that the proposed fuel performance should be able to be met, the staff does not know when this may be demonstrated for the MHTGR fuel. It is expected that the fuel performance that can be demonstrated will be used to determine the leak rate of the containment consistent with the acceptable dose consequences, as discussed by DOE in DOE-HTGR-90321.

8.3.2 Applied Technology Designation

The Applied Technology designation and the problems with its restriction on the disclosure of information on the MHTGR design are discussed in Sections 1.8 and 4.2.9 of this report.

DOE has designated a significant amount of the information it has submitted on the MHTGR as "Applied Technology." This designation does not allow distribution of such information to third parties representing foreign interests, foreign governments, foreign companies, and foreign subsidiaries or foreign divisions of U.S. companies without written permission from DOE. Because individuals representing foreign interests could take information from a public document room, NRC has not placed "Applied Technology" information in the NRC Public Document Room during the preapplication review of the MHTGR.

This nondisclosure of MHTGR information has not been of major importance during the preapplication review because NRC was not approving the MHTGR design during the review; nonetheless, the submittal of a design approval application with important or essential material withheld from public disclosure raises significant legal and policy issues for NRC. For design certification, there would be at least a technical violation of a statutory requirement to publish the design certification rule, because the rule would ordinarily include all essential parts of the application, and a policy issue as to the desirability of a rule, with access granted only to selected persons.

8.4 Conclusions

Because Fort St. Vrain was licensed, an HTGR with the TRISO multicoated fuel can be licensed. However, DOE has not demonstrated the necessary performance proposed for the MHTGR fuel even though German fuel test data has shown this performance. The staff expects that the MHTGR reference fuel should eventually demonstrate the required performance, but there is a question as to when this will happen. Without demonstration of the proposed MHTGR fuel performance, a much lower containment leakage would be required at the design approval review stage, along with other possible changes to the design.

In the matter of the Applied Technology designation, DOE should provide the basis for designating reactor plant design information as being required to be withheld from the public.

9.0 REFERENCES

ANSI/ASME NQA-2	American National Standards Institute/American Society of Mechanical Engineers, "Quality Assurance Requirements for Nuclear Facility Applications." Standard ANSI/ASME NQA-2, 1989.
ASME Case N-499	American Society of Mechanical Engineers, "Use of SA-533 Grade B, Class 1 Plate and SA-508 Class 3 Forgings and Their Weldments for Limited Elevated Temperature Service, Section III, Division 1," Case N-499, September 13, 1991.
ASME Code, 1992	American Society of Mechanical Engineers Boiler and Pressure Vessel Code, "1992 Code Cases: Nuclear Components," 1992 Edition, July 1, 1992.
10 CFR 2	U.S. Nuclear Regulatory Commission, Code of Federal Regulations, Title 10, "Energy," Part 2, "Rules of Practice for Domestic Licensing Proceedings and Issuance of Orders."
10 CFR 20	U.S. Nuclear Regulatory Commission, Code of Federal Regulations, Title 10, "Energy," Part 20, "Standards for Protection Against Radiation."
10 CFR 50	U.S. Nuclear Regulatory Commission, Code of Federal Regulations, Title 10, "Energy," Part 50, "Domestic Licensing of Production and Utilization Facilities."
10 CFR 50(A)	U.S. Nuclear Regulatory Commission, Code of Federal Regulations, Title 10, "Energy," Part 50, "Domestic Licensing of Production and Utilization Facilities," Appendix A, "General Design Criteria for Nuclear Power Plants."
10 CFR 50(I)	U.S. Nuclear Regulatory Commission, Code of Federal Regulations, Title 10, "Energy," 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."
10 CFR 50(J)	U.S. Nuclear Regulatory Commission, Code of Federal Regulations, Title 10, "Energy," Part 50, "Domestic Licensing of Production and Utilization Facilities," Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors."

10 CFR 50(K)	U.S. Nuclear Regulatory Commission, Code of Federal Regulations, Title 10, "Energy," Part 50, "Domestic Licensing of Production and Utilization Facilities," Appendix K, "ECCS Evaluation Models."
10 CFR 52	U.S. Nuclear Regulatory Commission, Code of Federal Regulations, Title 10, "Energy," Part 52, "Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants," Subpart B, Standard Design Certification.
10 CFR 100	U.S. Nuclear Regulatory Commission, Code of Federal Regulations, Title 10, "Energy," Part 100, "Reactor Site Criteria."
40 CFR 190	U.S. Environmental Protection Agency, Code of Federal Regulations, Title 40, "Protection of the Environment," Part 190, "Environmental Radiation Protection for Nuclear Power Operations."
DOE-HTGR-85-001	U.S. Department of Energy, "Licensing Plan for the Standard HTGR," HTGR-85-001, Revision 3, February 1986.
DOE-HTGR-86004	U.S. Department of Energy, "Overall Plant Design Specification Modular High Temperature Gas-Cooled Reactor," DOE-HTGR-86004, Revision 9, May 1990 (Applied Technology).
DOE-HTGR-86011	U.S. Department of Energy, "Probabilistic Risk Assessment for the Standard Modular High Temperature Gas-Cooled Reactor," DOE-HTGR-86011, Vols. 1 and 2, Revision 5, April 1988.
DOE-HTGR-86-024	U.S. Department of Energy, "Preliminary Safety Information Document for the Standard MHTGR," HTGR-86-024, up to Amendment 13, Vols. 1-6, 1986.

Amendment 1	March 5, 1987
Amendment 2	April 2, 1987
Amendment 3	May 14, 1987
Amendment 4	June 11, 1987
Amendment 5	July 9, 1987
Amendment 6	July 30, 1987
Amendment 7	August 27, 1987
Amendment 8	November 27, 1987
Amendment 9	April 1, 1988
Amendment 10	February 3, 1989
Amendment 11	December 6, 1991
Amendment 12	March 31, 1992
Amendment 13	August 17, 1992

DOE-HTGR-86-034	U.S. Department of Energy, "Licensing Basis Events for the Modular HTGR," HTGR-86-034, April 1986 (Applied Technology)
DOE-HTGR-86-064	U.S. Department of Energy, "Regulatory Technology Development Plan for the Standard Modular High-Temperature Gas-Cooled Reactor," DOE-HTGR-86-064, Revision 1, August 1987 (Applied Technology).
DOE-HTGR-86-118	U.S. Department of Energy, "Concept Description Report, Reference Modular High Temperature Gas-Cooled Reactor Plant," DOE-HTGR-86-118, October 1986.
DOE-HTGR-87-001	U.S. Department of Energy, "Emergency Planning Bases for the Standard Modular High-Temperature Gas-Cooled Reactor," DOE-HTGR-87-001, Revision 1, August 1987 (Applied Technology).
DOE-HTGR-87089	U.S. Department of Energy, "MHTGR Assessment of NRC LWR Generic Safety Issues," DOE-HTGR-87089, September 1987 (Applied Technology).
DOE-HTGR-88311	U.S. Department of Energy, "Containment Study for MHTGR," DOE-HTGR-88311, Revision 0, September 1991.
DOE-HTGR-90257	U.S. Department of Energy, "MHTGR Fuel Process and Quality Control Description," DOE-HTGR-90257, Revision 0, September 1991 (Applied Technology).
DOE-HTGR-90286	U.S. Department of Energy, "Documentation of ASME Code Case for Elevated-Temperature Service of MHTGR Reactor Vessel Materials," Revision 0, September 1991 (Applied Technology).
DOE-HTGR-90321	U.S. Department of Energy, "450 MW(t) MHTGR Source Term and Containment Study," DOE-HTGR-90321, Revision 1, March 1993 (Applied Technology).
DOE-HTGR-90358	U.S. Department of Energy, "450 MW(t) MHTGR Reactor Graphite and Ceramics Development Plan," Revision 0, June 1993 (Applied Technology).
EPA-520/1-75-001	U.S. Environmental Protection Agency, "Manual of Protective Action Guides and Protective Actions for Nuclear Incidents," EPA-520/1-75-001, Springfield, Virginia, June 1980.
EPRI, 1987	Electric Power Research Institute, ALWR Utility Requirements Document, "Advanced Light Water Reactor Requirements Document," Chapter 1, "Overall Requirements," April 1987.

50 FR 32138	U.S. Nuclear Regulatory Commission, "Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants," Federal Register, Vol. 50, No. 153, pp. 32138-32150, August 8, 1985.
51 FR 24643	U.S. Nuclear Regulatory Commission, "Regulation of Advanced Nuclear Power Plants, Statement of Policy (10 CFR Part 50)," Federal Register, Vol 51, No. 130, pp. 24643-24648, July 8, 1986.
51 FR 28044	U.S. Nuclear Regulatory Commission, "Safety Goals for the Operation of Nuclear Power Plants: Policy Statement," Federal Register, Vol.51, No. 149, pp. 28044-28049, August 4, 1986.
59 FR 26530	U.S. Nuclear Regulatory Commission, Draft Commission Paper, "Emergency Planning Under 10 CFR Part 52," Federal Register, Vol. 59, No. 97, page 26530, May 20, 1994.
59 FR 52255	U.S. Nuclear Regulatory Commission, Proposed Rules, 10 CFR Parts 50, 52, and 100, "Reactor Site Criteria Including Seismic and Earthquake Engineering Criteria for Nuclear Power Plants and Proposed Denial of Petition From Free Environment, et al.," Federal Register, Vol. 59, No. 199, pp. 52255-52273, October 17, 1994.
60 FR 17921	U.S. Nuclear Regulatory Commission, "Standard Design Certification for the U.S. Advanced Boiling Water Reactor Design," Federal Register, Vol. 60, No. 67, pp. 17902-17924, April 7, 1995.
60 FR 17944	U.S. Nuclear Regulatory Commission, "Standard Design Certification for the System 80+ Design," Federal Register, Vol. 60, No. 67, pp. 17924-17948, April 7, 1995.
GA-A13200	General Atomics, "General Atomics Standard Safety Analysis Report (GASSAR)," GA-A13200, Docket Number STN 50-535, February 3, 1975.
NRC, 1968	U.S. Nuclear Regulatory Commission, "Safety Evaluation by the Division of Reactor Licensing, in the Matter of Public Service Company of Colorado, Fort St. Vrain Nuclear Generating Station, Docket No. 50-267," June 21, 1968.
NRC, 1972	U.S. Nuclear Regulatory Commission, "Safety Evaluation by the Division of Reactor Licensing, in the Matter of Public Service Company of Colorado, Fort St. Vrain Nuclear Generating Station, Docket No. 50-267," January 20, 1972.
NRC, 1974	U.S. Nuclear Regulatory Commission, "Safety Evaluation of the Summit Power Station, Units 1 and 2," Delmarva Power and Light, October 1974.

NUREG-0111 U.S. Nuclear Regulatory Commission, "Evaluation of High Temperature Gas Cooled Reactor Fuel Particle Coating Failure Models and Data," NUREG-0111, November 1976.

NUREG-0800, 1981a U.S. Nuclear Regulatory Commission, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition," NUREG-0800, Section 3.9.3, "ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures," July 1981.

NUREG-0800, 1981b U.S. Nuclear Regulatory Commission, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition," NUREG-0800, Section 7.4, "Safe Shutdown Systems," Revision 2, July 1981.

NUREG-0800, 1981c U.S. Nuclear Regulatory Commission, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition," NUREG-0800, Sections 13.1.2 - 13.1.3, "Operating Organization," Revision 2, July 1981.

NUREG-0800, 1982 U.S. Nuclear Regulatory Commission, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition," NUREG-0800, July 1982.

NUREG-0800, 1984 U.S. Nuclear Regulatory Commission, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition," NUREG-0800, Section 5.4.7, "Residual Heat Removal (RHR) System," Branch Technical Position RSB 5-1, "Design Requirements of the Residual Heat Removal System," Revision 3, April 1984.

NUREG-0933 U.S. Nuclear Regulatory Commission, "A Prioritization of Generic Issues," Section 5, "Chernobyl Issues," NUREG-0933, Supplement 14, Vols. 1 and 2, December 31, 1991.

NUREG-1226 U.S. Nuclear Regulatory Commission, "Development and Utilization of the NRC Policy Statement on the Regulation of Advanced Nuclear Power Plants," NUREG-1226, June 1988.

NUREG-1251 U.S. Nuclear Regulatory Commission, "Implications of the Accident at Chernobyl for Safety Regulation of Commercial Nuclear Power Plants in the United States," NUREG-1251, Vols. 1 and 2, April 1989.

NUREG-1338 (draft) U.S. Nuclear Regulatory Commission, "Draft Preapplication Safety Evaluation Report for the Modular High-Temperature Gas-Cooled Reactor," NUREG-1338 (draft), March 1989.

NUREG-1368 U.S. Nuclear Regulatory Commission, "Preapplication Safety Evaluation Report for the Power Reactor Innovative Small Module (PRISM) Liquid-Metal Reactor," NUREG-1368 (final report), February 1994.

NUREG-1462 U.S. Nuclear Regulatory Commission, "Final Safety Evaluation Report Related to the Certification of the System 80+ Design," NUREG-1462, Vols. 1 and 2, August 1994.

NUREG-1465 U.S. Nuclear Regulatory Commission, "Accident Source Terms for Light-Water Nuclear Power Plants," NUREG-1465, Final Report, February 1995.

NUREG-1503 U.S. Nuclear Regulatory Commission, "Final Safety Evaluation Report Related to the Certification of the Advanced Boiling Water Reactor Design," NUREG-1503, Vols. 1 and 2, July 1994.

NUREG-1511 U.S. Nuclear Regulatory Commission, "Reactor Pressure Vessel Status Report," NUREG-1511, December 1994.

NUREG-1512 U.S. Nuclear Regulatory Commission, "Draft Safety Evaluation Report for the Westinghouse AP600 Design," NUREG-1512, Chapter 1.2.2.7, November 1994.

NUREG/CR-4981 U.S. Nuclear Regulatory Commission "A Safety Assessment of the Use of Graphite in Nuclear Reactors Licensed by the U.S. NRC," BNL-NUREG-52092, Brookhaven National Laboratory, September 1987.

NUREG/CR-5514 U.S. Nuclear Regulatory Commission, "Modelling and Performance of the MHTGR Reactor Cavity Cooling System," NUREG/CR-5514, ORNL/TM-11451, Oak Ridge National Laboratory, April 1990.

NUREG/CR-6218 U.S. Nuclear Regulatory Commission, "A Review of the Technical Issues of Air Ingression During Severe Reactor Accidents," Appendix A, NUREG/CR-6218, SAND94-0731, Sandia National Laboratories, September 1994.

NUREG/CR-6358 U.S. Nuclear Regulatory Commission, "Assessment of United States Industry Design Codes and Standards for Application to Evolutionary and Advanced Nuclear Power Reactors," NUREG/CR-6358, Stevenson and Associates, October 1995.

PSAR Fulton Fulton Generating Station, Units 1 and 2, Preliminary Safety Analysis Report, Philadelphia Electric Company, Docket 50-463/464, November 16, 1973.

PSAR Summit Summit Power Station, Units 1 and 2, Preliminary Safety Analysis Report, Delmarva Power and Light, Docket 50-450/451, April 30, 1973.

PSC, 1991 Public Service Company of Colorado, "Supplement to Applicant's Environmental Report - Post Operating License Stage - Decommissioning [Fort St. Vrain Nuclear Generating

Station]," Chapter 2, attached to letter from A. Clegg Crawford, Vice President, Public Service Company of Colorado, to NRC, July 10, 1991.

PSC, 1992 — Public Service Company of Colorado, "Supplement to Applicant's Environmental Report - Post Operating License Stage - Decommissioning (Fort St Vrain Nuclear Generating Station]," Chapter 2, attached to letter from A. Clegg Crawford, Vice President, Public Service Company of Colorado, to NRC, April 30, 1992.

RG 1.3 — U.S. Nuclear Regulatory Commission, Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors," Revision 2, June 1974.

RG 1.4 — U.S. Nuclear Regulatory Commission, Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors," Revision 2, June 1974.

RG 1.76 — U.S. Nuclear Regulatory Commission, Regulatory Guide 1.76, "Design Basis Tornados for Nuclear Power Plants," Revision 0, April 1974.

RG 1.139 — U.S. Nuclear Regulatory Commission, Regulatory Guide 1.139 (for Comment), "Guidance for Residual Heat Removal," May 1978.

RG 1.155 — U.S. Nuclear Regulatory Commission, Regulatory Guide 1.155, "Station Blackout," Revision 0, August 1988.

SECY-88-203 — U.S. Nuclear Regulatory Commission, "Key Licensing Issues Associated with DOE Sponsored Advanced Reactor Designs," SECY-88-203, July 15, 1988.

SECY-90-016 — U.S. Nuclear Regulatory Commission, "Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements," SECY-90-016, January 12, 1990.

SECY-90-016 — U.S. Nuclear Regulatory Commission, Staff Requirements Memorandum, "SECY-90-016 - Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements," June 26, 1990.

SECY-91-074 — U.S. Nuclear Regulatory Commission, "Prototype Decisions for Advanced Reactor Designs," SECY-91-074, March 19, 1991.

SECY-91-229 U.S. Nuclear Regulatory Commission, "Severe Accident Mitigation Design Alternatives for Certified Standard Designs," SECY-91-229, July 31, 1991.

SECY-91-272 U.S. Nuclear Regulatory Commission, "Role of Personnel and Advanced Control Rooms in Future Nuclear Power Plants," SECY-91-272, August 27, 1991.

SECY-93-087 U.S. Nuclear Regulatory Commission, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs," SECY-93-087, April 2, 1993.

SECY-93-092 U.S. Nuclear Regulatory Commission, "Issues Pertaining to the Advanced Reactor (PRISM, MHTGR, and PIUS) and CANDU 3 Designs and Their Relationship to Current Regulatory Requirements," SECY-93-092, April 8, 1993.

SECY-93-092 U.S. Nuclear Regulatory Commission, Commission Staff Requirements Memorandum, "SECY-93-092 - Issues Pertaining to the Advanced Reactor (PRISM, MHTGR, and PIUS) and CANDU 3 Designs and Their Relationship to Current Regulatory Requirements," July 30, 1993.

SECY-94-084 U.S. Nuclear Regulatory Commission, "Policy and Technical Issues Associated With the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs," SECY-94-084, March 28, 1994.

SECY-94-084 U.S. Nuclear Regulatory Commission, Commission Staff Requirements Memorandum, "SECY-94-084 - Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems," June 30, 1994.

SECY-94-302 U.S. Nuclear Regulatory Commission, "Source Term-Related Technical and Licensing Issues Pertaining to Evolutionary and Passive Light-Water-Reactor Designs," SECY-94-302, December 19, 1994.

SECY-95-090 U.S. Nuclear Regulatory Commission, "Emergency Planning Under 10 CFR Part 52," SECY-95-090, April 11, 1995.

SECY-95-132 U.S. Nuclear Regulatory Commission, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems (RTNSS) in Passive Plant Designs (SECY-94-084)," SECY-95-132, May 22, 1995.

SECY-95-132 U.S. Nuclear Regulatory Commission, Commission Staff Requirements Memorandum, "SECY-94-084 - Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety System (RTNSS) in passive plant designs," June 28, 1995.

SECY-95-172 — U.S. Nuclear Regulatory Commission, "Key Technical Issues Pertaining to the Westinghouse AP600 Standardized Passive Reactor Designs," SECY-95-172, June 30, 1995.

TER 2-2-93 — Oak Ridge National Laboratory, Technical Evaluation Report, TER 2-2-93, "Evaluation of Computer Codes Used to Calculate MHTGR Accident Dose Consequences," dated February 2, 1993

TID-14844 — U.S. Atomic Energy Commission, TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," March 23, 1967.

UofT, 1994 — University of Tennessee, College of Engineering, The Nuclear Engineering Department, "Research on Fuel Performance in Modular High-Temperature Gas-Cooled Reactors," NRC-04-92-092 (RES-C92-231), Final Report, 1994.

April 26, 1990 — U.S. Nuclear Regulatory Commission, Advisory Committee on Reactor Safeguards, letter from Carlyle Michelson (Chairman) to Chairman Carr (NRC), "Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements," April 26, 1990.

April 27, 1990 — U.S. Nuclear Regulatory Commission, Memorandum from James Taylor (EDO) to the Commissioners, "Staff Response to ACRS Conclusions Regarding Evolutionary Light Water Reactor Certification Issues," April 27, 1990.

June 26, 1990 — U.S. Nuclear Regulatory Commission, Staff Requirements Memorandum, "SECY-90-016 - Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements," June 26, 1990.

May 13, 1992 — U.S. Nuclear Regulatory Commission, Advisory Committee on Reactor Safeguards, letter from David Ward (Chairman) to James Taylor (EDO), "Issues Pertaining to Evolutionary and Passive Light Water Reactors and Their Relationship to Current Regulatory Requirements," May 13, 1992.

August 17, 1992 — U.S. Nuclear Regulatory Commission, Advisory Committee on Reactor Safeguards, letter from David Ward (Chairman) to James Taylor (EDO), "Issues Pertaining to Evolutionary and Passive Light Water Reactors and Their Relationship to Current Regulatory Requirements," August 17, 1992.

September 16, 1992 — U.S. Nuclear Regulatory Commission, Advisory Committee on Reactor Safeguards, letter from David Ward (Chairman) to James Taylor (EDO), "Issues Pertaining to Evolutionary and Passive Light Water Reactors and Their Relationship to

Current Regulatory Requirements," September 16, 1992.

September 16, 1992	U.S. Nuclear Regulatory Commission, Advisory Committee on Reactor Safeguards, letter from David Ward (Chairman) to Chairman Ivan Selin (NRC), "Digital Instrumentation and Control System Reliability," September 16, 1992.
June 12, 1992	U.S. Nuclear Regulatory Commission, letter from James Taylor (EDO) to David Ward (Chairman) Advisory Committee on Reactor Safeguards, "Advisory Committee on Reactor Safeguards (ACRS) Comments Regarding the Draft Commission Paper, 'Issues Pertaining to Evolutionary and Passive Light Water Reactors and Their Relationship to Current Regulatory Requirements,'" June 12, 1992.
October 22, 1992	U.S. Nuclear Regulatory Commission, letter from James Taylor (EDO) to David Ward (Chairman) Advisory Committee on Reactor Safeguards, October 22, 1992.
October 23, 1992	U.S. Nuclear Regulatory Commission, letter from James Taylor (EDO) to David Ward (Chairman) Advisory Committee on Reactor Safeguards, "Defense Against Common Mode Failures in Digital Instrumentation and Control (I&C) Systems," October 23, 1992.
October 29, 1992	U.S. Nuclear Regulatory Commission, letter from James Taylor (EDO) to David Ward (Chairman) Advisory Committee on Reactor Safeguards, October 29, 1992.
February 19, 1993	U.S. Nuclear Regulatory Commission, Advisory Committee on Reactor Safeguards, letter from Paul Shewmon (Chairman) to Chairman Selin (NRC), "Issues Pertaining to the Advanced Reactor (PRISM, MHTGR, and PIUS) and CANDU 3 Designs and Their Relationship to Current Regulatory Requirements," February 19, 1993. (Place in MHTGR chronology)
July 21, 1993	U.S. Nuclear Regulatory Commission, Staff Requirements Memorandum, "SECY-93-087 - Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs," July 21, 1993.
July 30, 1993	U.S. Nuclear Regulatory Commission, Commission Staff Requirements Memorandum, "SECY-93-092 - Issues Pertaining to the Advanced Reactor (PRISM, MHTGR, and PIUS) and CANDU 3 Designs and Their Relationship to Current Regulatory Requirements," July 30, 1993.
November 10, 1993	U.S. Nuclear Regulatory Commission, Advisory Committee on Reactor Safeguards, letter from J. Wilkins, Jr. (Chairman) to Chairman Ivan Selin (NRC), "Draft Commission Paper,'Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive

	Plant Designs,'" November 10, 1993.
February 2, 1994	U.S. Nuclear Regulatory Commission, letter from James Taylor (EDO) to J. Wilkins, Jr. (Chairman) Advisory Committee on Reactor Safeguards, "Draft Commission Paper, 'Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs,'" February 2, 1994.
May 24, 1994	Westinghouse Electric Corporation, letter from Nicholas Liparulo (Manager) to Document Control Desk (Attention: Samuel Chilk) (NRC), May 24, 1994.
June 30, 1994	U.S. Nuclear Regulatory Commission, Commission Staff Requirements Memorandum, "SECY-94-084 - Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems," June 30, 1994.
September 20, 1994	U.S. Nuclear Regulatory Commission, Advisory Committee on Reactor Safeguards, letter to Chairman Selin (NRC), "Proposed Final Version of NUREG-1465, 'Accident Source Terms for Light-Water Nuclear Power Plants'," September 20, 1994.
October 7, 1994	University of Tennessee, College of Engineering, The Nuclear Engineering Department, letter from Paul Kasten, Research Professor, to Donald Carlson, Office of Nuclear Regulatory Research (NRC), "Final Report, Research on Fuel Performance in Modular High-Temperature Gas-Cooled Reactors, Grant Number: NRC-04-92-092 (RES-C92-231)," October 7, 1994.
October 24, 1994	U.S. Nuclear Regulatory Commission, letter from Dennis Crutchfield to Nicholas Liparulo (Westinghouse Electric Corporation), Docket No. 52-003, October 24, 1994.
October 25, 1991	U.S. Nuclear Regulatory Commission, Commission Staff Requirements Memorandum, "SECY-91-229 - Severe Accident Mitigation Design Alternatives for Certified Standard designs," October 25, 1991.
February 27, 1995	U.S. Nuclear Regulatory Commission, memorandum from James Taylor to the Commission, "Simplification of Emergency Planning for Reactors with Greater Safety Margins," February 27, 1995.
May 9, 1995	U.S. Nuclear Regulatory Commission, meeting summary, "Summary of March 30, 1995, Meeting to Discuss Passive System Reliability for the Westinghouse AP600 Design," Docket No. 52-003, May 9, 1995.

May 17, 1995	U.S. Nuclear Regulatory Commission, meeting summary, "Summary of April 20, 1995, Meeting to Discuss Passive System Reliability for the Westinghouse AP600 Design," Docket No. 52-003, May 17, 1995.
May 18, 1995	U.S. Nuclear Regulatory Commission, letter from D. Crutchfiel (NRC) to N. Liparulo (Westinghouse), "Draft Commission Paper on Staff Positions on Technical Issues Pertaining to the Westinghouse AP600 Standardized Passive Reactor Design," May 18, 1995.
June 28, 1995	U.S. Nuclear Regulatory Commission, Commission Staff Requirements Memorandum, "SECY-95-132 - Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety System (RTNSS) in passive plant designs (SECY-94-084)," June 28, 1995.

NUREG-1338

Preapplication Safety Evaluation Report for the Modular High-Temperature Gas-Cooled Reactor (MHTGR)

Draft Copy of the Final Report
Volume 2: Appendices

Manuscript Completed : June 1995
Date Published: December 1995

Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

APPENDICES

The following 10 appendices are included with the Preapplication Safety Evaluation Report (PSER) for the standard Modular High Temperature Gas-Cooled Reactor (MHTGR) design submitted by the U.S. Department of Energy (DOE) and described in HTGR-86-024, "Preliminary Safety Information Document for the Standard MHTGR," up to Amendment 13, August 17, 1992:

- Appendix A — Chronology of Correspondence and Meetings
- Appendix B — Technical Description of the MHTGR Design
- Appendix C — DOE PSER Tracking System for Draft NUREG-1338 Issues
- Appendix D — DOE Submittals on Draft NUREG-1338 Issues
- Appendix E — Commission Paper SECY-93-092 (Paper, Staff Requirements Memorandum, and Advisory Committee on Reactor Safeguards (ACRS) Response)
- Appendix F — Commission Paper SECY-90-016 (Paper, Staff Requirements Memorandum, and ACRS Response)
- Appendix G — Commission Paper SECY-93-087 (Paper, Staff Requirements Memorandum, and ACRS Response)
- Appendix H — Commission Paper SECY-94-084 (Paper, Staff Requirements Memorandum, and ACRS Response)
- Appendix I — Commission Paper SECY-95-132 (Paper, Staff Requirements Memorandum, Westinghouse letter, and NRC letter)
- Appendix J — Contractor Reports on the MHTGR Design

APPENDIX A

CHRONOLOGY OF CORRESPONDENCE AND MEETINGS

1. INTRODUCTION

This appendix contains the chronological record of correspondence and meetings between the Nuclear Regulatory Commission (NRC) and the Department of Energy (DOE) on the MHTGR design since the NRC issued the draft NUREG-1338, "Draft Preapplication Safety Evaluation Report [PSER] for the Modular High-Temperature Gas-Cooled Reactor," in March 1989. This appendix also contains meetings held with the NRC Advisory Committee on Reactor Safeguards (ACRS) and a few other relevant records.

The correspondence and meetings held during the development of the draft NUREG-1338 are listed in Chapter 18 and Table 1.1, respectively, of that document.

The letters from DOE which contained "Applied Technology" information, discussed in Sections 1.8 and 4.2.9 of this report, are identified with the words "(Applied Technology)" in the description of the letter, in the right-hand-side column below.

2. CHRONOLOGY

The list of (1) correspondence between the NRC and DOE, (2) meetings held with DOE and ACRS on the MHTGR design, and (3) other relevant records on the MHTGR Project 672 docket not listed in draft NUREG-1338 is the following:

January 15, 1987	U.S. Department of Energy, letter from Francis Gavigan (DOE) to Dr. Themis Speis (NRC), submitting the Regulatory Technology Development Plan (RTDP) [DOE-HTGR-86-064] for the MHTGR program, January 15, 1987.
August 17, 1988	U.S. Nuclear Regulatory Commission, letter from Victor Stello (NRC) to Theodore Garrish, requesting information on DOE's approach to the containment for the MHTGR and the New Production Reactor.
September 16, 1988	U.S. Department of Energy, letter from Theodore Garrish (DOE) to Mr. Victor Stello (NRC), in response to NRC's letter of August 17, 1988, addressing the apparent conflict between the approach to containment for the New Production reactor and the MHTGR.
October 13, 1988	U.S. Nuclear Regulatory Commission, Advisory Committee on Reactor Safeguards, letter from William Kerr (Chairman) to Chairman Lando Zech (NRC), "Preapplication Safety Evaluation Report for the Modular High Temperature Gas Cooled Reactor," October 13, 1988.

February 28, 1989	U.S. Nuclear Regulatory Commission, letter from Eric Beckjord (NRC) to M.A. Novak (DOE) providing copies of Draft NUREG-1338, draft preapplication safety evaluation for the MHTGR, and requesting information on (1) engineering studies evaluating the containment and decay heat removal for the MHTGR and (2) the differences in the containment between the MHTGR and the NPR-MHTGR (New Production Reactor-MHTGR), February 28, 1989.
November 28, 1989	U.S. Department of Energy, letter from William Young (DOE) to Eric Beckjord (NRC), responded to NRC letter of February 28, 1989, and submitted report, DOE-HTGR-88311, November 28, 1989.
December 18, 1989	U.S. Nuclear Regulatory Commission, letter from Eric Beckjord (NRC) to William Young (DOE), in response to DOE's letter of November 28, 1989, addressing when the revised NUREG-1338 report on the MHTGR might be issued and the use of the term "containment system" by DOE to describe the containment structure for the MHTGR.
May 9, 1990	U.S. Nuclear Regulatory Commission, letter from Bill Morris to Sol Rosen (DOE) requesting additional information on DOE report DOE-HTGR-88311, "Containment Study for MHTGR," May 9, 1990.
June 27, 1990	U.S. Department of Energy, letter from Sol Rosen (DOE) to Bill Morris (NRC) responding to May 9, 1990, NRC letter, June 27, 1990.
April 23, 1991	Meeting between NRC and DOE. NRC meeting summary issued June 24, 1991.
May 21, 1991	General Atomics, letter from G. Bramblett (GA) to J.N. Donohew (NRC) enclosing Copy #89 of DOE report DOE-HTGR-86011, Revision 5, Volumes 1 and 2 (April 1988) on probabilistic risk assessment for the MHTGR (Applied Technology) and stating that the GA claimed proprietary information has been released to the U.S. government, May 21, 1991.
May 22, 1991	U.S. Department of Energy, letter from P.M. Williams (DOE) to Robert Pierson and Z.R. Rosztoczy (NRC) stating that DOE was looking forward to their attendance on June 4, 5, and 6 at a technical briefing, May 22, 1991.
May 28, 1991	U.S. Nuclear Regulatory Commission, letter from J.N. Donohew (NRC) to Sol Rosen (DOE) requesting participation of DOE and its MHTGR contractors at a meeting on June 13-14 with the Nuclear Safety Research Review Committee, an advisory group to the NRC Office of Nuclear Regulatory Research, May 28, 1991.

May 31, 1991	U.S. Department of Energy, letter from P.M. Williams (DOE) to Robert Pierson and Z.R. Rosztoczy (NRC) enclosing the notes prepared by DOE for the April 23 DOE/NRC meeting, May 31, 1991.
June 4-6, 1991	U.S. Department of Energy presentation of MHTGR technical details to the NRC staff. NRC internal memorandum issued July 31, 1991.
June 24, 1991	U.S. Nuclear Regulatory Commission, meeting summary, "Summary of the April 23, 1991, Meeting with the Department of Energy on the Modular High-Temperature Gas-Cooled Reactor (MHTGR)," Project No. 672, June 24, 1991.
July 9, 1991	U.S. Department of Energy, letter from P.M. Williams (DOE) to Robert Pierson and Z.R. Rosztoczy (NRC) transmitting seven documents pertaining to MHTGR fuel integrity, July 9, 1991 (Applied Technology).
July 12, 1991	U.S. Department of Energy, letter from Peter Williams (DOE) to J.N. Donohew (NRC) submitting information requested during the June 4 - 6, 1991, presentation on MHTGR technical information, July 12, 1991 (Applied Technology).
July 16, 1991	U.S. Department of Energy, letter from P.M. Williams (DOE) to Robert Pierson and Zoltan Rosztoczy (NRC) referring to the letter of July 9, 1991, and correcting the statements about Applied Technology information on seven documents submitted in the July 9, 1991, letter pertaining to MHTGR fuel integrity, July 16, 1991.
July 31, 1991	U.S. Department of Energy, letter from P.M. Williams (DOE) to Robert Pierson (NRC) transmitting the DOE/HTGR Program's "PSER Issues Tracking System," July 31, 1991.
July 31, 1991	U.S. Nuclear Regulatory Commission, presentation summary memorandum, "Summary of Presentations by the Department of Energy at General Atomics Offices During June 4 - 6, 1991, on the Modular High-Temperature Gas-Cooled Reactor (MHTGR)," Project No. 672, July 31, 1991.
August 6, 1991	U.S. Department of Energy, presentation to ACRS Subcommittee on Advanced Reactor Designs, August 6, 1991.
August 8, 1991	U.S. Nuclear Regulatory Commission, letter from J.N. Donohew (NRC) to P.M. Williams (DOE), "Request for Additional Information on the Modular High Temperature Gas-Cooled Reactor (MHTGR) Design," Project No. 672, August 8, 1991.

August 21, 1991	U.S. Department of Energy, letter from Jerry Griffith (DOE) to T.E. Murley (NRC) stating that DOE is resuming its formal interactions with NRC and ACRS on the MHTGR, August 21, 1991.
August 28, 1991	U.S. Nuclear Regulatory Commission, letter from J.N. Donohew (NRC) to P.M. Williams (DOE), "Request for Schedules for Future Submittals on the Modular High Temperature Gas-Cooled Reactor (MHTGR) Design," August 28, 1991.
October 2, 1991	U.S. Department of Energy, letter from P.M. Williams (DOE) to M. El-Zaftawy (NRC-ACRS) responding to a question on thermal fatigue of steam generator tubes raised at the August 6, 1991, meeting with ACRS, October 2, 1991.
October 2, 1991	U.S. Department of Energy, letter from P.M. Williams (DOE) to J.E. Dyer (NRC) enclosing a General Atomics report, DOE-HGTR-90257, "MHTGR Fuel Process and Quality Control Description," on the MHTGR fuel process and the quality control description for this process, October 2, 1991 (Applied Technology).
October 11, 1991	U.S. Department of Energy, letter from P.M. Williams (DOE) to NRC submitting schedule information for responding to Category A and B1 items of the DOE PSER Issues Tracking System, October 11, 1991.
October 17, 1991	U.S. Department of Energy, letter from P.M. Williams (DOE) to NRC transmitting MHTGR busbar cost study, DOE-HTGR-87-086, Revision 2, October 17, 1991 (Applied Technology).
October 23, 1991	NRC/DOE meeting on fuel performance and research. NRC meeting summary issued January 13, 1992.
November 7, 1991	U.S. Department of Energy, letter from P.M. Williams (DOE) to Document Control Desk (NRC), submitting report DOE-HTGR-90286, "Documentation of ASME Code Case for Elevated-Temperature Service of MHTGR Reactor Vessel Materials," Revision 0 (September 1991), November 7, 1991 (Applied Technology).
November 8, 1991	U.S. Department of Energy, letter from P.M. Williams (DOE) to J.N. Donohew (NRC) requesting confirmation of numbered copies of the DOE Preliminary Safety Information Document (PSID) for the standard MHTGR retained by NRC, November 8, 1991.

November 26, 1991	U.S. Department of Energy, letter from P.M. Williams (DOE) to Document Control Desk (NRC) transmitting DOE report, DOE-HTGR-85107, Revision A, March 1989, on U.S. and German accident fuel performance models, November 26, 1991 (Applied Technology).
December 4, 1991	U.S. Department of Energy letter from C.L. Reid (PDCO) to PSID Holders submitting Amendment 11 to the PSID, December 4, 1991.
December 9, 1991	U.S. Department of Energy, letter from P.M. Williams (DOE) to Document Control Desk (NRC) enclosing PSID Amendment 11 in response to NRC August 8, 1991, letter, December 9, 1991 (Applied Technology).
December 17-20, 1991	NRC/DOE meeting on fuel performance and fission product transport. NRC meeting summary issued March 10, 1992.
December 20, 1991	General Atomics, letter from George Bramblett (GA) to J.N. Donohew (NRC) documenting the transmittal of Applied Technology information to NRC contractors during the December 19 and 20 DOE/NRC meeting, December 20, 1991.
January 13, 1992	U.S. Nuclear Regulatory Commission, meeting summary, "Summary of Presentations During the October 23, 1991, Meeting on the Modular High Temperature Gas-Cooled Reactor (MHTGR)," Project No. 672, January 13, 1992.
January 22, 1992	NRC/DOE meeting on equipment safety classification. NRC meeting summary issued April 10, 1992.
January 28, 1992	U.S. Department of Energy, letter from P.M. Williams (DOE) to Document Control Desk (NRC) transmitting Oak Ridge National Laboratory (ORNL) report ORNL/TM-11846, October 1991, on preliminary analyses of water-vapor injection experiments, January 28, 1992 (Applied Technology).
February 6, 1992	U.S. Department of Energy, letter from P.M. Williams (DOE) to Document Control Desk (NRC) documenting the transfer of Applied Technology material to NRC consultants attending December 19 and 20, 1991, meeting, February 6, 1992.
February 18, 1992	U.S. Nuclear Regulatory Commission, letter from Dennis Crutchfield (NRC) to Sol Rosen (DOE) requesting information on DOE's plans for the MHTGR, February 18, 1992.
February 20, 1992	NRC/DOE meeting on fuel performance, including the DOE gas-cooled New Production Reactor (NPR). NRC meeting summary issued April 15, 1992.

February 26-27, 1992	DOE presentation to ACRS at Oak Ridge, Tennessee, on the fuel, fission-product transport, and ASME Code case inquiry.
March 4, 1992	U.S. Nuclear Regulatory Commission, letter from J.N. Donohew (NRC) to P.M. Williams (DOE), "Request for Additional Information on the Fuel Design and Testing for the Gas-Cooled New Production Reactor (NPR) Design," Project No. 672, March 4, 1992.
March 4, 1992	U.S. Nuclear Regulatory Commission, letter from J.N. Donohew (NRC) to P.M. Williams (DOE), "Equipment Safety Classification Differences Between the Modular High Temperature Gas-Cooled Reactor (MHTGR) and Gas-Cooled New Production Reactor (NPR) Designs," Project No. 672, March 4, 1992.
March 10, 1992	U.S. Nuclear Regulatory Commission, meeting summary, "Summary of Presentations During the December 17-20, 1991, Meetings on the Modular High Temperature Gas-Cooled Reactor (MHTGR)," Project No. 672, March 10, 1992.
March 19, 1992	U.S. Department of Energy, letter from Peter Williams to All MHTGR PSID Holders stating the MHTGR PSID has been designated Applied Technology material and should be handled in accordance with the enclosed guidance, March 19, 1992.
March 31, 1992	U.S. Department of Energy letter from C.L. Reid (Plant Design Control Office, PDCO) to PSID Holders submitting Amendment 12 to the PSID, March 31, 1992.
April 10. 1992	U.S. Nuclear Regulatory Commission, meeting summary, "Summary of Presentations During the January 22, 1992, Meeting on Equipment Classification for the Modular High Temperature Gas-Cooled Reactor (MHTGR)," Project No. 672, April 10, 1992.
April 15, 1992	U.S. Nuclear Regulatory Commission, meeting summary, "Summary of the Meeting Held with DOE and Its Contractors on February 20, 1992," Project No. 672, April 15, 1992.
May 21, 1992	U.S. Department of Energy, letter from Sol Rosen (DOE) to Dennis Crutchfield (NRC) enclosing plans and schedules for the MHTGR program, May 21, 1992.
June 24, 1992	NRC/DOE meeting on advanced reactor policy issues. NRC meeting summary issued August 20, 1992.
June 24, 1992	U.S. Department of Energy, letter from P.M. Williams (DOE) to Document Control Desk (NRC) submitting in response to NRC requests dated March 4, 1992, on fuel

design and testing, and safety classification, June 24. 1992 (Applied Technology).

June 25, 1992	U.S. Department of Energy, letter from P.M. Williams (DOE) to Document Control Desk (NRC) transmitting report DOE-HTGR-88486, "The Effect of Water Vapor on the Release of Gaseous Fission Products from High Temperature Gas-Cooled Reactor Fuel Compacts Containing Exposed Uranium Oxycarbide Fuel," June 25, 1992 (Applied Technology).
August 17, 1992	U.S. Department of Energy letter from P.M. Williams (DOE) to Document Control Desk (NRC) submitting Amendment 13 to the PSID, August 17, 1992
August 19, 1992	U.S. Department of Energy, letter from Peter Williams to NRC transmitting schedule information on the MHTGR program plans for the MHTGR main circulator development, August 19, 1992.
August 20, 1992	U.S. Nuclear Regulatory Commission, meeting summary, "Summary of the Meeting Held with DOE on June 24, 1992, on the Modular High Temperature Gas-Cooled Reactor Design," Project No. 672, August 20, 1992.
December 14, 1992	U.S. Nuclear Regulatory Commission, letter from Dennis Crutchfield (NRC) to Jerry Griffith (DOE) transmitting final Commission paper SECY-92-393, "Updated Plans and Schedules for the Preapplication Reviews of the Advanced Reactors (MHTGR, PRISM, and PIUS) and CANDU 3 Designs," December 14, 1992.
December 16, 1992	U.S. Nuclear Regulatory Commission, letter from Dennis Crutchfield (NRC) to P.M. Williams (DOE) transmitting two Commission papers, draft SECY 93-092, "Issues Pertaining to the Advanced Reactor (PRISM, MHTGR, and PIUS) and CANDU 3 Designs and Their Relationship to Current Regulatory Requirements," and final SECY-92-393, December 16, 1992.
January 7-8, 1993	NRC staff meeting with the ACRS subcommittee on Commission paper SECY-93-092. The ACRS letter dated February 19, 1993, addressed this meeting.
January 19, 1993	U.S. Nuclear Regulatory Commission, Advisory Committee on Reactor Safeguards, letter from John Larkins to J. David Nulton (DOE) extending an invitation to DOE to discuss the MHTGR on February 11-13, 1993, at the ACRS meeting, January 19, 1993.
January 25, 1993	U.S. Department of Energy, letter from P.M. Williams (DOE) to J.N. Donohew (NRC) responding to NRC letter of December 16, 1993, and commenting on the draft Commission

paper SECY-93-092, January 25, 1993.

January 27, 1993	U.S. Department of Energy, letter from E.C. Brolin (DOE) to Dr. Thomas Murley (NRC) disagreeing with the MHTGR schedule in SECY-92-393, "Updated Plans and Schedules for the Preapplication Reviews of the Advanced Reactor Designs (MHTGR, PRISM, PIUS, and CANDU 3)," January 27, 1993.
January 28, 1993	Gas-Cooled Reactor Associates, letter from David Hoffman (GCRA) to Dennis Crutchfield (NRC) on Commission papers SECY-92-393 and SECY-93-092. January 28, 1993.
February 11-13, 1993	NRC staff meeting with the ACRS full committee on Commission paper SECY-93-092. The ACRS letter dated February 19, 1993, addressed this meeting.
February 11, 1993	DOE presentation to the ACRS on Commission paper SECY-93-092.
February 19, 1993	U.S. Nuclear Regulatory Commission, Advisory Committee on Reactor Safeguards, letter from Paul Shewmon (Chairman) to Chairman Selin (NRC), "Issues Pertaining to the Advanced Reactor (PRISM, MHTGR, and PIUS) and CANDU 3 Designs and Their Relationship to Current Regulatory Requirements," February 19, 1993.
March 12, 1993	U.S. Department of Energy, letter from Elizabeth Buffam to All Applied Technology Holders transmitting procedures for handling Applied Technology information, March 12, 1993.
March 12, 1993	U.S. Nuclear Regulatory Commission, letter from J.N. Donohew (NRC) to P.M. Williams (DOE), "Requested Review of Two NRC Documents Containing Applied Technology Classified Material on the Modular High-Temperature Gas-Cooled Reactor (MHTGR) Design," Project No. 672, March 12, 1993.
March 29, 1993	U.S. Nuclear Regulatory Commission, letter from Dr. Thomas Murley (NRC) to E.C. Brolin (DOE) responding to DOE letter dated January 27, 1993, and stating that NRC is reevaluating its plans for preapplications reviews of the MHTGR and PRISM, March 29, 1993.
April 1, 1993	U.S. Department of Energy, letter from P.M. Williams (DOE) to NRC stating that two NRC reports, NUREG/CR-5983, "Safety Aspects of Forced Flow Cooldown Transients in Modular High Temperature Gas-Cooled Reactors," and NUREG/CR-5984, "Code and Model Extensions of the THATCH Code for Modular High Temperature Gas-Cooled Reactors," have no Applied Technology information, April 1, 1993.

April 13, 1993	U.S. Department of Energy, letter from P.M. Williams (DOE) to NRC transmitting report DOE-HTGR-90321, Revision 1, March 1993, on the 450 MW(t) MHTGR source term and the containment, April 13, 1993 (Applied Technology).
April 29, 1993	U.S. Nuclear Regulatory Commission, letter from Dennis Crutchfield (NRC) to Jerry Griffith (DOE) asking to meet with DOE to discuss Applied Technology classification of MHTGR and PRISM information, April 29, 1993.
April 30, 1993	U.S. Nuclear Regulatory Commission, letter from Dennis Crutchfield (NRC) to P.M. Williams (DOE) transmitting the final Commission papers SECY-93-092 and SECY-93-104, April 30, 1993.
May 7, 1993	Gas-Cooled Reactor Associates, letter from David Hoffman (GCRA) to Chairman Ivan Selin and fellow Commissioners (NRC) transmitting statements submitted by Frederick Buckman to the Appropriations Subcommittee on Energy and Water development, U.S. House of Representatives, May 7, 1993.
May 26, 1993	U.S. Department of Energy, letter from Jerry Griffith (DOE) to Dennis Crutchfield (NRC) withdrawing the Applied Technology classification from the PRISM PSID, May 26, 1993.
June 22, 1993	U.S. Nuclear Regulatory Commission, letter from Dr. Thomas Murley (NRC) to E.C. Brolin (DOE) responding to DOE letter dated May 21, 1993, and setting a date for the final MHTGR PSER, June 22, 1993.
July 1, 1993	NRC/DOE meeting on final PSER schedule. NRC meeting summary issued July 8, 1993.
July 8, 1993	U.S. Nuclear Regulatory Commission, letter from J.N. Donohew (NRC) to P.M. Williams (DOE), "Review Requested of NRC Contractor Document Possibly Containing Applied Technology Material on the Modular High-Temperature Gas-Cooled Reactor (MHTGR) Design," contractor report TER 2-2-93, July 8, 1993.
July 8, 1993	U.S. Nuclear Regulatory Commission, meeting summary, "Summary of the Meeting Held with DOE on July 1, 1993, on the Modular High-Temperature Gas-Cooled Reactor Design," Project No. 672, July 8, 1993.
July 16, 1993	U.S. Department of Energy, letter from P.M. Williams (DOE) to NRC, submitting the report DOE-HTGR-90352, "Integrated Technology Plan to Support MHTGR Source Term and Containment Concept," Revision 0, April 1993, July 16, 1993 (Applied Technology).

July 16, 1993	U.S. Department of Energy, letter from P.M. Williams (DOE) to Document Control Desk (NRC) transmitting the following three Applied Technology reports: DOE-HTGR-90348, Revision 0, on MHTGR reactor physics development; DOE-HTGR-90357, Revision 0, on 450 MW(t) MHTGR reactor metals development; DOE-HTGR-90358, Revision 0, on 450 MW(t) MHTGR reactor graphite and ceramics development, July 16, 1993.
July 19, 1993	U.S. Department of Energy, letter from E.C. Brolin (DOE) to Secretary, NRC, transmitting DOE's comments on the NRC's review of the NRC fee policy, July 19, 1993.
July 21, 1993	U.S. Department of Energy, letter from J. Nulton to NRC submitting several changes to the NRC MHTGR service list, July 21, 1993.
July 23, 1993	U.S. Nuclear Regulatory Commission, letter from J.N. Donohew (NRC) to P.M. Williams (DOE), "Review Requested of Second Nuclear Regulatory Commission (NRC) Contractor Document Possibly Containing Applied Technology Material on the Modular High-Temperature Gas-Cooled Reactor (MHTGR) Design," contractor report TER 2-10-93, July 23, 1993.
July 28, 1993	U.S. Nuclear Regulatory Commission, letter from J.N. Donohew (NRC) to P.M. Williams (DOE), "Review Requested of Third Nuclear Regulatory Commission (NRC) Contractor Document Possibly Containing Applied Technology Material on the Modular High-Temperature Gas-Cooled Reactor (MHTGR) Design," contractor report TER 12-3-92, July 28, 1993.
August 26, 1993	U.S. Department of Energy, letter from P.M. Williams (DOE) to NRC responding to NRC letter dated July 8, 1993, on Applied Technology information in NRC contractor report, TER 2-2-93, "Evaluation of Computer Codes Used to Calculate MHTGR Accident Dose Consequences," August 26, 1993 (Applied Technology).
August 26, 1993	U.S. Department of Energy, letter from P.M. Williams (DOE) to NRC responding to NRC letter dated July 23, 1993, on Applied Technology information in NRC contractor report, TER 2-10-93, "Review and Evaluation of Recent Publications Bearing on the Fuels Sections of the Draft MHTGR PSER," August 26, 1993 (Applied Technology).

August 26, 1993	U.S. Nuclear Regulatory Commission, letter from J.N. Donohew (DOE) to P.M. Williams (DOE), "Structural and Seismic Design Information for the Modular High Temperature Gas Cooled Reactor (MHTGR) Design," Project No. 672, August 26, 1993.
August 30, 1993	U.S. Department of Energy, letter from Peter Williams to NRC responding to NRC letter dated July 28, 1993, on Applied Technology information in NRC contractor report, TER 12-3-92, "Update of Independent Analysis Section 15.4, Preapplication Safety Evaluation Report for the Modular High Temperature Gas-Cooled Reactor (MHTGR), NUREG-1338," August 30, 1993 (Applied Technology).
September 17, 1993	U.S. Department of Energy, letter from Jerry Griffith to NRC enclosing addition information on the seismic and structural design methodologies for the MHTGR, September 17, 1993 (Applied Technology).
October 5, 1993	U.S. Nuclear Regulatory Commission, letter from J.N. Donohew (NRC) to P.M. Williams (DOE), "Technical Review Requested of Nuclear Regulatory Commission (NRC) Contractor Report 2-2-93 on the Modular High-Temperature Gas-Cooled Reactor (MHTGR) Design," October 5, 1993.
October 5, 1993	U.S. Nuclear Regulatory Commission, letter from J.N. Donohew (NRC) to P.M. Williams (DOE), "Technical Review Requested of Nuclear Regulatory Commission (NRC) Contractor Report 12-3-92 on the Modular High Temperature Gas-Cooled Reactor (MHTGR) Design," October 5, 1993.
October 12, 1993	U.S. Nuclear Regulatory Commission, letter from J.N. Donohew (NRC) to Peter Williams (DOE), "Technical Review Requested of Nuclear Regulatory Commission (NRC) Contractor Report 2-10-93 on the Modular High Temperature Gas-Cooled Reactor (MHTGR) Design," October 12, 1993.
December 20, 1993	U.S. Department of Energy, letter from Jerry Griffith (DOE) to NRC stating that DOE has no schedule for responding to the three NRC letters of October 5 and 12, 1993 (listed above), December 20, 1993.
March 30, 1994	U.S. Department of Energy, letter from Daniel Dreyfus to Dr. Gail Marcus (NRC) requesting NRC review of the attached draft DOE report, "Draft 5-Year Plan for Advanced Reactor Activities under the Energy Act of 1992," March 30, 1994.
March 31, 1994	U.S. Nuclear Regulatory Commission, letter from Dennis Crutchfield to E.C. Brolin (DOE) providing the date for the issuance of the final MHTGR Preapplication Safety Evaluation Report and requesting information on DOE's

plans for the MHTGR, March 31, 1994.

April 14, 1994	U.S. Department of Energy, letter from Daniel Dreyfus to Dr. Gail Marcus (NRC) stating that the deadline for the review of the draft report in the March 30, 1994, letter was extended and NRC was invited to a DOE briefing on the report, April 14, 1994.
April 26, 1994	U.S. Nuclear Regulatory Commission, letter from James Taylor to Daniel Dreyfus (DOE) commenting on DOE's March 30, 1994, draft report, "5-Year Plan for Advanced Reactor Activities," April 26, 1994.
May 4, 1994	U.S. Department of Energy, letter from J.D. Nulton (DOE) to Dennis Crutchfield (NRC) responding to the NRC letter dated March 31, 1994, requesting plans and schedules for the MHTGR program, May 4, 1994.
May 23-24, 1994	DOE presentation of MHTGR technical details. NRC presentation summary issued July 12, 1994.
June 7, 1994	General Atomics, letter from R. Forssell to Dr. Gail Marcus (NRC) acknowledging the NRC visit to GA and presentations on the MHTGR design the previous week, June 7, 1994.
June 27, 1994	Plant Design Control Office, letter from C. Reid (PDCO) to All Applied Technology Holders transmitting the recent revisions to procedures for handling Applied Technology information, June 27, 1994.
July 12, 1994	U.S. Nuclear Regulatory Commission, presentation summary, "Summary of Presentations by the Department of Energy at General Atomics Offices, On May 23 and 24, 1994, on the Modular High-Temperature Gas-Cooled Reactor (MHTGR) Design," Project No. 672, July 12, 1994.
August 17, 1994	U.S. Department of Energy, letter from Warren Chernock (DOE) to Dennis Crutchfield (NRC) requesting that all correspondence and communications concerning PRISM and MHTGR be addressed to him, August 17, 1994.
September 28, 1994	U.S. Department of Energy, letter from Warren Chernock (DOE) to NRC submitting a revised distribution list for the participants in the DOE HTGR program, September 28, 1994.
September 29, 1994	NRC/DOE meeting on final PSER content and schedule. NRC meeting summary issued October 7, 1994.

October 7, 1994	U.S. Nuclear Regulatory Commission, meeting summary, "Summary of the September 29, 1994, Meeting with DOE on the Modular High-Temperature Gas-Cooled Reactor (MHTGR) Design," Project No. 672, October 7, 1994.
October 11, 1994	U.S. Nuclear Regulatory Commission, letter from J.N. Donohew (NRC) to Warren Chernock (DOE), "Review Requested of NRC Contractor Documents Possibly Containing Applied Technology Material on the Modular High-Temperature Gas-Cooled Reactor (MHTGR) Design," October 11, 1994.
October 12, 1994	U.S. Department of Energy, letter from Warren Chernock to NRC responding to the NRC letter dated October 5, 1993, and submitting technical comments on NRC contractor Technical Evaluation Report (TER) 12-3-93, October 12, 1994.
December 13, 1994	U.S. Department of Energy, letter from Warren Chernock (DOE) to J.N. Donohew (NRC) responding to NRC letter dated October 11, 1994, stating there is no Applied Technology information in five NRC contractor TERs, December 13, 1994.
January 18, 1995	U.S. Department of Energy, letter from John Herczeg (DOE) to Document Control Desk (NRC) responding to NRC letter dated October 12, 1993, and submitting technical comments on NRC contractor report TER 2-10-93, January 18, 1995.
February 7, 1995	U.S. Department of Energy, letter from John Herczeg (DOE) to Document Control Desk (NRC) stating that the proprietary information in Volume 2 of the MHTGR Probabilistic Risk Assessment Report had been released with unlimited rights to the U.S. Government, February 7, 1995.
February 8, 1995	U.S. Department of Energy, letter from John Herczeg to J.N. Donohew (NRC) stating that the Applied Technology classification is being removed from the following documents: HTGR-86-024, Volumes 1 through 6, "Preliminary Safety Information Document for the Standard MHTGR," and DOE-HTGR-86-011, Volumes 1 and 2, "Probabilistic Risk Assessment for the Standard Modular High-Temperature Gas-Cooled Reactor," February 8, 1995.
February 23, 1995	U.S. Department of Energy, letter from Sterling Franks (DOE) to Document Control Desk (NRC) submitting management changes to the Gas Reactor Program and requesting that correspondence concerning the MHTGR be addressed to Mr. Ernest A. Condon, February 23, 1995.

February 24, 1995	U.S. Department of Energy, letter from Sterling Franks (DOE) to Document Control Desk (NRC) commenting on Oak Ridge National Laboratory Technical Evaluation Report (TER) 2-2-93, "Evaluation of Computer Codes Used to Calculate MHTGR Accident Dose Consequences," February 24, 1995.
April 13, 1995	U.S. Nuclear Regulatory Commission, letter from Dennis Crutchfield (NRC) to Mark Forssell (General Atomics) requesting the plans and schedule for submitting a design certification application for the MHTGR in the future, April 13, 1995.
May 5, 1995	General Atomics, letter from Walter Simon (GA) to Dennis Crutchfield (NRC) responding to the April 13, 1995, letter stating that an application to NRC on the MHTGR is expected in 1998 with support from Congress, May 5, 1995.
May 10, 1995	U.S. Department of Energy, letter from Sterling Franks (DOE) to Document Control Desk (NRC) sending technical comments on the five contractor reports in the October 11, 1994 NRC letter, May 10, 1995. PSER
June 26, 1995	U.S. Nuclear Regulatory Commission, letter from Jack Donohew (NRC) to Ernest Condon (DOE) requesting a review of the draft of final MHTGR PSER (and one contractor report) for Applied Technology information, June 30, 1995.
July 17, 1995	U.S. Department of Energy, letter from Sterling Franks (DOE) to Document Control Desk (NRC) responding to NRC letter of June 26, 1995, stating that there is no Applied Technology information in the draft of the final MHTGR PSER, including the appendices with the contractor reports, July 17, 1995.

APPENDIX B

TECHNICAL DESCRIPTION OF THE MHTGR DESIGN

1. INTRODUCTION

The following technical description of the Modular High Temperature Gas-cooled Reactor (MHTGR) design was taken from the descriptions presented in the draft Preapplication Safety Evaluation Report, draft NUREG-1338, on the MHTGR design. The source document for this information on the MHTGR design is the Preliminary Safety Information Document (PSID) on the standard MHTGR design (DOE-HTGR-86-024) from the Department of Energy (DOE), the applicant for the MHTGR.

The safety objectives of the structures, systems, and components (SSCs) described in this appendix are given at the end of each section.

In this appendix, the staff discusses the following subjects:

- Section 2.0 Reactor
- Section 3.0 Vessel and Heat Removal Systems
- Section 4.0 Plant Arrangement and Containment
- Section 5.0 Plant Protection, Instrumentation, and Control Systems
- Section 6.0 Electrical Systems
- Section 7.0 Service Systems
- Section 8.0 Steam and Energy Conversion Systems
- Section 9.0 Radwaste System
- Section 10.0 References

2. REACTOR

The reactor, fuel, and core internals are described in Chapter 4 of the PSID.

2.1 System Characteristics

The reactor core will be supported in a steel reactor vessel. For normal plant operation and normal plant shutdown conditions, the design provides for downward forced helium flow through the annular core and surrounding reflector regions. A separate vessel, connected by a coaxial flow cross-duct vessel, will contain the steam generator and the other components of the Heat Transport System (HTS) which includes the main helium circulator and a helium flow shut-off valve. The reactor vessel will be above and off to the side of the steam generator vessel, negating natural circulation cooling of the core. This design reduces the ingress of steam or water to the core in the event of steam generator tube failures if the expected trip of the main helium circulator has been achieved, and protects steam generator tubing from damage from hot gas plumes from the core if feedwater flow to the steam generator is lost.

The reactor core subsystem (RCSS) is described in PSID Section 4.2 and consists of hexagonal, prismatic block-type graphite fuel and reflector elements, plenum elements, startup sources, and reactivity control material. The active core will be formed by the hexagonal fuel elements stacked in columns of ten fuel elements per column to form an annulus with equivalent internal and external diameters of 1.65 meters (65 inches) and 3.5 meters (258 inches), respectively. Each fuel element is 0.793 meter high (31.2 inches) by 36 centimeters (14.2 inches) across flats and contains blind holes for fuel compact rods and full-length channels for helium coolant flow. Corner holes will contain boron carbide lumped burnable poison (LBP) rods, dowel pins and sockets connect the fuel blocks axially, and a center hole accommodates a fuel handling tool. The stacked fuel and axial reflector columns will be supported from below by the graphite core support structure, and their lateral motion is limited at the top by close-fitting keyed connections provided by the upper-plenum elements.

The fuel elements will be of two types: the "standard" element and the similar, "reserve shutdown" element which provides for the insertion of pellets of boron carbide absorber material in a graphite matrix.

Similarly sized and replaceable graphite reflector blocks will surround the active core annulus. These will also be of two types: the "standard" and the "control," which allows insertion of a single control rod per element.

The coolant holes in both the standard and the reserve shutdown fuel elements and in the axial reflector elements will be 0.625 inch in diameter. These coolant holes will be interspersed among the half-inch-diameter fuel "compacts" or "rods" in the fuel elements. The design provides for conduction of the heat out from the fuel to the coolant channels, protection of the fuel compacts by graphite webbing, and a reasonably small overall core pressure drop (nominally 4.3 psi.). Except for the low enriched uranium (LEU) fuel composition, these fuel elements are the same as those used in Fort St. Vrain. The normal transit time of the helium from the top to the bottom of the core at full flow will be 0.3 second.

The annular core configuration was selected, in combination with a core average power density of 5.91 MW(t) per cubic meter, to achieve a thermal power rating of 350 MW(t) and to permit passive core heat removal while maintaining maximum fuel temperature below about 1600 °C (2912 °F) during certain event categories postulated in the PSID. The active core outer diameter was sized to maintain a minimum outer reflector thickness of 1.0 meter (39.4 inches). The reactor vessel has an inner diameter of 6.55 meters (258.0 inches). These dimensions will allow for a lateral restraint structure between the reflector and vessel which provides for both thermal expansion and seismic restraint. The inner core diameter was selected on the basis of studies on the reactivity worth of control rods with annular cores. To meet a 13-percent projected reactivity control requirement using reflector control rods (inner and outer), the annular width of the core can be no greater than 1 meter (39.4 inches). The core height is limited to 7.9 meters (311.0 inches) to allow a maximum power rating while assuring axial power stability to xenon transients over the entire burnup cycle.

Core reactivity is controlled by a combination of the fixed LBP in the fuel blocks, moveable poison, and a negative temperature coefficient. The moveable poison is in the form of metallic-clad, boron carbide control rods and boronated pellets which are a part of the neutron control subsystem. There are top head refueling penetrations that house the top entry, and gravity driven control rod assemblies that insert control rods into both the inner and outer reflector regions. No control rods enter the core directly.

Forced convection cooling, under normal and shutdown conditions, will be provided to the reactor by the main circulator (MC) of the heat transport system (HTS) in the steam generator vessel or under shutdown conditions only by the shutdown cooling system (SCS) in the reactor vessel below the core. For normal conditions the core will be cooled by helium leaving the MC at a temperature of 260 °C (497 °F) and a pressure of 64 bars (925 psia). The helium will pass through the outer annulus of the cross-duct vessel, up the outer annulus of the reactor pressure vessel between the core barrel and vessel in rectangular ducts, and then into the upper plenum of the reactor pressure vessel. The coolant then will flow downward into the steel plenum elements, the top reflector, the fuel elements in the active core zone, the bottom reflector elements, and the graphite core support blocks into the lower plenum. The hot core exit gas will begin to mix as it impinges on the graphite core support post structures, turns 90 degrees and then exits via the insulated hot duct pipe contained within the cross-duct vessel. The mixed core outlet temperature will be 690 °C (1268 °F) (vs. 785 °C for Fort St. Vrain). Approximately 90 percent of the helium coolant is expected to flow through the annular active core. The remaining coolant flow, considered the "by-pass flow," will flow through small gaps between the center and side reflector blocks, and through other miscellaneous channels and gaps within the core barrel.

2.2 Fuel Design

The MHTGR will use a low-enriched uranium/thorium (LEU/Th) fuel cycle which has an initial cycle length of 1.9 years. Subsequent burnup cycles will be 3.3 years with one-half of the active core being replaced each 1.65 years. This fuel cycle is predicted to achieve a design burnup of 26 percent fissions per initial metal atom (FIMA) while minimizing fuel cycle costs and ensuring a strong negative temperature coefficient of reactivity over all normal operations and abnormal temperature ranges.

Power will be tailored by fuel zoning. Each fuel zone will be loaded with different fissile and fertile concentrations, providing heavier concentrations of fissile material (uranium-235) in the higher power zones, but keeping the total core and reload fuel loadings unchanged. In the current proposed zoning scheme there will be three radial and three axial zones. The three axial zones will consist of layers that are five, three, and two fuel elements high in the top, middle, and bottom zones, respectively. The three radial zones will correspond to the three annular rings of fuel elements, i.e., 18, 24, and 24 columns of fuel elements per ring. This fuel zoning decreases the average power in the inner two fuel zones and increases the average power in the outer fuel zone so that the ringwise relative power densities of 0.87, 1.00, 1.10 are achieved and maintained over most of the operating cycle. The axial power fractions will be 0.65 for the top zone, 0.25 for the middle zone, and 0.10 for the bottom zone. These power distributions will ensure that a maximum fuel temperature of 1250 °C (2280 °F) is not exceeded during normal operations.

The MHTGR concept uses fuel particle, fuel element, and core designs derived from the Fort St. Vrain reactor, but the fuel integrity requirements and certain design details are different. The fuel safety objectives for the MHTGR are more demanding because the fuel particle coatings are considered by the safety analysis in the PSID to be the primary fission-product containment barrier.

Both the fertile material and fissile fuels are in the form of separate dense microspheres that are mixed within fuel compacts. The fissile fuel, identified hereafter also as the "reference" fuel, is formed into kernels of a two-component mixture of 19.9 weight-percent enriched uranium dioxide and uranium dicarbide, usually referred to as UCO, having an oxygen-to-uranium atomic ratio of 1.7. The fertile material is similarly formed into kernels of thorium dioxide. These kernels are coated from inside to outside with four successive protective shells, including a layer of silicon carbide that serves as the main fission-product barrier. This coating is known by the trade name TRISO. The fissile- and fertile-coated particles are blended and bonded together with a carbonaceous binder into the form of fuel "rods" or "compacts." Rods are inserted into the fuel holes that have been drilled through the graphite fuel blocks.

The fuel is designed to retain radionuclides within fuel particle coatings under all postulated conditions. The innermost shell surrounding the fuel kernel of the TRISO fuel particle is a buffer layer of a porous carbon. Next is a dense isotropic carbon layer known as the inner pyrolytic carbon (IPyC) shell, followed by a silicon carbide (SiC) layer and an outer pyrolytic carbon (OPyC) shell. In the DOE design, the overall particle diameters are 800 and 880 micrometers for the fissile and fertile particles, respectively.

The fuel kernel's ability to minimize fission-product release is dependent on kernel density, sphericity, diameter, and composition. Composition is important both for kernel-coating interaction problems and for potential fission-product attack on the coatings. The porous carbide buffer shell carbon attenuates fission recoils and, by virtue of its porous volume, acts to reduce fission gas pressure. The inner layer of dense carbon provides a smooth receptive surface for silicon carbide deposition and prevents chlorine ingress to the kernel during the silicon carbide coating process. The silicon carbide layer provides the major resistance to structural failure and to the transport of gaseous and metallic fission products. The outer carbon layer provides additional structural integrity and resistance to fission-product transport, and a bonding surface for the fuel rod matrix. The IPyC and OPyC layers are effectively impermeable to gases. DOE states that even with defective coatings, at normal operating conditions the fuel kernel will still retain more than 95 percent of the radiologically important, short-lived fission gases such as krypton-88 and iodine-131.

2.3 Nuclear Core and Control Subsystems

This section reviews the RCSS design given in portions of Section 4.2 of the PSID and the neutron control subsystem (NCSS) described in PSID Section 4.3. The NCSS ~~will~~ monitors and controls the neutron generation rate in the core, functions to control direct radiation exposure to operating personnel and serves the fuel handing system. Although the core configuration differs significantly from Fort St. Vrain, the NCSS is similar in concept and in many design features.

For the MHTGR, semi-articulated control rods with a stroke of about 30 feet can be inserted into six symmetric locations in the inner reflector and 24 symmetric locations in the outer reflector. During plant operation, the control rods will be withdrawn in groups of three symmetrically located rods. There are 2 inner control rod groups and eight outer control rod groups. There are 6 separate inner neutron control assemblies (INCAs) for the 6 inner control rods, and 24 separate drive assemblies for the 24 outer rods, 2 independently functioning drives clustered in each of the 12 outer neutron control assemblies (ONCAs). All of these assemblies will penetrate and be housed in the reactor vessel upper head. If needed to assure shutdown margin, 12 symmetrically located columns of reserve shutdown fuel elements, adjacent to the inner reflector, will contain an off-center, 3.75-inch-diameter hole to allow the insertion of borated graphite pellets by actuation of the reserve shutdown control equipment (RSCE). The RSCE will be part of the six INCAs and contains the pellets and release mechanisms in two hoppers within each INCA. Reactivity is also controlled by LBP rods in the corners of fuel elements as previously described. Nuclear instrumentation will consist of six ex-vessel neutron detector assemblies, three startup detector assemblies, and five in-vessel flux mapping units.

The safety-related and investment-protection-related shutdown objectives of the nuclear design are to provide for (1) effective hot shutdown by inherent feedback from the expected prompt and near-prompt negative temperature coefficients of reactivity and (2) the insertion of control rods and/or reserve shutdown material by the NCSS in response to trip signals from either the safety protection system, the investment protection subsystem, or the operator to bring the reactor ultimately to cold shutdown (i.e., refueling temperature of 192 °C). The NCSS also will have power operation objectives to control reactivity by the motion of control rods in response to the non-safety-related neutron flux controller or the operator. The inner control rods will be inhibited from entering the reactor following an automatic scram signal from operating power levels for investment protection purposes and must be manually activated to bring the reactor to cold shutdown or to maintain hot shutdown margins after xenon decay.

The inherent shutdown mechanism to hot standby is derived from the negative reactivity input characteristics of the uranium-238 Doppler coefficient with rising core temperature. When significant xenon-135 is present, subcriticality is achieved. This subcriticality is sustained for about 37 hours when equilibrium quantities of xenon are initially present. Hot shutdown for an initial, "clean" core, or following significant xenon decay, is said to result in a power level somewhat less than 1 percent of full power.

2.4 Thermal and Fluid Flow design

The essential features of the helium cooling design were described in Section 2.1 above. Emergency heat removal from the core is described in Section 3.5 below. The safety objective of the thermal and fluid flow design is to ensure for forced helium cooling that fuel and component temperatures can be maintained with margin for normal and transient design conditions and that fluid mechanical forces do not affect the structural integrity of the reactor.

2.5 Reactor Internals

The reactor internals will consist of an arrangement of metallic and graphite structures, together with certain insulating materials, that support and locate the graphite core and reflectors within the reactor vessel and protect the reactor vessel from high temperature helium and excessive neutron fluence. The metallic core internals will consist of the metallic core support structure (MCSS), the core lateral restraint (CLR), the upper plenum thermal protection structure (UPTPS) and the hot duct. The graphite internals are the permanent side reflectors (PSRs) and the graphite core support structure (GCSS). The reactor internals are described below.

Metallic Core Support Structure (MCSS)

The MCSS will be a 2-1/4 Cr-1Mo steel structure having the form of a spoked wheel that rests on a ring forging. This forging is integral to the reactor vessel and all major loads are transferred to the vessel through this support. The MCSS will support the other core internals and the reactor fuel, provide certain ducting for the reactor coolant, and maintain structural integrity during the postulated LBEs.

Core Lateral Restraint (CLR)

The CLR is a group of metallic structures, all Alloy 800H, located between the reactor vessel and the permanent graphite side reflectors. It is comprised of the core barrel, seismic keys, coolant channels, and the hot duct boss. The CLR is to make failure of this structure by either seismic or thermal means not credible in order to maintain geometry for conduction and radiation and for the insertion of movable poisons.

Graphite Core Support Structure (GCSS)

The GCSS will be an arrangement of posts and blocks which provide a lower plenum and a hot leg path for the primary coolant, and support of the core and reflector elements above the MCSS. The graphite posts and blocks are specified as Stackpole grade 2020 (or equivalent), a grade exhibiting high-strength and oxidation resistance. The GCSS will support the core and inner reflector elements, provide for helium exiting the core and entry to the hot duct, and ~~to~~ maintain structural integrity during the postulated LBEs.

Permanent Side Reflectors (PSRs)

The PSRs, formed by axial columns of keyed or pinned stacked graphite blocks (Stackpole grade 2020), will extend over the full length of the core and, except in the region of the hot duct, will extend to and are supported by alumina pads on the MCSS. The PSRs will provide radial restraint during all plant conditions, provide a conduction path for the removal of heat, and protect the reactor vessel and the core lateral restraint structure from excessive neutron fluence. Boron rods will be imbedded in the PSRs for this latter purpose.

Hot Duct

The hot duct will be an Alloy 800H pipelike structure that carries hot helium from the lower plenum of the core to the steam generator vessel. It will be located within the cross-duct vessel (see Section 3.2 of this appendix) and its exterior will be surrounded by coaxial flow of cold-leg helium from the steam generator vessel. For installation, removal, and maintenance, the hot duct will be formed of two straight (horizontal) sections and a curved elbow section (with expansion bellows) for vertical attachment to the steam generator. DOE has stated that the hot duct has no safety function.

Upper Plenum Thermal Protection Structure (UPTPS)

The UPTPS is designed to limit heat flow to the upper portion of the reactor vessel during the postulated spectrum of pressurized conduction cooldown events and serves in normal operation as the upper shroud for the core inlet plenum. Like the hot duct, it is made from Alloy 800H and is fitted with a similarly designed thermal barrier on its upper surface. The UPTPS is to protect the upper reactor vessel from excessively high temperatures during the postulated events.

3. VESSEL AND HEAT REMOVAL SYSTEMS

The vessels and heat removal systems are described in Chapter 5 of the PSID.

3.1 System Characteristics

The MHTGR vessel system (VS) and the two forced convection heat removal systems are discussed. The VS consists of the reactor vessel (RV), the cross-duct vessel and the steam generator vessel (SGV). The two forced convection systems are the heat transport system (HTS), contained within the SGV, and the standby cooling system (SCS), a separate system for decay heat removal in the bottom portion of the RV. The HTS and SCS have only non-safety-related functions and the only safety-related decay heat removal system is the passive Reactor Cavity Cooling System (RCCS). These systems are described below.

3.2 Vessel System (VS) and Subsystems

The VS consists of a RV and a SGV connected by a cross-duct vessel. The subsystems are the pressure relief system (located in the SGV) and the vessel support system. All of these systems with the exception of thermal insulation surrounding the cross-duct and steam generator vessels are classified by DOE as safety related.

The VS will use the same type of steel, SA533B, as used for light water reactors (LWRs). The dimensions of the reactor vessel will be 22 meters (72 feet) in overall height, an outside diameter of 6.8 meters (22.4 feet), and a wall thickness of 133 mm (5.25 inches), which are approximately the dimensions of a large BWR. The reactor vessel in order to function in a sustained conduction cooldown event (i.e., a loss-of-forced-cooling (LOFC) event) will be required to function at temperatures greater than the current code allowable value of 700 °F. An application to the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code) Committee, Section III, Division 1 has been made to extend the allowable temperature to 800 °F pressurized and 1000 °F depressurized. This is discussed in Section 4.2.8 of this report.

The pressure relief system is stated to meet the ASME Code and is similar to Fort St. Vrain except that in each of the two identical relief trains, a burst disc is downstream, rather than upstream, of the safety-relief valve. A block valve will precede the relief valve and an interlock system will prohibit the closing of both block valves at any given time. The pressure relief system will be located in the upper region of the steam generator vessel, downstream from the main loop shutoff valve (MLSV).

DOE stated that the overall safety objective for the reactor vessel is to meet a level of integrity comparable to LWR reactor vessels. Some of the differences between reactor vessel duty for MHTGRs and LWRs are listed in Table B.1 below (Table 5.1 of draft NUREG-1338).

3.3 Heat Transport System (HTS) and Subsystems

The HTS will consist of the steam generator (SG), main circulator subsystem (MCS), and the MLSV. These components will be located in the separate SGV. In normal operation, the HTS transfers energy from the reactor primary coolant (helium) to the secondary coolant (water), converting the incoming feedwater to superheated steam, to be sent to the steam turbine in the Energy Conversion Area (ECA). The steaming rate can range from 25 to 100 percent of the full-power feedwater-flow rate. During startup and shutdown, as well as during many postulated transients, the HTS could also serve to remove energy from the primary loop to achieve a relatively fast core cooldown and maintain a cold shutdown state, if required. The HTS could also operate without steaming, as required in some of these transients.

The normal flow path of the primary system will consist of hot helium from the core entering the SGV through the hot duct, the inner passage of the cross-duct vessel. Flow will be directed downward to the SG inlet plenum where it continues downward on the shell side of the steam generator tube bundles. At the SG outlet plenum, the cooled helium will be redirected upward through the annulus between the SG vessel and shell toward the MC inlet ducting. After passing through the MLSV, a flow-activated check valve, it will enter the MC and, after flowing downward again, be directed horizontally to the outer annulus of the cross-duct vessel and return to the reactor. The SGV outside

TABLE B.1 - DIFFERENCES BETWEEN MHTGR AND LWR REACTOR VESSEL DUTY

ITEM	MHTGR	LWR
1. Maximum Code Allowable Operating Temperatures, at operating pressure	800 °F[1]	700 °F
2. Maximum Code Allowable Operating Temperature, at ambient pressure	1000 °F[1]	______
3. Fluid	Helium	Water/steam
4. Intergranular Corrosion	Little or none	Susceptible
5. Hammer Effects	Little or none	Susceptible
6. Pressurized Thermal Shock	No potential	Susceptible
7. Failure Mode	Possibly Pneumatic	Hydrostatic
8. Neutron Fluence Characteristics	Lower irradiation temperature, harder spectrum	Higher irradiation temperature, softer spectrum
9. Total Neutron Fluence	Expected to be lower	Well known
10. Expected Frequency of Service Level C Occurrence per Plant Year for Pressurized Conditions	2.5×10^{-2}	$< 10^{-3}$
11. Expected Frequency of Service Level C Occurrence per Plant Year for Depressurized Conditions	3×10^{-3}	$< 10^{-3}$

1 The ASME Code has approved limited times of exposure not to exceed 1000 hours. This is discussed in Section 5.2.8 of draft NUREG-1338.

walls will be thermally insulated to minimize heat losses from the primary coolant. The main function of the MLSV will be to prevent damage to the SG system by providing limited bypass flow through the HTS when it is not in operation. A helium jet mechanism will be provided to enable its closure by operator action if necessary.

The SG will be a vertically oriented, cross-counterflow, shell-and-tube, once-through heat exchanger. The economizer-evaporation superheater (EES) section will be followed by the finishing superheater (FS) section, each consisting of 350 connecting tubes arranged in concentric helical coils, surrounded by shrouds and internal supports. The tubes will be of 22.2 mm in outside diameter with 3.3-mm wall thickness, substantially heavier than LWR tubes but slightly lighter than those in Fort St. Vrain. The EES tube section will be of Type 2-1/4 Cr-1 Mo steel, and the FS section is Alloy 800H. The bimetallic welds between EES and FS sections will be located in a quiescent region. Feedwater will enter the SGV at the bottom and be directed to a tube sheet from where it will flow upward through the helical tubing and exits as superheated steam through an upper, side-mounted tube sheet.

The main circulator subsystem will be located at the top of the SGV. It will include the MLSV and the MC, with its magnetic bearings, electric motor, and control and service module. The MC is a single-stage axial-flow compressor, driven by a variable-speed electric motor mounted on the same shaft; all are contained within the primary coolant boundary. The MC and its motor are to be fully floating on a set of active magnetic bearings, with a backup system of conventional anti-friction catcher bearings. Safety-grade trip logic and actuators will prevent operation of the MC when the rest of the HTS is shutdown.

The electric motor cavity will be kept at a pressure slightly above the HTS pressure by a continuous supply of purified helium. The heat exchanger that is to cool the motor winding and the magnetic bearings will also be located in this cavity and be water cooled. The water pressure will be below the primary system pressure to minimize the potential for water ingress from this source.

The safety objectives of the HTS are to prevent or minimize the following: (1) long-term degradation of or damage to fuel and other components by controlling temperatures within acceptable limits, (2) moisture levels that could result in fuel hydrolysis and the oxidation of graphite structural components, (3) challenges to the pressure relief system, and (4) the potential for a large steam ingress event.

3.4 Shutdown Cooling System (SCS) and Subsystems

The SCS is a non-safety-grade backup to the HTS described in the previous section. If the HTS is not available, the system cooldown will normally be performed by the SCS. During SCS operation, gravity and pressure forces (from the SCS) will cause closure of the MLSV, but a minor reverse flow through the HTS components (about 10 percent of SCS flow) will permit gradual cooldown of the SGV components as the reactor components are being cooled.

The SCS will remove decay heat under both pressurized and depressurized conditions, including refueling. It will consist of the shutdown cooling circulator subsystem (SCCS), the shutdown cooling heat exchanger subsystem (SCHES) and the shutdown loop shutoff valve (SLSV), located below the core support floor shield at the RV centerline in the bottom of the reactor vessel. The SCHES will be served by the shutdown cooling water subsystem (SCWS) which is composed of a single water cooling loop that serves all modules in the plant. Heat from SCWS will be rejected to the service water system (SWS).

During normal operation and during reactor cooldown by HTS, the SCS will be in a standby mode with the SCCS stopped and the SLSV in a closed position with a small coolant flow through the SCHES maintained to remove heat from a small flow through the SCS due to HTS operation. For initiation of the SCS cooldown mode, which will be automatic on signal of HTS shutdown, the SCWS coolant flow rate will be raised from its standby level of 15 percent of design flow rate to 100 percent flow. This will cause the SLSV downstream to open. The heat removal rate will then be controlled by varying the SCCS speed to maintain the SCWS outlet temperature of 232 °C, which corresponds to a peak cooling capacity of 23.7 MW. The SCS will be powered either by the normal or standby (non-Class 1E) electrical power.

In the SCS cooldown mode, hot helium will be drawn downward from the lower plenum through a central passage in the core support floor into the SCHES where it will continue downward over the helical coolant coils to enter the SCCS. From the SCCS, the helium flow will then be discharged through the SLVS to follow the normal coolant flow path to the upper plenum of the core and, hence, downward through the core to return to the lower plenum. With the pressure imposed by the SCCS, the flow path will be reversed through the cross-duct vessel and the SGV.

The safety objectives of the SCS are: (1) minimize the need for a long-term cooldown by the RCCS, during which reactor vessel temperatures become elevated and (2) the safety objectives listed in the previous section for the HTS, except for the fourth objective.

3.5 Reactor Cavity Cooling System (RCCS)

The RCCS is a safety-related, naturally convective, air-cooled structure designed to passively remove all the core decay heat from the reactor vessel surface, mostly by radiation, for all postulated LOFC events (i.e., when both the HTS and SCC are inoperable). The RCCS is the ultimate heat sink for the MHTGR.

There are no active components or moving parts in the design. The system will function at all times, including normal operation, and constantly removes about 0.8 MW from the uninsulated reactor vessel surface. Any operational problems are expected to become evident through degraded performance during normal operation. The RCCS is a closed system within the reactor cavity with the cooling panels serving as barrier between outside air and reactor cavity air. Radiation detectors provided in two exhaust ducts that monitor for increases in air activation are considered indicators of panel leakage.

The major thermal performance requirements stated by DOE for the RCCS are the following: (1) capability of maintaining the maximum fuel temperature below 1600 °C (2900 °F) during LOFC events, (2) for events other than sustained LOFCs maximum vessel temperature should not exceed 370 °C (700 °F), and (3) for LOFC events, the vessel temperature should not exceed 425 °C (800 °F) pressurized and 530 °C (1000 °F) depressurized. The RCCS will also be used to maintain the reactor cavity concrete at acceptable temperatures during both normal operations and conduction cooldown events. The principal mechanical design requirement, other than seismic or tornado design requirements, is that the panels will be designed to withstand differential pressures up to 10 psi for postulated over pressures in the reactor vessel cavity that could occur from depressurization events or feedwater or steamline breaks.

In the design of the panels, thermal insulation will be provided between the hot riser and the cold downcomer panels, as well as between the outlet (hot) air duct and the inlet (cold) air duct. Multiple and redundant ducting and flow paths, including those within the cooling panels, will be provided to ensure continuation of the cooling function in the case of single duct failure or flowpath blockages. A special "secondary chimney" design will be provided to address concerns of effects of high winds and regenerative heat transfer from the lower level inlet and outlet ducts.

The safety objective of the RCCS is to provide safety-related heat removal for the core and the reactor vessel during the loss of the HTS and SCS and during any accident.

4. PLANT ARRANGEMENT AND CONTAINMENT

The site buildings are described in Chapter 6 of the PSID.

4.1 Plant Arrangement

The plant will be divided into two distinct areas: the Nuclear Island (NI) and the ECA. Two buildings, the operations center (containing the control room) and the NI warehouse, as well as portions of the main steam and feedwater piping are part of ECA, but they interface with the NI. No portion of the ECA is proposed as "safety related protected" by DOE, including sources for cooling water, except for the control room.

The NI safety-related buildings will consist of four identical reactor buildings (RBs) (one for each reactor module), two identical reactor auxiliary buildings (RABs), and the reactor service building (RSB), all of which are mostly below grade. A steel-framed maintenance enclosure building with metal roofing and metal siding will shelter the entire operating floor formed by the at-grade slab covers over the four below-grade reactor buildings. Located on the north side of the RSB will be the NI cooling water building, personnel services building, and the radioactive waste management building. Also, part of the NI will be the free-standing helium storage building and the two liquid nitrogen enclosures separately adjacent to the east sides of each reactor service building.

The RSB will house facilities, systems or components shared by all four reactor modules. These will be the new fuel storage area, fuel handing machinery, a fuel sealing and inspection facility, and a hot service facility; provision will be made for the storage of activated or contaminated nuclear steam supply system (NSSS) components (e.g., helium purification filters, control rods, SCS circulators). The RSB will also house the remote shutdown area (RSA), portions of the safety-related essential dc and essential uninterruptable electrical power supply systems, and the PPIS. An at-grade washdown bay on the west wall of the maintenance enclosure will provide for the cleaning of incoming fuel casks and equipment and for the decontamination of outgoing spent fuel casks. Railroad and truck access to the RSB will be through the washdown bay. A 125-ton bridge crane will serve this and other areas of the RSB, as well as the two RABs and the four RBs. The two RABs will be spaced between each of two of the four RBs. These identical buildings will contain a spent fuel storage pool, house portions of the essential power supply systems, and provide for occupation and routine offsite radiation control.

The non-safety-related and the grade-level personnel service building will house the fuel handing control station and provides facilities for dealing with radioactive materials and personnel and equipment decontaminations (e.g., a hot chemistry laboratory, decontamination facilities, laundry and clothing storage). It will also house locker rooms, a health physics laboratory, a chemistry laboratory, and the supervisor's office.

The safety objectives for both the RSB and the RABs are to protect the safety-related equipment that they house from various internal and external hazards, permit refueling and other safety-related operations to be performed to standards equivalent to those for LWRs, and to provide occupational exposure control in the generally accessible areas to no more than 1.0 mrem per hour during all modes of normal plant operation for times of at least 40 hours per week.

4.2 Reactor Building (RB) and Containment

The RB will be predominantly a multi-cell, reinforced-concrete structure, set below ground. The lower cylindrical portion, or silo, will contain the RV and SGV with all related components. The portion containing the reactor is known as the reactor cavity and will house the RCCS panels, the RCCS header, and some RCCS inlet and outlet ducting as well. The upper rectangular prism portion will house most of the helium purification system (HPS) equipment, PPIS equipment and other auxiliary systems, and includes the additional portions of the RCCS ducting and portions of the vent paths for overpressure releases. The above-ground portions of the RB will be the RCCS intake and exhaust structures, terminal portions of the vent paths, including the fixed louvers, and the main steam isolation and relief valve enclosures.

The silo will extend from elevation -10.67 to -46 meters, with an 18.3-meter inside diameter and a 0.9 meter thick wall. The internal walls that divide the silo into multiple cells will be of varying thicknesses, depending on shielding and load requirements. The two major cells of the silo will house the reactor vessel and the steam generator vessel, with a 1.5-meter concrete wall separating these two cells, except for penetration by the cross-duct. The reactor cavity is normally isolated from the rest of the RB to limit argon-41 release and to reduce the heat load on the heating, ventilation, and air conditioning (HVAC) system. The top slab of the RB, at grade, will have several hatchways for equipment access which are normally closed with concrete plugs. This upper slab will provide biological shielding as well as protection from external hazards.

The upper part of the RB will be generally accessible during normal operation. To permit access of at least 40 hours per week, radiation levels in this area are restricted to 1 mrem per hour. The silo portion of the RB will only be accessible at some time following shutdown. DOE stated that the RB will conform to the user requirement of average plant population exposures of no more than 10 percent of the 10 CFR Part 20 limits.

The RB does not provide a pressure-retaining containment building, such as is used in conventional LWRs, but instead provides for controlled venting. Small releases will be filtered and contained by the HVAC system as in normal operation, but for larger primary system releases, or steamline or feedwater line breaks, vent pathways to the environment would provide overpressure protection for the RB and its contents.

A large primary coolant (helium) release from the RV would open the blowout panels between the reactor and steam generator cavities. A primary coolant or steam discharge from the steam generator cavity would also result in coolant release into the steam generator cavity. From there, gases and vapors would flow thorough side cavities of the silo, and through the hinged louvers, follow an up-and-down path through upper portions of the RB, and discharge to the atmosphere through the above-ground fixed-open louvers. To contain minor gas releases in the steam generator cavity, the hinged louvers are located between elevations -7 meters and -10.67 meters. These louvers are normally closed by a pressure differential of about 1 inch of water, provided by the HVAC system. Only in the event of internal pressure buildup would these louvers open.

The safety objectives for the RB are to ensure adequate structural support for the safety-related SSCs it houses, to ensure protection of the RCCS, and to provide some retention of radionuclides during accidents. The RB must maintain geometrical integrity of the vessel system and the RCCS, and protect itself and its contents from seismic loads, other external events, and internal pressures. Further, the RB must provide for continued operation of the PPIS and NCSS.

5. PLANT PROTECTION, INSTRUMENTATION, AND CONTROL SYSTEMS

The plant protection, instrumentation, and control systems are described in Chapter 7 of the PSID.

5.1 General Description and Design Process

The plant protection, instrumentation and control system (PPICS) provides for fully automatic control of the four reactor modules and two turbine generator systems that constitute the MHTGR power plant. Automatic control will be used for both normal operations and abnormal events. The goal of the PPICS is to maintain power generation, protect plant investment, avert challenges to the safety system, and to cope with the postulated events without manual operations. The plant safety protection function is performed by separate safety-related instrumentation and control equipment. The multimodule plant is controlled from a single main control room (CR), within the operations centers of the ECA area, and limited control functions (primarily safety functions for each reactor module) are available at a remote shutdown area (RSA), within the RSB of the NI area.

Three separate systems provide plant protection and automatic control for the MHTGR. These are the PPIS, which is partially safety related, and the following two non-safety-related systems: the plant control, data, and instrumentation system (PCDIS), and the miscellaneous control and instrumentation group (MCIG).

5.2 Plant Protection and Instrumentation System (PPIS)

The PPIS is an independent system of hardware and software from the balance-of-plant instrumentation and control system. It is provided to indicate plant status and to actuate automatically both the safety-related control system to protect the public and the control systems to protect the plant investment. It monitors selected process variables, compares the sensed values to preselected levels, and, as required, commands and initiates predetermined corrective actions. The PPIS subsystems are (1) the safety protection subsystem (SPSS), (2) the special nuclear area instrumentation subsystem (SNAIS), and (3) the investment protection system (IPSS).

The SPSS contains the safety-grade equipment of the PPIS. It provides the sensing and command features necessary to initiate a reactor trip using the outer control rods and the RSCE, and to initiate the reactor main loop shutdown.

The SNAIS provides certain plant protection interlock and monitoring features. This will include the closure interlock for the VS pressure relief block valve, equipment that monitors the status of plant protection systems, and equipment that monitors plant safety and investment under normal operating and accident conditions. As proposed, the SNAIS will contain only equipment that is not safety related.

The IPSS provides the sense and command features necessary to initiate protective actions to limit plant investment risk. These actions will include reactor trips, SG isolation and dump, and initiation of the SCS. It will not contain safety-related equipment.

The SPSS functions of the PPIS will be implemented on a per-reactor basis with a fully automatic, remote-multiplexed, microprocessor-based protection system. The protection system architecture will consist of multiple separate and redundant optical-digital-data highways from the local multiplex units that communicate with four separate, redundant computers to implement the four-channel protection systems for each reactor module.

Separate and independent SSPS operator interfaces for each reactor module will be located in the PPIS equipment room, within the RB, and in the RSA. The operator interfaces include color video displays, function input devices, and keyboards. Because DOE proposed that no operator action is required for safety, these interfaces are not classified as safety related. These operator interfaces are provided as part of the PPIS, and they are separate and independent of all other plant instrumentation and controls.

In addition, data on the SPSS will be transmitted through a unidirectional isolator to the data management subsystem (DMS) for a display by the plant supervisory control subsystem (PSCS) in the CR. The PPIS operator interfaces in the PPIS equipment room and the RSA will give an operator the capability of initiating reactor trip or main loop shutdown from a position remote from the CR. No manual inputs to the SPSS (i.e., safety-related) will be provided from the CR; however, the reactor can be shut down with normal plant control equipment (i.e., non-safety-related) from the CR, as discussed in Section 5.3 below.

5.3 Plant Control, Data, and Instrumentation System (PCDIS)

The PCDIS consists of instrumentation and control hardware and software that automatically control the MHTGR plant from startup to full power and return it to a shutdown condition. The PCDIS subsystems are (1) the PSCS, (2) the NSSS control subsystem, (3) the ECA control subsystem, and (4) the DMS.

The PSCS automatically supervises and coordinates balancing of load (power) levels among the energy production areas, namely the NSSS and the ECA of the balance of plant (BOP). There will be individual NSSS control subsystems for each reactor that control reactor conditions and the supply of steam to the main steam header in response to PSCS load demands. The BOP will provide monitoring and control for those systems that directly impact the continuity of power generation. The computer-based DMS will provide plantwide communication and centralized data processing. The DMS will support the PCDIS by transmitting control and monitoring communications between subsystems.

5.4 Miscellaneous Control and Instrumentation Group (MCIG)

The MCIG will sense, acquire, and process various data from the plant for display to the plant operator and/or retention for historical purposes. The MCIG subsystems are: (1) the NSSS analytical instrumentation system, (2) the radiation monitoring system, (3) the seismic monitoring system, (4) the meteorological monitoring system, and (5) the fire detection and alarm system. These subsystems are considered to not perform any safety-related functions and will not have any inter-tie with safety-related instrumentation and control systems.

6. ELECTRICAL SYSTEMS

The electrical power systems are described in Chapter 8 of the PSID.

6.1 Overall Design

The electrical systems will consist of the essential uninterruptible power supply (UPS) system (EUPSS), essential dc power system (EDCPS), offsite power and main generator transmission system, nonessential ac distribution system, non-essential UPS system, nonessential dc power system, grounding, lightning protection, heat tracing and cathodic protection, communication systems, and lighting and service power systems.

DOE proposed to place minimal safety-related requirements on the electrical systems because the few safety-related plant systems require very little power to perform their functions. Only the EUPSS and the EDCPS are considered to be safety related. The safety objectives for the essential electric power systems are proposed to be met without the large offsite and onsite power supplies required for LWRs (i.e., no emergency power diesel generators), and should be satisfied by onsite battery supplies and associated power conversion and distribution equipment.

6.2 Essential Uninterruptible Power Supply System (EUPSS)

The EUPSS will be designed to be a reliable electric system consisting of four redundant and independent channels, each with adequate capacity, capability, and reliability to supply power to the plant essential loads. It will include regulated, battery-backed power for four redundant and independent 120-V ac vital buses that feed essential control, instrumentation, and plant protection circuits for all four reactor modules.

Each EUPSS channel will be designed to normally provide uninterruptible 120-V ac power from the ac distribution system through a rectifier-inverter assembly. Backup power will be provided from the EDCPS through the inverter, and alternate ac power will also be provided from the ac distribution system through a regulating transformer. Essential 120- V ac power will be supplied to safety-related equipment within the PPIS and to some equipment not related to safety.

The four EUPSS channels will serve all four reactor modules and each module will have all four EUPSS channels (e.g., plant EUPSS channel A will serve all four reactor module PPIS loads that require EUPSS channel A power). Each channel consists of one rectifier-inverter assembly, a static transfer switch, a manual bypass switch, a regulating transformer, and a vital bus distribution panel. Each rectifier-inverter assembly will be provided with a normal ac power supply from a nonessential motor control center. The rectifier converts ac power to dc power, which is fed to the inverter which, in turn, converts dc to ac.

6.3 Essential DC Power System (EDCPS)

The EDCPS will consist of a 125-V dc, two-wire, ungrounded system of four batteries, four operating and four spare battery chargers, four distribution switchboards, and several distribution panel boards that comprise the four completely independent and redundant channels, each serving redundant essential dc loads. The four plant dc channels will serve all four reactor modules and each module will have all four dc channels (e.g., plant dc channel A will serve all four reactor module dc loads that require dc channel A power). Each channel has a normally operating battery charger that will rectify three-phase 480-V ac received from a nonessential motor control center to 125-V dc. A backup battery charger, fed from a separate non-essential motor control center, is proposed so that any unit can be removed from service without degrading the systems to which dc electric power is provided by each channel.

The battery chargers will normally supply dc power to the 125-V dc distribution switchboard loads and maintain the essential batteries in a fully charged state to provide a float charge; they will be capable of recharging the channel batteries within 12 hours from a fully discharged state. In the event of loss of all nonessential ac power, essential dc power is proposed to be provided from the batteries for at least 1 hour.

The safety objectives for the EDCPS are the following:

- supply power to the rectifier-inverter assembly, and the EUPSS which, in turn, supplies power needed by the SPSS of the PPIS to sense any upset conditions and initiate appropriate remedial actions, such as a reactor trip or main loop shutdown.

- shut the steam generator isolation valves (SGIV) to limit the total amount of water or steam available for ingress following a steam generator tube leak.

- actuate the RSCE, which can dump boronated pellets into the core, for failures of control rod insertion or large moisture ingress events.

- supply power to the battery room exhaust fans.

6.4 Nonessential Electrical Power Systems

The nonessential electrical power systems are the following:

Offsite Power and Main Generator Transmission System

The offsite power transmission system will consist of two physically separate and independent circuits from the transmission network which, through a common switchyard, will supply power to the onsite distribution system. The main generator transmission system will consist of two generators that transmit power to the grid through two transformers from the common switchyard.

Nonessential AC Distribution System (NEACDS)

The NEACDS will provide 4160-V, three-phase, and 480-V or less, three-phase and single-phase, 60 Hz electric power to electrical switchgear associated with each generator to feed the plant's auxiliary equipment and services. It will be normally fed from each generator through each auxiliary transformer unit. For plant startup, each generator's buses will be fed from the grid through startup auxiliary transformers. Two backup nonessential diesel generators will supply selected loads in case of loss of ac power.

Nonessential Uninterruptible Power Supply System (NEUPSS)

The NEUPSS will provide 120-V ac, single-phase, 60 Hz, electric power to the plant's control and instrumentation loads connected to the two 120-V UPS buses, each of which is associated with a single turbine-generator unit.

Non-Essential DC Power System (NEDCPS)

The NEDCPS will provide 125-V dc electric power to the plant's control and instrumentation loads connected to the two 125-V dc buses, each of which is associated with a single turbine-generator unit.

7. SERVICE SYSTEMS

The service systems for the MHTGR are described in Chapter 9 of the PSID. Only selected services systems are discussed below.

7.1 Fuel Handling and Storage

The fuel handling machine and the fuel transfer system are essentially an extension and further development of Fort St. Vrain technology that will be tailored for application to the MHTGR design: a steel vessel, a different radial core and control assembly (access) arrangement, and a taller core. Spent fuel will be stored in dry, helium-filled wells surrounded by water in one of two spent fuel storage pools; each pool will be contained in one of the two RABs. Decay heat will be transferred from the pool to the service water system by means of a closed loop with two 100-percent-capacity heat exchangers and four 50-percent-capacity pumps. Passive backup cooling is provided by pool boiloff, and water will be replaced by a makeup water supply.

The safety objectives are to avoid exceeding the dose limits of 10 CFR Part 20 by containment of fission-product contamination on the fuel elements and by avoiding fuel damage due to either structural challenges or overheating by decay heat.

7.2 Helium Purification System (HPS)

The HPS is essentially an extension and a further development of Fort St. Vrain HTGR technology that will be tailored for applicability to the MHTGR design. The experience with the Fort St. Vrain HPS has been good from the standpoint of both performance and reliability. The HPS will purify a helium side stream from the reactor primary coolant system, and will remove both oxidants and radioactive contaminants in drying and purifying the helium. The HPS will provide purified helium on a continuous basis to the buffer seals of both the HTS and SCS circulators, and will purify helium routed to storage during controlled depressurizations.

The HPS equipment for each module will be housed mainly in the RB and will consist of a high-temperature adsorber/filter section for iodine and particulate removal, an oxidizer/cooler and dryers, a low-temperature adsorber (LTA) for removing noble gases, and a purified helium recirculator compressor. There will be a separate regeneration train for the dryers and the LTA which will service two modules. Both the HPS train and the regeneration system are designed to operate at full primary system pressure. Liquid nitrogen for cooling of the LTA will be provided to the HPS by the liquid nitrogen system (LNS), with one LNS will serve two modules.

The HPS has the following three safety objectives: (1) remove oxidants from the primary coolant system and maintain chemical impurities to less than 10-ppm total oxidants, (2) provide a direct radionuclide control function by maintaining the concentration of radionuclides in the primary coolant at acceptably low levels so as to satisfy the 10 CFR Part 100 release criteria in the event of depressurization of the reactor vessel, and (3) provide a manually actuated means for emergency depressurization of the primary system to augment safety margins relating to reactor vessel integrity at elevated temperatures.

If the HPS for a given module is out of service, manual cross-connect valves will permit the use of an HPS from another module for an alternate depressurization pathway. The alternate HPS could also be used in parallel with the normal one to handle any loads that were higher than expected.

7.3 Liquid Nitrogen System (LNS)

The LNS is essentially an extension and further development of Fort St. Vrain reactor technology that is tailored for application to multiple reactor modules. It will provide liquid nitrogen to cool LTAs in the HPS and for use in various instruments in the NSSS analytical instrumentation system. It is designed to run continuously during both normal plant operation and refueling. Each of the two independent trains of the LNS will serve two reactor modules. Makeup of liquid nitrogen to the phase separator and storage tank will be provided by running, as required, one (during normal operation) or both (during depressurization events) of the nitrogen-recondenser compressors. The peak load during the initial stages of a depressurization can be accommodated without a second recondenser by using the excess storage capacity in the phase separator and storage tank. There are full-capacity liquid nitrogen pumps, and one will serve as a backup. These backup components can be isolated during normal operation for service or replacement.

7.4 Reactor Plant Cooling Water System (RPCWS)

The RPCWS will remove waste heat from the following reactor plant components: (1) HPS coolers and compressors, (2) HPS regeneration coolers and compressors, (3) MC motor of the HTS, (4) moisture monitor compressor modules, (5) neutron control assemblies (NCAs), and (6) miscellaneous components. The waste heat is rejected by means of a heat exchanger to the SWS. The RPCWS components are located in the nuclear island cooling water building (NICWB), with piping routed from there to various heat sources. The system employs two parallel 100-percent-capacity heat exchangers and two 100-percent-capacity pumps. The system is kept pressurized at 160 psi by a helium blanket in the surge tank. During normal plant operation, the RPCWS will run with one pump and one heat exchanger, the remaining components being normally on standby. The system is shut off, isolated and depressurized during plant shutdown. Primary control of the RPCWS is accomplished from a local panel in the NICWB, with process variables also being available in the CR.

In case of failure of either one pump or one heat exchanger of the RPCWS the corresponding backup component would be used to maintain RPCWS performance. If the backup component fails, or is not available, the plant would have to be shut down.

7.5 Heating, Ventilation, and Air Conditioning (HVAC)

The NI HVAC system will provide for equipment operability, personnel comfort and the monitoring/and filtering of any potentially radioactive atmospheres. Once-through conditioned supply air is provided for the following buildings: (1) the accessible portion of the RB, (2) the RAB, (3) the RSB, and (4) the personnel services building (PSB). The radioactive waste management building (RWMB) will have an air conditioning system similar to the above areas, except that its exhaust will be filtered continuously. The HVAC system for the RB and RAB will employ two parallel redundant trains for each set of two reactor buildings.

All areas will include monitoring of radiation levels in the exhaust stacks and automatic diversion from direct exhaust to exhaust through filters trains, which provide a prefilter, a high-efficiency particulate air (HEPA) filter, with room for further filters to be included, as required. The air will always be directed to flow from areas of low potential for contamination to areas of higher potential. Negative pressure control will be achieved by manual adjustment of inlet guide vanes to the exhaust fans of potentially contaminated areas.

Only ventilation and heating will be provided for the NI maintenance enclosure, the cooling water building, the liquid nitrogen enclosures, and the helium storage structure. Air intake in these areas will be through wall louvers, with exhaust through power-driven roof ventilators. Supplemental heating will be provided by units heated by hot water. Special and additional provisions will be applied to the reactor and steam generator cavities, as well as to other areas containing safety-related and/or other sensitive equipment.

During normal operation, the reactor cavity will be isolated and will not be cooled by the HVAC system because the RCCS will function to maintain thermal equilibrium conditions in this cavity. During shutdown the cavity will be cooled by a separate unit cooler with its own intake and exhaust units. Conditioned, once-through air will be provided during shutdown when access is needed. The steam generator cavity will be cooled during normal operation and during shutdown by its own closed-cycle unit cooler. If access were required, once-through air flow could be provided. Rooms containing other safety-related equipment and/or equipment significant to the protection of public health and safety will have separate unit coolers using chilled water. These unit coolers will also maintain the relative humidity at less than 50 percent. The NI HVAC system will be controlled from the CR, and local control will be possible from control panels near the respective fans. The primary functions of the HVAC system will be to maintain all equipment operable and to provide for personnel access as required to maintain power production and to control personnel radiation exposure.

For routine operation, the filtering system of the HVAC will be designed to meet the routine offsite release limits of 10 CFR Part 50, Appendix I, and the occupational doses limits of 10 CFR Part 20. In general, loss of the HVAC system would be cause for an orderly shutdown of a reactor module or of the plant, depending on the degree of failure.

7.6 Plant Fire Protection System (PFPS)

The PFPS will be designed to rapidly detect, control, and suppress fires. It will contain automatic detection systems, manual fire hose stations, portable fire extinguishers, and automatic water, carbon dioxide and Halon subsystems. The PFPS will interface directly to the NI and protect those SSCs which to protect the public health and safety. As backup, the PFPS will have independent, motive power that will be available during abnormal operating occurrences, including loss of all ac power. The PFPS subsystems will be the following:

Plant Fire Protection Water Subsystem (PFPWS)

The PFPWS will consist of two fire pumps and controllers. The primary pump will be electrically driven, the backup pump will be diesel driven. The diesel driven pump will have a battery-powered starting system and a gravity fuel oil feed from an eight-hour day tank. Two fire water storage tanks, each of 300,000 gallon capacity, will feed the pumps. Each pump will be separately connected to an underground fire water loop which will encircle the NI and supply water to the yard hydrant and hose house system; there will be several fire protection piping systems within the plant's structure.

The underground fire water loop will supply water to the NI. The NI portion will consist of yard fire hydrants, water sprays, deluge and wet pipe sprinkler systems, and wet standpipe fire hose stations. Standpipes and hose stations for safety-related buildings will be connected to the water main in the yard, independent of the connection to the non-safety-related, fixed, water suppression system serving the same fire area.

Plant Fire Protection Carbon Dioxide Subsystem (PFPCDS)

A total-flooding PFPCDS, designed for double-shot capability, will deliver carbon dioxide to the turbine-generator bearing and enclosure areas. It will consist of a low-pressure, refrigerated, carbon dioxide storage tank, piping, nozzles, and controls for master and selection valves, and detection and audio alarms. Carbon dioxide will not be used on the NI.

Plant Fire Protection Halon Subsystem (PFPHS)

The PFPHS is designed for double-shot discharge capability and will protect electrical panel areas and local control rooms in the operations center, RB, and ECA buildings and structures. The subsystem will consist of dedicated main and connected reserve cylinder banks, manifold piping, applicator nozzles, and detection and audio alarms.

Plant Fire Detection and Alarm System (PFDAS)

The PFDAS will detect and annunciate the presence and location of fire and/or combustion byproducts. It will be used in and around SSCs required for the protection of health and safety of the operating staff. Other areas that will be protected by the PFDAS are those in which radioactive materials will be handled in the RB, RAB, RSB, and PSB. The non-Class 1E UPS will permit the fire detection and alarm system to be operational during loss of all ac power.

8. STEAM AND ENERGY CONVERSION SYSTEMS

The secondary-side steam and energy conversion systems are described in Chapter 10 of the PSID.

8.1 Main Steam and Feedwater Supply Systems (MSSSs)

The primary function of the MSSSs will be to convey feedwater to the SGs from the condenser, and superheated steam from the SGs to the turbine-generators. Isolation valves are included in the design so that any one of the four SGs in a four-reactor module plant can be isolated from the others in the event of a tube leak. Additional valves in the main steam bypass system are also included in the MSSSs in order to control the flow of main steam to the turbine-generators during startup or whenever the turbine is off-line.

The feedwater supply system (FWS) will supply water from the condenser to the economizer inlet of the SGs. Condensate from the condenser will be normally pumped first through a polishing demineralizer to adjust water chemistry, then through the feedwater heaters to the deaerator. The feedwater will then be pumped at high pressure to the SGs. Valves are included for SG isolation, and to connect the FWS to the turbine bypass desuperheater and the steam and water dump tank.

Neither the main steam supply (MSS) nor the FWS will have a direct safety-related reactor cooling function because the RCCS will be (1) the ultimate heat sink for decay heat removal and (2) completely independent of the MSSSs and FWS .

8.2 Startup and Shutdown (SU/SD) Subsystem

The SU/SD subsystem will be a dedicated system of piping, pumps, valves, equipment, and tanks that is independent of the power operation systems for both the feedwater and steam cycles. Its function will be to provide for smooth operational transitions for a reactor module in the 0 to 25 percent (or 25 to 0 percent) power range. It will be sized and designed to operate for a single module, when the other modules are either in operation or shut down. In the case of simultaneous SU/SD of multiple modules and turbines, the main deaerators and feedpumps will be used in conjunction with the SU/SD subsystem.

The SU/SD subsystem will deliver feedwater to the SG and steam to the turbine at the desired temperature, pressure, and flow, and within prescribed water chemistry limits. It will not perform any safety-related functions and is not classified as safety related. In case of failure of part or all of the SU/SD subsystem, reactor cooldown could be achieved by various other non-safety-related systems or by the safety-related RCCS.

8.3 Steam and Water Dump System (SWDS)

The SWDS will accept and contain the inventory of an SG in the event of a tube leak. Since the normal operating pressure of the secondary coolant will be greater than that of the primary, a tube leak will provide a path for steam and water to be introduced into the reactor core. The function of the SWDS is to limit the introduction of steam and water into the primary coolant, both to minimize damage to the core by fuel hydrolysis and graphite oxidation, and to prevent excessive pressurization of the primary system.

The SWDS will be actuated by non-safety-related portions of the PPIS when a high level of moisture is detected in the primary coolant. The SWDS will contain the mass and energy inventory of the SG, as well as any primary coolant that leaks into the SWDS through a tube rupture. Since the primary coolant will have radioactivity, there will be piping connecting the SWDS with the gaseous and liquid radioactive waste system (GLRWS) to ensure that no primary coolant and radioactivity is released directly to the environment.

The SWDS will serve each SG independently. The portion of the subsystem associated with each SG will consist of a dump tank, two trains of dump valves, a drain pump, and piping and valves interconnecting with the GLRWS. The SG could be isolated by two power-operated valves mounted in series on each inlet and outlet of the SG. Dumping will be executed by two parallel trains of dump lines, each equipped with two dual-actuated motor-operated valves mounted in series.

The following instrumentation will be provided for each of the four subsystem loops at the system control station in the RB and in the CR: (1) dump tank pressure, (2) dump tank temperature, (3) dump tank level, (4) dump valve position (four), (5) main steam isolation valve position (two), (6) main feedwater isolation valve position (two), and (7) a radiation monitor.

The SWDS has important safety objectives during a steam generator leak transient. It will limit the amount of chemical damage to the core and positive reactivity that can be inserted due to water ingress into the core, and, during a postulated tube leak, the valves and piping of the SWDS may carry radioactivity from the primary coolant and act as a pressure boundary for the primary coolant. However, DOE stated that the SWDS is not needed to ensure core cooling or to control radionuclides during accident, and did not classify it as a safety-related system.

Features of the SWDS's subsystems are as follows:

Dump Tank

The carbon steel tank will contain the mass and energy inventory of the SG in its loop, and must be sized accordingly. It will be protected from overpressurization from the feedwater by a safety valve that has a pressure relief setpoint higher than that of the primary coolant safety valves. The SG inventory will be introduced into the dump tank through a sparger into an existing pool of water present to quench the incoming steam.

Dump Valves

The two dump valves in each of the two parallel trains will be motor operated when called upon by the PPIS and will open immediately after the main steam and feedwater isolation valves are closed in order to isolate the leaking steam generator from the remainder of the secondary coolant loop. These dump valves will then be closed after the SG has emptied its inventory.

Drain Pump and Connecting Piping

The drain pump will receive the liquid from the bottom of the dump tank and pump the liquid through the connecting piping to the GLRWS. The connecting piping for the gases in the dump tank will lead directly to the GLRWS.

8.4 Service Water System (SWS)

The SWS will remove waste heat from non-safety-related process systems located in various buildings on the NI and will convey the waste heat loads to the cooling tower. The system is designed to support non-safety-related normal operation and shutdown cooling of structures, systems, and components employed in the power generation processes. The SWS will originate at the cooling tower basin where two 100-percent capacity service water pumps are available for circulation to remove normal process heat from the reactor plant.

9. RADWASTE SYSTEM

The liquid, gaseous, and solid radwaste systems are described in Chapter 11 of the PSID. The liquid and gaseous radwaste systems designs in the PSID are similar to those systems at Fort St. Vrain, but the solid radwaste system is different. The radwaste systems are not designed to mitigate the consequences of DBAs and, thus, are not classified as safety related.

9.1 Liquid Radwaste System (LRWS)

The LRWS will consist of equipment, independent of the HPS, to collect, treat, and store liquid radioactive wastes generated in the plant. The equipment will include three waste receiver tanks, three transfer pumps, two filter/demineralizers, two test tanks, two test tank pumps, and piping with controls. The LRWS will be located in the Radioactive Waste Management Building (RWMB), and will be shared among all four reactor modules. Sources of liquid waste will be the condensate and regeneration waste from the HPS, decontamination waste, and liquids from drains and the steam water dump tank.

Liquid radwaste will be managed by segregating it into the following two categories: (1) low conductivity and also low in radioactivity wastes and (2) high in conductivity and high in tritium waste. The former will be collected in receiver tanks, processed through a filter/demineralizer, and transferred to a test tank. The waste will be then sampled and, if within Technical Specifications effluent specifications, will be discharged from the site.

The high conductivity, or decontamination, waste will be neutralized, if necessary, and then transferred to the waste solidification area for packaging or processed through a filter demineralizer to a test tank.

9.2 Gaseous Radwaste System (GRWS)

The GRWS will collect and treat all radioactive and potentially radioactive gaseous waste within the plant, excluding the leakage from reactor modules and other equipment in the RB. The system will be shared among all four reactor modules. The gaseous wastes will be segregated into low-level and high-level waste streams for processing prior to being discharged to the environment.

Low-level radioactive gases will be filtered, monitored, and released to the RB ventilation system. The process path will consist of redundant HEPA/charcoal filtration units for particulate and radioiodine removal, redundant exhaust blowers, and a monitor on the discharge from the filters. In the event of high-level activity, this gas flow is diverted to the high-level process path.

High-level radioactive gases will be held in temporary storage to allow decay before being discharged to the RB ventilation system. The process path will consist of the waste gas vacuum tank, two compressors, and three waste gas surge tanks. The high-level gases will be compressed from the vacuum tank into the surge tanks for holdup and decay.

9.3 Solid Radwaste System (SRWS)

The SRWS will be designed to provide holdup, solidification, packaging, and storage facilities for radioactive materials before they are shipped offsite for disposal. The system will be shared among all four reactor modules.

The SRWS will consist of a shielded and an unshielded storage area, a dry waste compactor, a resin transfer tank and pump, a drum roller for mixing wastes and solidification agents, and an industrial robot for cutting up non-compactible waste. Materials processed will include high-conductivity decontamination wastes, highly tritiated liquids, spent resins, spent filter cartridges, high-temperature filter units from the helium purification system (HPS), low-level compressible wastes (e.g., rags, paper, clothing), HEPA and charcoal filtration units, and miscellaneous solid materials that become radioactive, as tools.

APPENDIX C

DOE PSER TRACKING SYSTEM FOR DRAFT NUREG-1338 ISSUES

1. Introduction

In its letter of July 31, 1991, the U.S. Department of Energy (DOE) submitted a copy of the DOE/HTGR (High Temperature Gas-Cooled Reactor) Program's "PSER Issues Tracking System." The PSER is the draft NUREG-1338, "Draft Preapplication Safety Evaluation Report for the Modular High-Temperature Gas-Cooled Reactor," which was issued by the U.S. Nuclear Regulatory Commission (NRC) in March 1989. The PSER documents the preapplication review by the NRC Office of Nuclear Regulatory Research (RES) and discusses unresolved safety issues identified by RES on the MHTGR design. The PSER Issues Tracking System lists the issues in the draft NUREG-1338 by the following information:

- Item number.
- PSER Section.
- Description of issue.
- Disposition Category.
- DOE Comments.
- Revision to the tracking system.

The disposition category is a code letter identified at the beginning of the tracking system which described the importance of the issue to the NRC preapplication review of the MHTGR design following the issuance of the draft NUREG-1338.

The staff reviewed the tracking system and concluded that the tracking system was acceptable to identify and prioritize the issues in the draft NUREG-1338 (NRC letter of August 28, 1991).

2. PSER Issues Tracking System

The DOE PSER Issues Tracking System submitted in the DOE letter of July 31, 1991, is provided in the following pages of this appendix.

Copies of the documents that are listed in this appendix are in the NRC Public Document Room or in the NRC Central Files, for the modular high-temperature gas-cooled reactor design (MHTGR), NRC Project 672.

The documents are not provided in this appendix to reduce the size of Volume 2 for the issued draft MHTGR preapplication safety evaluation report (PSER). The documents will be provided in Volume 2 when the final PSER is issued.

APPENDIX D

DOE SUBMITTALS ON DRAFT NUREG-1338 ISSUES

1. Introduction

Since the draft NUREG-1338 was issued in March 1989 to document the preapplication review of the Modular High Temperature Gas-Cooled Reactor (MHTGR) design up to that date, the U.S. Department of Energy (DOE) has submitted information on unresolved issues in the draft NUREG. A copy of the DOE PSER Issues Tracking System, which lists the unresolved preapplication issues for the MHTGR design in the draft NUREG, is in Appendix B.

2. DOE Submittals

In the meeting of September 29, 1994, DOE provided a list of its submittals on the issues listed in its PSER Tracking System. The submittals are listed as additional data for the issues listed in the tracking system. The additional data is the following:

- Due Date of a DOE response.
- Submittal Date.
- PSID Response.
- Meetings.
- Other Submittals.

The PSID response refers to Chapter R of the Preliminary Safety Information Document (PSID) on the Standard MHTGR (DOE-HTGR-86-024). This Chapter contains staff comments and DOE responses to the comments. The Chapter is divided into general comments and comments on each of the other chapters in the PSID (i.e., Chapter 1 through Chapter 17.

The list of DOE submittals is in the following pages of this appendix. There have not been DOE submittals for every issues listed in the PSER Issues Tracking System.

Copies of the documents that are listed in this appendix are in the NRC Public Document Room or in the NRC Central Files, for the modular high-temperature gas-cooled reactor design (MHTGR), NRC Project 672.

The documents are not provided in this appendix to reduce the size of Volume 2 for the issued draft MHTGR preapplication safety evaluation report (PSER). The documents will be provided in Volume 2 when the final PSER is issued.

APPENDIX E

COMMISSION PAPER SECY-93-092

1. Introduction

The staff identified ten policy and technical issues for the advanced reactors other than the evolutionary and passive advanced light water reactors (LWRs). These advanced reactors are the Department of Energy's (DOE's) helium-cooled Modular High Temperature Gas-Cooled Reactor (MHTGR) and the sodium-cooled Power Reactor Innovative Small Module (PRISM) designs, and the ASEA Brown Boveri-Combustion Engineering Process Inherent Ultimate Safety (PIUS) design. The Atomic Energy of Canada, Limited, Technologies Canadian Deuterium-Uranium (CANDU 3) design was also included because it uses heavy water and is significantly different from United States designed LWRs.

2. SECY Documents

These ten issues were submitted to the Commission in SECY-93-092,""Issues Pertaining to the Advanced Reactor (PRISM, MHTGR, and PIUS) and CANDU 3 Designs and Their Relationship to Current Regulatory Requirements," April 8, 1993.

The Commission addressed the issues in this SECY paper in the Commission Staff Requirements Memorandum (SRM) "SECY-93-092 - Issues Pertaining to the Advanced Reactor (PRISM, MHTGR, and PIUS) and CANDU 3 Designs and Their Relationship to Current Regulatory Requirements," July 30, 1993.

The Advisory Committee for Reactor Safeguards (ACRS) addressed the SECY paper in the letter from Paul Shewmon (Chairman) to Chairman Selin (NRC), "Issues Pertaining to the Advanced Reactor (PRISM, MHTGR, and PIUS) and CANDU 3 Designs and Their Relationship to Current Regulatory Requirements," February 19, 1993.

The staff response to the above ACRS letter is in Enclosure 5 to the SECY paper.

The DOE response to the SECY paper is in the letter from Williams (DOE) to J. Donohew (NRC), with attachment, January 25, 1993.

3. Conclusions

The SECY paper, Commission SRM, ACRS letter, staff response, and DOE response are in the following pages to this appendix.

The policy issues applicable to the MHTGR and the Commission's conclusions are discussed in Section 5.2 of this report.

Copies of the documents that are listed in this appendix are in the NRC Public Document Room or in the NRC Central Files, for the modular high-temperature gas-cooled reactor design (MHTGR), NRC Project 672.

The documents are not provided in this appendix to reduce the size of Volume 2 for the issued draft MHTGR preapplication safety evaluation report (PSER). The documents will be provided in Volume 2 when the final PSER is issued.

APPENDIX I

COMMISSION PAPER SECY-95-132

1. Introduction

The staff discussed eight policy and technical issues for the advanced passive light water reactors (LWRs). These are the General Electric ABWR and Westinghouse AP600 designs.

2. SECY Documents

These eight issues were submitted to the Commission in SECY-94-084, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs," March 28, 1994. The Commission addressed the issues in this SECY paper in the Commission Staff Requirements Memorandum (SRM) "SECY-94-084 - Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems," June 30, 1994.

The staff response to the Commission SRM on SECY-94-084 is in SECY-95-132.

The Westinghouse letter of May 24, 1994, and the staff letter of October 24, 1994, are included with SECY-95-132. These letters are discussed in Section 5.3.14 on RTNSS.

3. Conclusions

The SECY paper and the two letters are in the following pages to this appendix. The policy issues in SECY-95-132 that are applicable to the MHTGR are discussed in Section 5.3 of this report.

Copies of the documents that are listed in this appendix are in the NRC Public Document Room or in the NRC Central Files, for the modular high-temperature gas-cooled reactor design (MHTGR), NRC Project 672.

The documents are not provided in this appendix to reduce the size of Volume 2 for the issued draft MHTGR preapplication safety evaluation report (PSER). The documents will be provided in Volume 2 when the final PSER is issued.

APPENDIX J

CONTRACTOR REPORTS ON THE MHTGR DESIGN

1. Introduction

During the preapplication review of the Modular High Temperature Gas-Cooled Reactor (MHTGR) design, the Nuclear regulatory Commission (NRC) staff conducted technical assistance on the design at the University of Tennessee and at the following Department of Energy (DOE) national laboratories: Brookhaven National Laboratory (BNL) and the Oak Ridge National Laboratory (ORNL).

Since the draft NUREG-1338, the staff's draft preapplication safety evaluation report (PSER) for the MHTGR design, was issued in March 1989, the technical assistance work completed by these contractors for the NRC is reported in the 20 contractor reports listed in Table 6.1 of this report and discussed in Section 6.3 of this report.

The technical assistance completed for the draft NUREG-1338 is discussed in Section 1.9 of the NUREG.

2. Discussion

The following reports are provided in this appendix:

- Oak Ridge National Laboratory, TER 2-10-93, "Review and Evaluation of Recent Publications Bearing on the Fuels Sections of the Draft PSER," February 10, 1993 (Applied Technology).
- Oak Ridge National Laboratory, TER 12-3-92, "Update of Independent Analyses Section 15.4, Preapplication Safety Evaluation Report for the MHTGR, NUREG-1338," December 3, 1993 (Applied Technology).
- Oak Ridge National Laboratory, TER, "An Assessment of MHTGR Cavity Overpressure Accidents that May Impair Functionality of the Reactor Cavity Cooling System," June 22, 1992.
- Oak Ridge National Laboratory, Letter Report 1-20-93A, "Estimate of Air Shock Pressures Induced in the MHTGR Reactor Cavity by a Range of Vessel Failures," January 20, 1993.
- Brookhaven National Laboratory, Letter Report L-2213 11/93, "Initial Assessment of the Data Base for Modelling of Modular High Temperature Gas-Cooled Reactors," November 1993 (Applied Technology).
- University of Tennessee, letter report, "Final Report, Research on Fuel Performance in Modular High-Temperature Gas-Cooled Reactors," letter dated October 7, 1994, from Paul Kasten.

- Brookhaven National Laboratory, NUREG/CR-5261, BNL-NUREG-52174, "Safety Evaluation of MHTGR Licensing Basis Accident Scenarios," April 1989.
- Brookhaven National Laboratory, NUREG/CR-5983, BNL-NUREG-52356, "Safety Aspects of Forced Flow Cooldown Transients in Modular High Temperature Gas-Cooled Reactors," March 1993.

These reports were discussed in Sections 6.3.1.2 through 6.3.1.5, 6.3.1.11, 6.3.1.12, 6.3.2.1, and 6.3.2.7 of this report. Where the original contractor report had Applied Technology information identified by DOE, this information has been been removed from the copies in this appendix.

Copies of the documents that are listed in this appendix are in the NRC Public Document Room or in the NRC Central Files, for the modular high-temperature gas-cooled reactor design (MHTGR), NRC Project 672.

The documents are not provided in this appendix to reduce the size of Volume 2 for the issued draft MHTGR preapplication safety evaluation report (PSER). The documents will be provided in Volume 2 when the final PSER is issued.