NUREG-1338

Draft Preapplication Safety Evaluation Report for the Modular High-Temperature Gas-Cooled Reactor

U.S. Nuclear Regulatory Commission

Office of Nuclear Regulatory Research

P.M. Williams, T.L. King, J.N. Wilson

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P.M. Williams, T.L. King, J.N. Wilson

Division of Regulatory Applications
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555

ABSTRACT

This draft safety evaluation report (SER) presents the preliminary results of a preapplication design review for the standard modular high-temperature gas-cooled reactor (MHTGR) (Project 672). The MHTGR conceptual design was submitted by the U.S. Department of Energy (DOE) in accordance with the U.S. Nuclear Regulatory Commission (NRC) "Statement of Policy for the Regulation of Advanced Nuclear Power Plants" (51 FR 24643), which provides for early Commission review and interaction. The standard MHTGR consists of four identical reactor modules, each with a thermal output of 350 MWt, coupled with two steam turbine-generator sets to produce a total plant electrical output of 540 MWe. The reactors are helium cooled and graphite moderated and utilize ceramically coated particle-type nuclear fuel. The design includes passive reactor-shutdown and decay-heat-removal features.

The staff and its contractors at the Oak Ridge National Laboratory and the Brookhaven National Laboratory have reviewed this design with emphasis on those unique provisions in the design that accomplish the key safety functions of reactor shutdown, decay-heat removal, and containment of radioactive material.

This report presents the NRC staff's technical evaluation of those features in the MHTGR design important to safety, including their proposed research and testing needs. In addition, this report presents the criteria proposed by the NRC staff to judge the acceptability of the MHTGR design and, where possible, includes statements on the potential of the MHTGR to meet these criteria. However, it should be recognized that final conclusions in all matters discussed in this report require approval by the Commission.

Final determination on the acceptability of the MHTGR standard design is contingent on receipt and evaluation of additional information requested from DOE pertaining to the adequacy of the containment design and on the following:

(1) satisfactory resolution of open safety issues identified in this report and possible additional safety issues that may become identified at later stages of review

(2) satisfactory completion of final design and licensing reviews by NRC

(3) conformance with applicable NRC rules, regulations, and other guidelines current at the time of any future licensing action

(4) successful completion of required research and development activities, including design, construction, testing, and operation of a prototype reactor before design certification.

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PREFACE

This safety evaluation report (SER) for the modular high-temperature gas-cooled reactor (MHTGR) is being issued in draft form before final review and approval by the Commission. The review was performed at the request of the U.S. Department of Energy (DOE) consistent with the U.S. Nuclear Regulatory Commission (NRC) Advanced Reactor Policy Statement (51 FR 24643), which provides for early Commission review and interactions for new nuclear power plant concepts. The issuance of this document provides an opportunity for DOE and its contractors to consider details, caveats, and insights from the staff review to date in DOE's further studies of the design.

Final review and issuance of this SER is expected to follow submittal of additional information from DOE regarding the MHTGR containment adequacy. The need for this information became evident in August 1988 when the NRC became aware of DOE's recommendation that a similar MHTGR concept be considered for use as a new production reactor (NPR). As the NPR version of the MHTGR includes a containment structure, the NRC has requested that DOE provide the basis for this apparently significant difference before final NRC conclusions can be given for the civilian MHTGR. The additional information from DOE is expected in June 1989. Accordingly, the use of this document prior to NRC's consideration and evaluation of this forthcoming information must be undertaken with caution, and no conclusions on the overall acceptability of the civilian MHTGR concept should be drawn from this draft SER. Similarly, this draft SER should not be used to draw conclusions regarding the NPR version of the MHTGR, since the NRC did not review the NPR-MHTGR concept and there are several design differences between the civilian and NPR-MHTGR concepts.

In reviewing this draft SER, it should be recognized that the staff positions and conclusions on all matters discussed in this draft SER are subject to change, and in particular, it should be noted that SER sections pertaining to the key issues of (1) the selection of the accidents to be analyzed, (2) the use of a mechanistically derived radionuclide source term for plant-siting evaluations, (3) the acceptability of the design without a conventional containment structure, and (4) the acceptability of the proposed offsite emergency planning will require Commission guidance before final issuance of the SER. These sections are marked with an asterisk. It must be emphasized that in order to resolve the containment issue, the Commission will need thorough and detailed justification to support any design proposal that does not include a containment structure.

ACRONYMS AND INITIALISMS

ABWR	advanced boiling-water reactor
ACI	American Concrete Institute
ACRS	Advisory Committee on Reactor Safeguards
AEC	U.S. Atomic Energy Commission
AISC	American Institute of Steel Construction
ALARA	as low as is reasonably achievable
ANSI	American National Standards Institute
AOO	anticipated operational occurrence
APCSB	Auxiliary and Power Conversion Systems Branch
APWR	advanced pressurized-water reactor
ASME	American Society of Mechanical Engineers
ATWS	anticipated transient(s) without scram
AVR	Arbeitsgemeinschaft Versuchs Reaktor (a reactor in the Federal Republic of Germany)
BDBE	beyond-design-basis event
BE	bounding event
BISO	type of fuel-particle coating (omits SiC layer)
BLBE	beyond-licensing-basis event
BNL	Brookhaven National Laboratory
BOL	beginning of life
BOP	balance of plant
BTP	branch technical position
CAM	continuous air monitor
CFR	Code of Federal Regulations
CLR	core lateral restraint
CR	control room
CRD	control rod drive
CRDM	control rod drive mechanism
C/Th	carbon-to-thorium atomic ratio
C/U	carbon-to-uranium atomic ratio
CWB	chilled water building
DBA	design-basis accident
DBE	design-basis event
DBT	design-basis tornado
DMS	data management subsystem
DOE	U.S. Department of Energy
EAB	exclusion area boundary
EC	event category
ECA	energy-conversion area
EES	economizer-evaporator superheater
EOC	end of cycle
EPA	U.S. Environmental Protection Agency
EPB	emergency-planning basis
EPBE	emergency-planning-basis event

EPZ	emergency planning zone
FDA	final design approval
FIMA	fission(s) per initial (heavy) metal atom
FP	fission product
FR	Federal Register
FRG	Federal Republic of Germany
FRS	floor response spectrum(a)
FS	finishing superheater
FSSAR	final standard safety analysis report
FSV	Fort St. Vrain
FWS	feedwater supply
GA	General Atomics (before 1988, GA Technologies, Inc.)
GASSAR	General Atomic Standard Safety Analysis Report
GCSS	graphite core support structure
GDC	general design criterion(a)
GLRWS	gaseous and liquid radioactive waste system
GSI	generic safety issue
HEPA	high-efficiency particulate air filter
HEU	high-enriched uranium (fuel)
HPS	helium purification system
HTGR	high-temperature gas-cooled reactor
HTS	heat transport system
HVAC	heating, ventilating, and air conditioning
IEEE	Institute of Electrical and Electronics Engineers
INCA	inner neutron control assembly
IPS	investment protection subsystem
IPyC	inner pyrolytic carbon
ISI	inservice inspection
JAERI HENDL	the designation of a Japanese experimental test loop
LBB	leak before break
LBE	licensing-basis event
LBP	lumped burnable poison
LEU	low-enriched uranium (fuel)
LMR	liquid-metal reactor
LN_2	liquid nitrogen
LNS	liquid nitrogen system
LOFC	loss of forced cooling
LOSP	loss of offsite power
LPZ	low-population zone
LTA	low-temperature adsorber
LWR	light-water reactor
MC	main circulator
MCIG	miscellaneous control and instrumentation group
MCS	main circulator subsystem
MCSS	metallic core support structure
MHTGR	modular high-temperature gas-cooled reactor
MLSV	main loop shutoff valve

MMI	man-machine interface
MOC	middle of cycle
MPC	maximum permissible concentration
MSSS	main steam supply system
NCA	neutron control assembly
NCSS	neutron control subsystem
NDTT	nil-ductility transition temperature
NI	nuclear island
NICWB	nuclear island cooling water building
NPR	new production reactor
NRC	U.S. Nuclear Regulatory Commission
NSSS	nuclear steam supply system
NUREG	NRC technical report designation
OBE	operating-basis earthquake
ONCA	outer neutron control assembly
OPyC	outer pyrolytic carbon
ORNL	Oak Ridge National Laboratory
PAG	protective action guideline
PAM	postaccident monitoring
PCDIS	plant control, data, and instrumentation system
PCRV	prestressed-concrete reactor vessel
PCS	plant control system
PDA	preliminary design approval
PDC	principal design criterion(a)
PDCO	plant-design control office
PFDAS	plant fire detection and alarm system
PFPS	plant fire protection system
PFPCDS	plant fire protection carbon dioxide subsystem
PFPHS	plant fire protection Halon subsystem
PFPWS	plant fire protection water subsystem
PPIS	plant protection and instrumentation system
PRA	probabilistic risk assessment
PSB	personnel services building
PSCS	plant supervisory control subsystem
PSID	Preliminary Safety Information Document
PSR	permanent side reflector
PSSAR	preliminary standard safety analysis report
PV	pressure vessel
PWR	pressurized-water reactor
PyC	pyrolytic carbon
QA	quality assurance
RAB	reactor auxiliary building
RB	reactor building
R/B	release-rate-to-birth-rate ratio
RCCS	reactor cavity cooling system
RCPB	reactor coolant pressure boundary
RCSS	reactor core subsystem
RISS	reactor internals subsystem
RG	regulatory guide
RPCWS	reactor plant cooling water system

RS	reactor system
RSA	remote-shutdown area
RSB	reactor service building
RSCE	reserve shutdown control equipment
RSCM	reserve shutdown control material
RSE	reserve shutdown equipment
RSS	reserve shutdown system
RTDP	Reactor Technology Development Plan
RV	reactor vessel
RVS	reactor vessel system
SCC	shutdown cooling circulator
SCCS	shutdown cooling circulator subsystem
SCHE	shutdown cooling heat exchanger
SCHES	shutdown cooling heat exchanger subsystem
SCS	shutdown cooling system
SCWHE	shutdown cooling water heat exchanger
SCWS	shutdown cooling water subsystem
SDA	startup detector assembly
SER	safety evaluation report
SG	steam generator
SGS	steam generator subsystem
SGV	steam generator vessel
SiC	silicon carbide
SLSV	shutdown loop shutoff valve
SPS	safety protection subsystem
SRDC	safety-related design condition
SRP	Standard Review Plan
SRSS	square root of the sum of the squares
SSAR	standard safety analysis report
SSC	structure(s), system(s), and component(s)
SSE	safe-shutdown earthquake
SST	siting source term
SU/SD	startup and shutdown
SWDS	steam and water dump system
SWS	service water system
TDN	technology development need
THTR	Thorium High Temperature Reactor (a reactor in the Federal Republic of Germany)
TID	U.S. Atomic Energy Commission technical report designation
TRISO	type of fuel-particle coating (includes SiC layer)
UCO	uranium oxicarbide fuel
U.K.	United Kingdom
UPS	uninterruptible power supply
UPTPS	upper plenum thermal protection structure
USI	unresolved safety issue
U/Th	uranium-to-thorium ratio
VHTR-C	the designation of a Japanese critical experimental facility
VS	vessel system
VSS	vessel support subsystem

1 INTRODUCTION AND SUMMARY

1.1 General

The staff of the U.S. Nuclear Regulatory Commission (Commission or NRC) has reviewed a conceptual, standardized design for a modular high-temperature gas-cooled reactor (MHTGR). This draft safety evaluation report (SER) presents the preliminary results of that review (Project 672). The MHTGR design was submitted by the U.S. Department of Energy (DOE) in accordance with the NRC Statement of Policy for the Regulation of Advanced Nuclear Power Plants, which was published in the Federal Register on July 8, 1986 (51 FR 24643). The standard MHTGR consists of four identical reactor modules, each with a thermal output of 350 MWt, coupled with two steam turbine-generator sets to produce a total plant electrical output of 540 MWe. The reactors are helium cooled and graphite moderated and utilize ceramically coated particle-type nuclear fuel. The design includes passive reactor-shutdown and decay-heat-removal features, and the proposed overall program has an objective of receiving design certification in the late 1990's.

The design documentation provided by DOE was a Preliminary Safety Information Document (PSID) (Gavigan, 1986) that was supplemented and amended as the review progressed. The staff's review was considered to be a preapplication review with the purpose of providing guidance early in the design process on the regulatory acceptability of the MHTGR design. As such, this SER does not constitute regulatory approval of the MHTGR design, but, rather, it documents the staff's preliminary guidance regarding licensing requirements, including the regulatory acceptability of the DOE-proposed supporting research and development programs.

This SER, in accordance with the Advanced Reactor Policy Statement's provisions for early review and Commission interaction with reactor designers, is intended to guide the development of further documentation to support licensing of the MHTGR concept; however, a licensing determination can only be made by the Commission after the staff has found the MHTGR design to be acceptable and an applicant has complied fully with the administrative processes of nuclear reactor licensing, including public notification and participation, as required in Title 10, "Energy," of the Code of Federal Regulations (CFR), and Title 40, "Protection of the Environment," of the CFR.

1.2 MHTGR Objectives, Approach, and Schedule

DOE has stated that the overall programmatic objective for the MHTGR is its development for a broad range of applications utilizing its unique safety and high-temperature characteristics. The safety characteristics have led to a design that (1) utilizes passive reactor-shutdown and decay-heat-removal features, (2) minimizes the need for operator action and the sensitivity of the design to operator error, and (3) provides long time intervals for corrective actions. As a result of these characteristics, the MHTGR design has reduced the number of systems, components, and structures classified as safety related in comparison with both light-water-reactor (LWR) and other high-temperature gas-cooled-reactor (HTGR) designs. The main control room and the balance-of-plant (BOP) items are proposed to be of commercial, industrial grade, as well as are many items associated with the nuclear island (NI), such as the diesel generators and cooling water systems. In addition, DOE contends that because of these characteristics, certain key changes in traditional approaches to safety are justified. Specifically, DOE has proposed the use of mechanistic siting source terms in lieu of the nonmechanistic siting source term documented in U.S. Atomic Energy Commission (AEC) report TID-14844 (AEC, 1962), no conventional containment building, and no requirements for preplanned offsite emergency evacuation or drills.

DOE is developing the MHTGR with the support of a user utility group, Gas-Cooled Reactor Associates, and a team of contractors. This team consists of General Atomics and Combustion Engineering, Inc. (nuclear steam system vendors) and Bechtel National, Inc., and Stone and Webster Engineering Corp. (architect-engineers). Research and development support is being provided by the Oak Ridge National Laboratory assisted by EG&G Idaho, Inc. General Atomics is responsible for the design of the reactor, fuel, and primary-system machinery, Combustion Engineering for the reactor vessel and steam generator systems, Bechtel for the nuclear island, and Stone and Webster for plant control and the BOP. Originally the General Electric Company was responsible for the plant control system and provided the design reviewed herein.

DOE's interactions with the staff were initiated in June 1984 with a technical briefing on the options then being considered for the MHTGR. Other DOE actions before issuance of the PSID included the submittal to the staff of a draft licensing plan, briefings to the staff on design criteria, accident-selection criteria, safety criteria, and concept selection, as well as briefings on design and programmatic objectives to the Subcommittee on Advanced Reactors of the Advisory Committee on Reactor Safeguards (ACRS) and the full ACRS. Table 1.1 presents selected milestones in the MHTGR review process by both the staff and the ACRS.

The schedule from DOE's licensing plan document (DOE, 1986-1) is given in Figure 1.1 and is based on the assumption of no funding restraints. The milestones proposed for the MHTGR are currently being revised by DOE. DOE has begun the preliminary design phase as scheduled, but site selection for the demonstration project has not yet been announced.

1.3 Background and Design Overview

The origins of commercial gas-cooled reactors are found in the graphite-moderated carbon-dioxide-cooled "Magnox" reactors developed in the early 1950's in the United Kingdom (U.K.) and France. The high-temperature aspect, that is the high-temperature gas-cooled-reactor (HTGR) concept, dates in the United States from the latter 1950's when the design of the fully ceramic core and the use of an inert gas, helium, for cooling was pioneered by the predecessor organizations of General Atomics. This development effort resulted in the 40-MWe Peach Bottom 1 HTGR, which operated between 1967 and 1974, and the 330-MWe Fort St. Vrain HTGR, which began commercial operation in 1976 and continues to operate. Also in the late 1950's, the Federal Republic of Germany (FRG) began designing the "pebble-bed" type of HTGR based on ceramic-fuel developments in the United States. France and the U.K. were also early contributors to HTGR development but dropped their interests in the late 1970's following the termination of the internationally funded 20-MWt Dragon reactor located in the U.K. Two HTGRs are operating in the FRG, the experimental 15-MWe AVR (Arbeitsgemeinschaft Versuchs Reaktor), which began operation in 1967, and the Thorium High Temperature Reactor (THTR), a 300-MWe large HTGR prototype, which started up in 1985. All told, there have been over 50 gas-cooled reactors worldwide with over 900 reactor-years of operation to date.

About 45 reactor-years of this experience has been with HTGRs. DOE maintains an "umbrella" agreement with the FRG for the exchange of technical information and has also developed a technical information exchange agreement with Japan, which is considering building an experimental HTGR for developmental purposes.

Table 1.2 shows the principal design characteristics of the MHTGR in comparison with those of earlier and existing HTGRs and plants designed by General Atomics in the 1970's and early 1980's but not built. The major trends that can be observed in the more recent HTGR designs by comparison with the older HTGR designs are (1) increased system pressures, reflecting the objective of obtaining better heat transfer to the primary coolant; (2) increased use of low-enriched uranium (LEU) fuel, in compliance with the nuclear nonproliferation treaty of 1976; (3) the choice of steel pressure vessels for the smaller HTGRs, including the MHTGR, versus the prestressed-concrete reactor vessels (PCRVs) used for the larger designs; and (4) the goal of greater fuel integrity.

The general safety advantages of the MHTGR, like those of the other HTGRs, are (1) its slow response to core-heatup events, because of the large heat capacity and low power density of the core and (2) the very high temperature that the fuel can sustain before the initiation of fission-product release (about 1600°C). Also, like other HTGRs, its major potential vulnerabilities derive from the need to protect metal components from continued exposure at elevated temperatures to hot helium during postulated transients and to prevent uncontrolled access of air or moisture to hot graphite and fuel particles. The safety of the MHTGR is, to a large extent, based on its proposed design features that utilize (1) passive removal of decay heat, (2) passive reactor shutdown with a modest temperature rise, and (3) high-integrity coated fuel particles. These particles are to maintain their integrity during normal operation and at elevated temperatures under transient conditions or under conditions of chemical attack by steam and air, and are proposed to function both as the initial fission-product barrier and primary reactor containment system. Accordingly, the staff review of the MHTGR concentrated in these areas.

The MHTGR reference configuration was established by DOE after tradeoff evaluations that indicated the selection of (1) "prismatic" fuel blocks over "pebble-bed" spheres; (2) steel primary-system vessels over PCRVs; (3) modular-sized reactors over a larger, single reactor; and (4) separation of the reactor from the remainder of the primary-system components in a side-by-side design rather than the containment of all components "in-line" within a single vessel. Figure 1.2 illustrates the configuration and identifies and locates the major components. Later figures in this SER illustrate other general features of the design, including the annular-core arrangement (Figure 4.2), the fuel-element design (Figures 4.3 and 4.6), the primary and secondary coolant flow paths (Figure 4.4), the passive heat removal system (Figure 5.2), and the arrangement of the modules within the reactor building (Figure 6.2).

1.4 Scope of Review

The major documents supplied by DOE and reviewed by the staff were:

(1) Preliminary Safety Information Document (PSID), Vols. 1-5 (DOE, 1986-3)

(2) Probabilistic Risk Assessment (PRA), Vol. 1 and Vol. 2 (proprietary) (DOE, 1987-1)

(3) Regulatory Technology Development Plan (RTDP) (DOE, 1987-3)

(4) Emergency Planning Basis (EPB) Report (DOE, 1987-2)

(5) Assessment of NRC Light-Water-Reactor Generic Safety Issues (LWR-GSI) (DOE, 1987-4)

These documents and other DOE documents and information supplied by the DOE contractors are explicitly identified in Chapter 18, "References."

Volume 5 of the PSID contains both the staff's written comments and DOE's responses that were developed during the course of the review. The comments included requests for additional information and statements of staff positions. The responses from DOE were also included, as appropriate, by changes in the text of the submitted documents. Ten amendments were made to the PSID and additions were made to the PRA and EPB Report. The LWR-GSI document includes information pertaining to NRC report NUREG-0737. Commitments were made for timely submittals of a revised RDTP and documents describing DOE's plans for prototype-plant testing and the use of industrial-grade equipment for limited safety purposes. Many of the documents supplied for review by DOE and its contractors are classified as "applied technology" by DOE and are identified by asterisk in the references section. Documents classified as applied technology by DOE are not currently in the public domain, and requests for any such documents should be made through DOE. Other documents pertaining to this review, including transcripts of ACRS meetings held on June 22, 1988, July 15, 1988, August 3, 1988, August 12, 1988, October 6, 1988, and October 7, 1988, and reports from the staff's contractors, are available in the NRC Public Document Room, 2120 L Street, N.W., Washington, DC.

The NRC review of the MHTGR design concentrated on those features, issues, and research and development activities considered key to safety and viability. In this process, because of the differences in design from that of an LWR and the way in which the MHTGR design proposes to accomplish the safety functions, certain key safety issues of a policy as well as a technical nature arose that required Commission review and guidance. These policy issues are

(1) the selection of events that must be considered in the design
(2) the siting-source-term calculation and use
(3) the adequacy of the containment concept
(4) the adequacy of emergency planning

Discussions of these issues are provided in Sections 1.7 and 3.2. These issues were also discussed in a paper sent to the Commission on July 15, 1988, SECY 88-203, "Key Licensing Issues Associated With DOE Sponsored Advanced

Reactor Designs." At the present time, the Commission has not taken a position on these issues. The approach and conclusions described in this SER are consistent with the criteria presented in SECY 88-203. The key issues were also reviewed by the ACRS (Kerr, 1988-1). Many other technical issues arose during the review, and these are addressed in the appropriate sections of this SER, as discussed in Section 1.6.

Each chapter or major section within each chapter of this SER identifies the scope of the review and points out the review limitations and the deferred review items. The resources of the reviewers were directed principally toward the areas of fuel design, reactor physics, the reactor vessel, the passive heat removal system, and the safety analysis. Other significant review efforts were made in the areas of heat-transport equipment, components of the primary system boundary, instrumentation, control and electrical systems, selected auxiliary systems, occupational exposures, human factors, safeguards and security, quality assurance, and certain balance-of-plant items. Review was not performed in areas where conventional approaches or experiences with early HTGRs had been fully satisfactory, such as radioactive-waste handling. Because of resource limitations, review also was not performed in areas that the staff believes are capable of successful resolution at a later design stage. These include mechanical equipment design (for example, control rod drives and steam generators), seismic design of unique structures, structural graphite components, and the modeling of fission-product transport and other phenomena involving chemical processes for which experimental data are key to the staff acceptance of any models proposed. A list of those items deferred to a later review stage is given in Table 1.3.

1.5 Review Approach and Criteria

The basic guidance on conducting the review was provided by the Commission's Advanced Reactor Policy Statement of July 8, 1986 (51 FR 24643). The policy statement calls for the "earliest possible interactions with applicants, vendors, and government agencies," and the recognition of reactors that "utilize simplified, inherent or other innovative means to accomplish their safety functions." It further states that advanced reactors should provide, as a minimum, at least the same degree of protection of the public and the environment that is required for current-generation LWRs, but that enhanced safety is expected.

The staff's review approach was first described in a paper to the Commission, SECY 86-368, "NRC Activities Related to the Commission's Policy on the Regulation of Advanced Reactors" (December 10, 1986). Later, NRC report NUREG-1226, "Development and Utilization of the NRC Policy on the Regulation of Advanced Nuclear Power Plants" (June 1988), was issued to document in a more general sense the staff's review approach.

While NUREG-1226 provided guidance on the use of specialized and less prescriptive criteria for the novel aspects of advanced-reactor designs, it also provided guidance with respect to building on established LWR criteria. Hence the guidance the staff utilized in reviewing the MHTGR was also that provided by the recently issued Commission policies on severe accidents (50 FR 32138), safety goals (51 FR 28044), and standardization (52 FR 34884).

A detailed discussion of the development of criteria and use of these policies with respect to the staff's MHTGR review is provided in Section 3.2 of this SER. In general, the review approach used by the staff paralleled that used for LWRs. It consisted of an evaluation of the many factors that contribute to LWR safety (such as conservative design practices directed toward accident prevention and the use of redundancy and diversity in accomplishing key safety functions) and ensuring that similar factors or adequate substitutes were provided for the MHTGR design. Table 1.4 summarizes key examples of this evaluation approach. Accordingly, acceptability of the design was not determined by measurement against a single parameter (such as the safety goals) or by comparison with PRA results for LWRs. While PRA analysis is a useful tool in evaluating a design, the staff does not consider it to be developed to the point where it can be used as the primary measure of reactor safety or acceptability. The staff relied primarily on a deterministic review.

Defense-in-depth was considered in the staff's review of the design and used as a basis for ensuring the MHTGR provides at least equivalent protection to the public and the environment as that provided by current-generation LWRs. Central to the staff's evaluation was the treatment of the policy issues discussed in Sections 1.4 and 3.2. Those policy issues arose because of the different approach used in the MHTGR design to accomplish key safety functions. The staff in its review attempted to develop criteria and a technical position to directly address the acceptability of the key features and policy issues associated with the MHTGR. For example, because of the high potential for preventing core damage in the MHTGR design, a mechanistic analysis of radionuclide releases for a range of low-probability bounding events (equivalent to severe accidents in LWRs and identified in Table 3.7) was proposed as a substitute for the traditional nonmechanistic large source term that is representative of a source term from a core-melt accident utilized in LWR siting. Guidance from the Commission's Safety Goal Policy Statement was used to help define the range of low-probability events to be considered; however, provision was maintained for engineering judgment to bound uncertainties in the selection of these events. Inherent in this approach is a shift in emphasis in defense-in-depth from accident mitigation to accident prevention and plant protection. Consistent with the above, the review also considered the need for a conventional containment structure and preplanned offsite evacuation in light of the increased emphasis on accident prevention. Otherwise, the review followed the general approach of the Standard Review Plan (SRP) (NUREG-0800), with the chapter-by-chapter format of the SER following this organization. Within each major section of the SER are subsections entitled "Description and Safety Objectives," "Scope of Review," "Review and Design Criteria," "Research and Development," "Safety Issues," and "Conclusions."

One of the products of this review approach was a logical assessment of available and needed review and design criteria. In summary, many of the general design criteria (10 CFR Part 50, Appendix A) and sections of the SRP were found to be highly durable and appropriate or adaptable to the MHTGR. However, adequately developed criteria do not yet exist in a number of important areas and will have to be developed at a later design stage in order to support an actual application. The major criteria needs, in addition to those being developed for the key policy issues, have been identified in the areas of coated-fuel-particle design and manufacture, helium heat transfer and fluid flow, structural graphites, reactor-vessel use at elevated temperatures, primary-system components, use of industrial-grade equipment to perform designated safety functions, passive heat removal, safety-system performance at elevated temperatures, service systems unique to the MHTGR, multiple-unit automatic plant control, human factors, recovery actions, and prototype-plant testing.

The staff's review was aided by independent analyses by contractors at the Oak Ridge National Laboratory (ORNL) and the Brookhaven National Laboratory (BNL). Such independent analyses were directed toward confirming the potential of the key safety features to perform their functions and looking for vulnerabilities through sensitivity studies. A summary of the independent analyses is presented in Appendixes A and B. These contractors also performed reviews of selected topics in fuel design, reactor physics, and safety analysis that have contributed to this SER.

In reviewing the MHTGR, the staff developed definitions for various event categories to be used in the evaluation. Specifically, the staff defined four event categories (ECs), which, in general, correspond to traditional LWR event categories, as follows:

EC-I	Abnormal operational occurrences
EC-II	Design-basis accidents
EC-III	Severe accidents
EC-IV	Emergency-planning-basis events

These event categories were developed to avoid confusion over what events needed to be considered and how they were to be selected. In addition, consideration of EC-III was intended to ensure that low-probability events beyond the traditional design-basis envelope were considered so as to provide a sufficient challenge to the plant to allow the use of a mechanistic siting source term. The consideration of such events is also intended to meet the objectives of the Commission's Severe Accident Policy Statement. Descriptions of these event categories and their use are included in Section 3.2.2.

1.6 Identification of Safety Issues

In each major section of this document, safety issues are identified in the fifth subsection (that is, X.X.5) by alphabetical designation for consistency of reference. Those safety issues that pertain to changes, additions, or quality classification upgrades of equipment are identified in Table 1.5. These are the safety issues that the staff believes must be resolved by changes in the MHTGR design, although some, identified by an asterisk, are believed to have some potential for resolution by research findings or further study. The other major safety issues are listed in Table 1.6, which identifies those safety issues judged resolvable by further design analysis, safety analysis, research, equipment testing, and plant testing. The outcome of the analysis, research, or testing could lead to actual changes in plant design or equipment qualification. Background and discussion of all safety issues are presented in the sections pertaining to the major review areas. Safety issues not included in either Table 1.5 or Table 1.6 are discussed in some individual sections of this SER, but they are generally minor considerations.

1.7 Policy Issue Considerations and Relationships*

The staff's review and evaluation of the MHTGR concept has emphasized the identification and assessment of those technical features and characteristics of the design having major bearing on the four policy issues listed in Section 1.4. The technical assessments pertaining to the first two listed issues ("the selection of events that must be considered in the design" and "the siting-source-term calculation and use") were based on evaluation of the proposed design coupled with the potential for resolving open items through further analyses, design changes, research, and testing. Regulatory technical and policy decisions pertaining to the last two issues ("the adequacy of the containment concept" and "the adequacy of emergency planning") depend partially on the technical aspects of the first two issues but also depend on engineering and public policy confidence judgments pertaining to the successful resolution of the open issues and on the additional information on containment adequacy to be supplied by DOE in June 1989 as discussed in the "Preface" to this SER. The important relationships between the policy issues are described further by the summary discussions of each policy issue below.

1.7.1 Selection of Events That Must Be Considered in Design*

The rationale, details, and criteria for the selection of events and event categories for safety analyses are discussed in Section 3.2.2.1, "Accident Selection," and the use of this material for the assessment of the hazards of the MHTGR is described in Chapter 15, "Safety Analyses." Table 3.7 is a list of staff-imposed bounding events that were developed for the MHTGR in accordance with the definition of EC-III in Section 3.2.2.1. The staff imposed these events on the basis of engineering judgment to bound uncertainties in DOE's probabilistic analysis and to compensate for the use of non-safety-grade equipment. The staff believes that the events covered by EC-I through EC-IV are the appropriate set of events against which to evaluate the safety provisions of the MHTGR design at this stage of review. At a later review stage, the events in each category will be reexamined and revised as necessary as further information becomes available, including more detailed information on sabotage and external events. At the present time, the staff and its contractors and DOE and its contractors have not been able to postulate accidents of reasonable credibility that would cause radionuclide releases exceeding the values proposed as acceptable for each event category as defined in Section 3.2.2.1.

1.7.2 Siting-Source-Term Calculation and Use*

A mechanistic means for the determination of radionuclide release to the environs in the performance of safety analyses has been proposed by DOE. That is, a technically based, analytical means is to be used to estimate radionuclide releases for plant-siting evaluations instead of the customary nonmechanistic bases described in AEC document TID-14844 (AEC, March 1962). The development of a reactor design that supports consideration of a mechanistic siting source term is a major departure from both LWRs and earlier HTGR designs. The staff's imposed bounding events were used by DOE in a mechanistic analysis as a test of the MHTGR's ability to potentially meet proposed siting criteria. Calculated doses at the exclusion area boundary are given in Table 15.3.

*Indicates statement is particularly sensitive to change by evaluation of forthcoming DOE information.

The relationship between the events that must be considered in the design of the MHTGR and the use of the proposed mechanistic source term for siting evaluations is thus seen to be of critical importance. Final acceptance and use of a mechanistic source term is contingent on the satisfactory resolution of many related technical issues and policy considerations.

The source term proposed by DOE is based almost wholly on the radionuclide inventory plated out on primary-system surfaces during normal operation and, to a lesser extent, on that circulating with the helium coolant. DOE proposes that the MHTGR fuel would perform sufficiently well so that this radionuclide inventory would not be appreciably augmented under a spectrum of DOE-postulated transients and the staff-imposed bounding events, which include environments of elevated temperatures of substantial duration and chemical attacks by steam, water, and air.

The technical features supporting the use of the mechanistic source term in computing doses at the exclusion area boundary are (1) the potential fission-product-containment capability of the coated-particle-type fuel, (2) the potential for achievement of very few defective fuel particles at the fuel manufacturing stage (the necessary product quality specifications allow only 25 defective particles per 100,000), (3) the potential for the passive shutdown and heat removal systems under adverse conditions to maintain maximum fuel-particle temperatures below about 1600°C, and (4) the resuspension (lift-off) and transport phenomena pertaining to the radionuclides within the primary system. It should be emphasized that substantial research and testing need to be done to confirm the above potentials.

1.7.3 Adequacy of the Containment Concept*

DOE has proposed that the MHTGR does not require a containment structure. The basis for this proposal is that if a mechanistic analysis is used to calculate the release of radioactive material from the plant under all events included in EC-I through EC-III (including the bounding events), the dose guidelines of 10 CFR Part 100 can be met without a containment structure.

The staff's proposed criteria for containment adequacy are presented and discussed in Section 3.2.2.3. These criteria and DOE's proposed containment requirements will be reexamined during evaluation of the new information on the MHTGR containment, as discussed in the "Preface." While acceptance of the DOE-proposed containment requirements will be dependent on NRC's acceptance of the proposed mechanistic siting source term, it does not necessarily follow that acceptance of the proposed source term will subsequently result in acceptance of the proposed containment requirements. The staff believes, however, that the establishment of a mechanistic source term could be a safety enhancement and encourages its continued development, even if a conventional containment structure becomes required for the MHTGR.

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*Indicates statement is particularly sensitive to change by evaluation of forthcoming DOE information.

1.7.4 Adequacy of Emergency Planning*

DOE has proposed that the MHTGR offsite emergency plan need not require elements of preplanned public notification, evacuation, or sheltering. This recommendation is based on DOE's contention that credible accidents in the MHTGR will not lead to offsite doses in excess of the protective action guidelines (PAGs) of the U.S. Environmental Protection Agency (EPA, 1980). The staff's proposed criterion for this issue is described and discussed in Section 3.2.2.4, "Offsite Emergency Planning." The potential of the MHTGR to meet the staff's proposed criterion is indicated. Section 13.1, "Emergency Preparedness," describes and discusses the staff's review of the proposed emergency plan.

Similar to the staff's position on containment adequacy, the criteria proposed in Section 3.2.2.4 and DOE's proposed emergency-planning requirements will be reexamined during evaluation of the new information to be provided on the containment issues, as discussed in the "Preface." Like the containment adequacy issue, acceptance of the DOE's emergency-planning proposal will be dependent on NRC's acceptance of the proposed mechanistic siting source term, but it will not necessarily follow that acceptance of the source term will result in the acceptance of DOE's proposed ad hoc approach to offsite emergency planning.

1.8 Conclusions and Recommendations*

On the basis of the results of this review, the staff concludes that the MHTGR, as generally conceived, is responsive to the Commission's Advanced Reactor Policy Statement. As stated in the "Preface," the staff expects new information from DOE in June 1989 pertaining to the adequacy of its proposed containment requirements. With this new information, the staff will reexamine both its own proposed criteria for the key policy issues and DOE's proposed approaches for containment design and emergency planning. A final SER will be issued which will provide NRC's conclusions on these topics. It must be emphasized that in order to resolve the containment issue, the Commission will need thorough and detailed justification to support any design proposal that does not include a containment structure.

However, notwithstanding resolution of the key policy issues, the staff believes that the MHTGR has the potential to eventually demonstrate a number of favorable characteristics. Among these are:

(1) the potential for only minor core damage and fission-product release over a wide range of severe challenges to the plant

(2) the objective of reduced dependence on human actions and reduced vulnerability to human error

(3) the calculated long response time of the reactor under accident conditions, which provides time for evaluation and corrective action

(4) the capability to demonstrate by test the significant safety features and performance of the plant over a wide range of events

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*Indicates statement is particularly sensitive to change by evaluation of forthcoming DOE information.

(5) the potential for the development and successful performance of high-quality fuel and passive safety characteristics

In addition, the results of independent analyses by ORNL and BNL indicated good agreement with the designer's predicted performance.

However, the final determination of the acceptability of the design is contingent on the following:

(1) satisfactory resolution of the key policy issues associated with the MHTGR, including the containment adequacy issue

(2) satisfactory resolution of the open safety issues identified in this SER and possible additional safety issues that may become identified at later stages of review

(3) satisfactory completion of final design and licensing reviews by NRC

(4) conformance with applicable NRC rules, regulations, and other guidelines current at the time of any future licensing action

(5) satisfactory completion of research and development activities, including successful design, construction, testing, and operation of a prototype reactor before design certification, as discussed in Chapter 14

The ACRS has completed its review (Kerr, 1988-2) and has found that design and development should continue and research and development could resolve important safety issues before licensing. Its report to Chairman Zech is reproduced in Appendix C. The staff's views of the MHTGR's potential to satisfactorily address the Commission's expectations for safety enhancement, as stated in the Advanced Reactor Policy Statement (51 FR 24643), are presented in Appendix D.

1.9 Participants in the Review

The principal participants in the MHTGR review and the development of this SER are as follows:

Management and Overall Technical Review

Thomas L. King	Chief, Advanced Reactors and Generic Issues Branch
Jerry N. Wilson	Leader, Advanced Reactors and Standardization Section
Peter M. Williams	MHTGR Project Manager

NRC Staff Technical Reviewers

Reviewer	SER Chapter or Section
Leo Beltracchi	7, 13.2
Donald P. Cleary	2
Moni Dey	6.3
John H. Flack	9.6
James C. Glynn	4, 5, 6.2, 15, 15.3
Charles S. Hinson	12
Richard E. Johnson	5.2

Melinda Malloy	17
Barry T. Mendelsohn	13.3
John A. O'Brien	3.5
Edward M. Podolak	13.1
Syed K. Shaukat	3.4, 3.5
Dale F. Thatcher	7, 8

ORNL Contractors

Contributor	SER Chapter or Section
Sydney J. Ball (principal investigator)	4.1, 4.2, 4.4, 5.5, 10.2, Appendix A
John C. Cleveland	Appendix A
James C. Conklyn	5.5, Appendix A
Uri Gat	4.2
David L. Moses	4.2, 4.3, 9.1, 9.2, 9.3, 10.1, 10.3, referenced report on reactor physics
Joseph W. Minarick*	Referenced report on PRA
Donald J. Naus	5.5
C. Barry Orland	5.5
Walter K. Sartory	5.2

BNL Contractors

Contributor	SER Chapter or Section
Peter G. Kroeger (principal investigator)	5.3, 5.4, 5.5, 6.2, 9.4, 9.5, 10.4, Appendix B
Hsiang-Shou Cheng	4.3

*Subcontractor

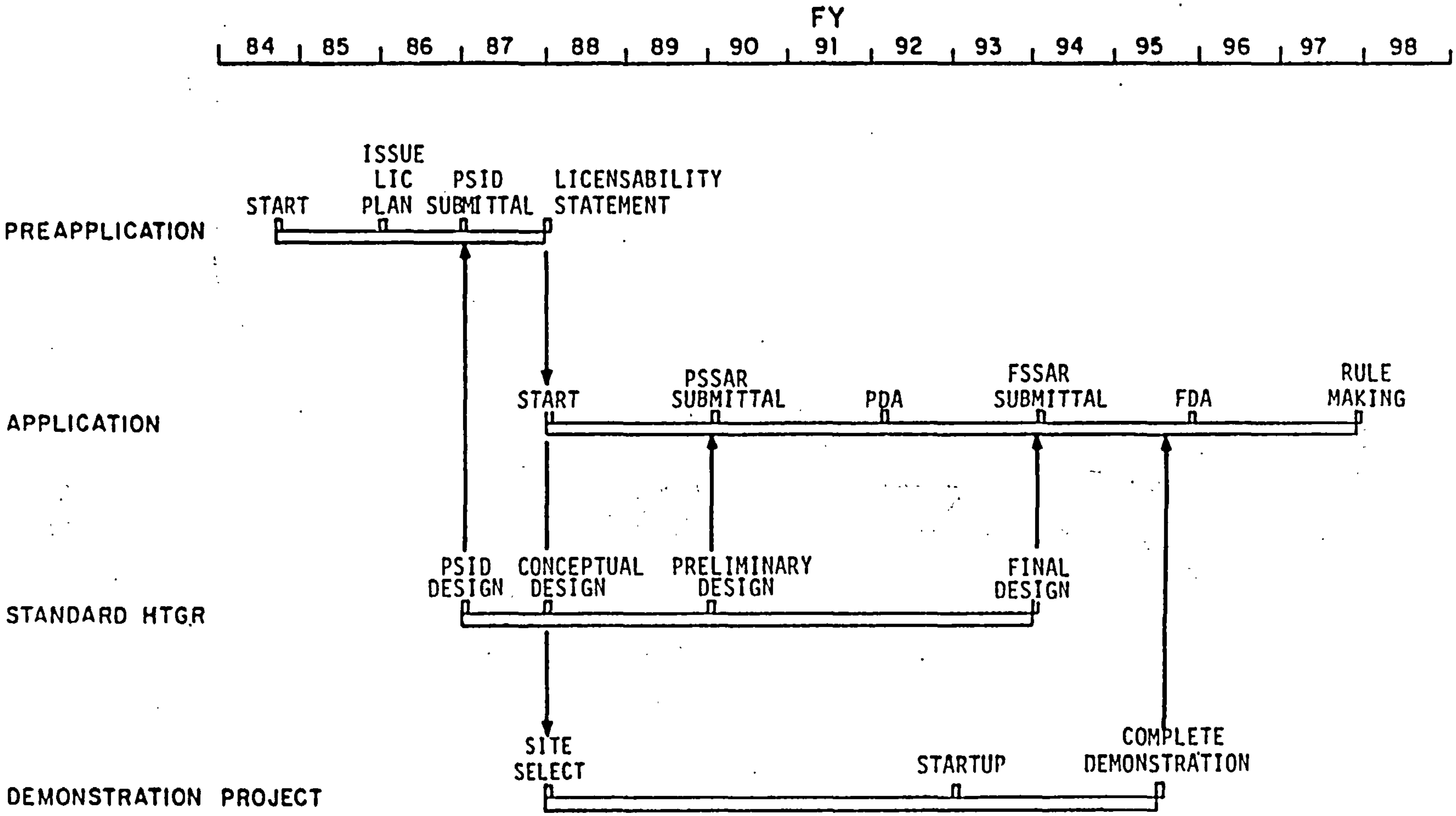

Figure 1.1 DOE-proposed schedule for MHTGR licensing activities
Source: DOE, 1986-1

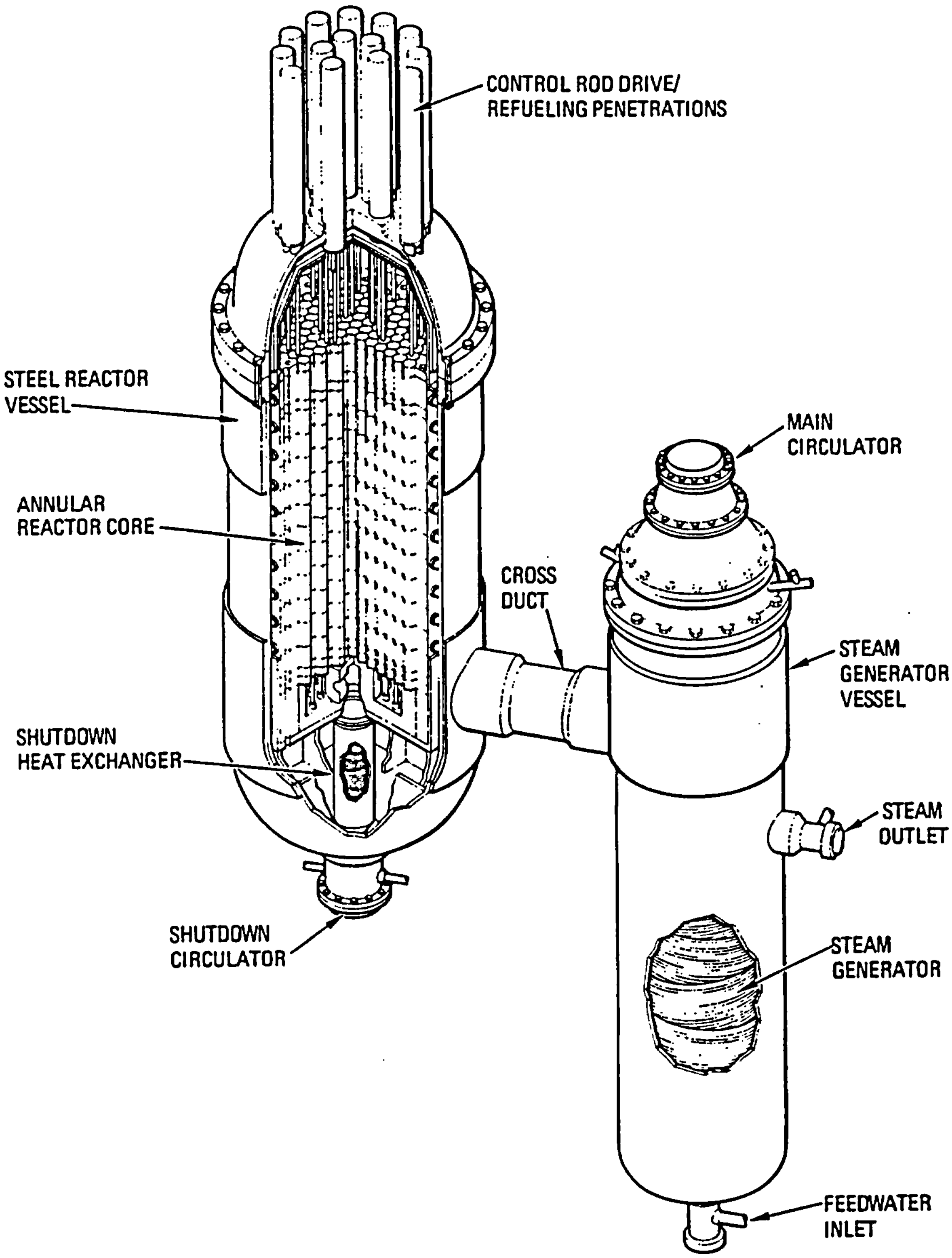

Figure 1.2 350-MWt MHTGR module
Source: DOE, 1986-1

Table 1.1 Selected milestones of the MHTGR review

Event	Date
Technical briefing to staff on MHTGR options	June 13, 1984
Staff acceptance of DOE licensing plan	July 11, 1985
First briefing to ACRS Subcommittee on Advanced Reactors	January 30, 1986
Initiation of formal staff review	October 1, 1986
Seven formal staff and DOE technical meetings	January 20, 1987 through July 16, 1987
First followup staff and DOE technical meeting	October 16, 1987
Second followup staff and DOE technical meeting	March 17 and 18, 1988
Draft safety evaluation report provided to ACRS Subcommittee on Advanced Reactors	June 3, 1988
First meeting on SER with ACRS Subcommittee on Advanced Reactors	June 22, 1988
First meeting on SER with full ACRS	July 15, 1988
Second meeting on SER with ACRS Subcommittee on Advanced Reactors	August 3, 1988
Second meeting on SER with full ACRS	August 12, 1988
Issuance of ACRS letter report to Chairman Zech	October 13, 1988

Table 1.2 Comparison of HTGR designs

Features	Plant designations							
	MHTGR	FSV	THTR	AVR*	Peach Bottom 1	Dragon	GASSAR	Lead plant
Design origin	U.S.	U.S.	FRG	FRG	U.S.	U.K.	U.S.	U.S.
Years of power production	1996	1976-present	1985-present	1967-present	1967-1974	1966-1975	Not constructed	Not constructed
Plant output, MWt/MWe	4 x 350/540	842/330	750/300	46/15	115/40	20/0	3000/1120	2240/850
Reactor core								
Active core dimensions (m)								
Diameter	3.5 (OD) 1.65 (ID)	6.0	5.6	3.0	2.8	1.1	8.5	7.5
Height	7.9	4.8	6.0	2.5	2.5	1.6	6.8	6.3
Core power density (W/cc)	5.9	6.3	6.0	2.5	8.3	14.0	8.4	5.8
Fuel cycle	LEU/Th 19.9% enriched	HEU/Th	HEU/Th	HEU/Th and LEU	LEU/Th	LEU/Th	HEU/Th 93% enriched	LEU/Th 19.9% enriched
Reactor vessel	Steel	PCRV	PCRV	Steel	Steel	Steel	PCRV	PCRV
Primary cooling system								
Pressure (bar)	64	48	40	11	24	17	50	50
Core inlet gas temperature (°C)	260	405	250	270	340	350	319	319
Core exit gas temperature (°C)	690	785	750	950	725	750	755	756
Fuel								
Fissile particle	UCO-TRISO	$(Th,U)C_2$-TRISO	$(Th,U)O_2$-BISO	$(Th,U)C_2$-BISO	$(Th,U)C_2$-BISO	UO_2-TRISO (Zr,U)C	UC_2-TRISO	UCO-TRISO
Fertile particle	ThO_2-TRISO	ThC_2-TRISO	$(Th,U)O_2$-BISO	$(Th,U)O_2$-BISO	$(Th,U)C_2$-BISO	(Th,U)C-BISO	ThO_2-BISO	ThO_2-TRISO
Fuel-element type	Prism	Prism	Sphere (pebble bed)	Sphere (pebble bed)	Cylinder	Hexagonal rods pin-in-block	Prism	Prism
Fuel-element lifetime (years)	3.3	6	3	3	3	Variable	4	4
Gas circulator								
Number	4 (1/module)	4	6	2	2	6	6	4
Compressor type	Single-stage axial	Single-stage axial	Single-stage radial	Single-stage radial	Single-stage radial	Single-stage radial	Single-stage axial	Single-stage radial
Bearing	Magnetic	Water	Oil	Oil	Oil	Gas	Water	Water
Steam generator								
Number	4 (1/module)	12	6	1	2	6	6	4
Type	Helical, nonreheat	Helical, with gas reheat	Helical, with gas reheat	Evolvent, nonreheat	U-tube, with steam drum	Helical heat exchanger	Helical, with gas reheat	Helical, nonreheat
Residual heat removal								
Primary	Main loop	Two separate main loops	Two separate main loops	Main loop	Two separate main loops	Main loop	Two separate main loops	Two separate main loops
Secondary	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	Core auxiliary cooling system
Tertiary	Reactor-cavity cooling system	PCRV liner cooling	None	None	Reactor-vessel cooling panels	None	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	Containment above grade

*Mixed fissile/fertile particle is used

NOTES: MHTGR = modular high-temperature gas-cooled reactor; FSV = Fort St. Vrain; THTR = Thorium High Temperature Reactor; AVR = Arbeitsgemeinschaft Versuchs Reaktor; GASSAR = General Atomic Standard Safety Analysis Report; FRG = Federal Republic of Germany; U.K = United Kingdom; OD = outside diameter; ID = inside diameter; LEU/Th = low-enriched uranium/thorium; HEU/Th = high enriched uranium/thorium; PCRV = prestressed-concrete reactor vessel; UCO = uranium oxicarbide fuel; TRISO = type of fuel-particle coating (includes SiC layer); BISO - type of fuel-particle coating (omits SiC layer)

Table 1.3 Deferred review items

Item	Section in which discussed
<u>Site Location and Description</u>	2
<u>Criteria Directed Toward Ensuring at Least an Equivalent Level of Safety as That of Light-Water Reactor (LWR)</u>	
Final determination of applicability of general design criteria and other specific criteria	3.2.1.1
Systematic review of applicable LWR generic issues	3.2.1.1
<u>Safety Classification of Structures, Systems, and Components</u>	
Final determination	3.3
<u>Design of Structures</u>	
Wind and tornado loadings	3.4.5.A
Missile protection	3.4.5.B
Loads and loading combinations	3.4.5.D
<u>Seismic Design</u>	
Reactor vessel system including support system	3.5.2
Core and reactor internals	3.5.2
Reactor cavity cooling system	3.5.2
Damping values	3.5.5.B
Development of floor-response spectra	3.5.5.D
Torsional effects	3.5.5.F
<u>Fuel Design</u>	
Thorium-containing particles	4.2.2
Fuel-block cracking under thermal stress	4.2.5.F
<u>Nuclear Design</u>	
Research programs on reserve shutdown and burnable poison material, control rod guide tube vibration testing, and neutron control assembly flow and leak testing	4.3.4
Safety classification of flux mapping and inner control rods	4.3.5.G
Mechanical design details of control rods, drive system, and reserve shutdown control equipment	4.3.5.H

Table 1.3 (Continued)

Item	Section in which discussed
<u>Thermal and Fluid-Flow Design</u>	
Flow-modeling test program	4.4.4
Hot streaks	4.4.5.D
Laminar flow effects	4.4.5.E
<u>Reactor Internals</u>	
Expert review of graphite structural issues, design and inspection codes, seismic design, cyclic stresses and displacements, inservice deterioration of materials, inservice inspection, and related research programs	4.5.2
Safety classification of hot duct	4.5.5.B
<u>Vessel System and Subsystems</u>	
Crossduct and steam generator vessels, pressure relief system	5.2.2
Vessel support system	5.2.2
Probability of gross vessel failure	5.2.5.D
ASME approval for elevated-temperature service	5.2.5.D
Pneumatic failure mode	5.2.5.E
Leakage detection	5.2.5.F
Thermal stress, strains, and creep-fatigue interaction	5.2.5.G
<u>Heat Transport System and Subsystems</u>	
Relation of circulator testing program to safety	5.3.4
<u>Shutdown Cooling System and Subsystems</u>	
Safety classification and high reliability	5.4.5.B
Single heat sink for multiple modules	5.4.5.C
Diversion of flow from core by main loop shutoff valve failure	5.4.5.D
<u>Reactor Cavity Cooling System</u>	
Repair and recovery	5.5.5.B

Table 1.3 (Continued)

Item	Section in which discussed
Reactor Cavity Cooling System (Con.)	
Modeling conservatisms and sensitivities to uncertainties (geometrical and asymmetrical effects)	5.5.5.C
Reactor-cavity temperature (details)	5.5.5.D
Inservice inspection program	5.5.5.E
Safety classification of instrumentation	5.5.5.G
Failure modes (details)	5.5.5.I
Plant Arrangement	
Overall plant layout and building designs	6.1.2
Reactor Building	
Protection of reactor cavity cooling system	6.2.5.B
Heat transmission to earth (cavity integrity)	6.2.5.C
Overpressure protection of building and contents	6.2.5.D
Recovery of reactor cavity cooling system	6.2.5.E
Retention of radionuclides	6.2.5.F
Combustible-gas control	6.3.5.G
Hinged-louver and blowout-panel design (details)	6.2.5.H
Cavity flooding	6.2.5.I
Plant Protection and Instrumentation System	
Comprehensive study of reactor and equipment trip systems	7.2.2
Sharing of protection and control instrumentation sensors and other items	7.2.5.B
Nonsafety classification of portions of the plant protection and instrumentation system	7.2.5.C
Possible design changes or developments needed to resolve human concerns	7.2.6
Plant Control, Data, and Instrumentation System	
Power-generation stability	7.3.5.A, 10.1.5.A
Isolation of normal plant control systems from the plant protection and instrumentation system	7.3.5.B
Control-system failures	7.3.5.C

Table 1.3 (Continued)

Item	Section in which discussed
Miscellaneous Control and Instrumentation Group	
Nuclear steam supply system analytical monitoring, radiation monitoring, seismic monitoring, meteorological monitoring, fire detection, and alarm	7.4.2
Electrical Systems	
Acceptable levels of capacity and duration of essential power systems	8.1.6
Fault clearing on essential uninterruptible power supply channels	8.2.5.B
Sharing of essential dc power among reactor modules	8.3.5.A
Physical independence	8.3.5.B
All other electrical systems	8.4
Service Systems	
Detailed design descriptions and safety analysis of all systems	9
Steam and Energy Conversion Systems	
Detailed design descriptions, safety analyses, and conclusions for all systems	10
Equipment qualification concerns for high-energy-line breaks	10.1.5.B
Operational Radionuclide Control	
Assumptions used in "back-calculations" of radionuclide design criteria	11.1.5.A
Model for "back-calculations"	11.1.5.B
Design basis for the helium purification system	11.1.5.C
Radwaste systems	11
Dose assessment of discharged radionuclides	11
Plant normal operations	11
Response to anticipated operational occurrences	11
Occupational Radiation Protection	
Details of operational radiation protection program	12

Table 1.3 (Continued)

Item	Section in which discussed
Emergency Preparedness	
Details of emergency plan	13.1
Role of Operators	
Human factors engineering plan	13.2.4
Completion of control-system design	13.2.5.B
Postaccident monitoring and communications	13.2.5.C
Major operator error	13.2.5.H
Safeguards and Security	
Completion of review with submittal of safeguards information	13.3.1
Staff development of safeguards guidance for advanced reactors	13.3
Prototype-Plant Testing	
Detailed test plan and acceptance criteria	14
Safety Analysis	
Candidates for event category IV	15.1.2
Combustible-gas generation (final determination)	15.2.6.1
Graphite fire (final determination)	15.2.6.2
Additional probabilistic risk assessment studies	15.3.5
Completion of independent analyses	15.4
Final selection of siting source term	15.5
Demonstration that safety analysis is comprehensive and sufficient	15.6
Technical Specifications and Administrative Controls	16
Quality Assurance	
Review details	17
Means to ensure quality in essential foreign research and development	17
Quality assurance program necessary to ensure quality of as-fabricated fuel	17

Table 1.4 Summary of staff approach to ensuring at least the same degree of protection as for light-water reactors (LWRs)

Factors contributing to safety	Staff approach: LWRs	Staff approach: MHTGR
Accident prevention	Use of accepted and conservative design codes and practices	Use of applicable LWR criteria and standards; development of additional criteria and standards, as necessary
	Assurance of high-quality design construction, operation, and maintenance	Same, with possible enhancement in certain areas
Protection and mitigation of accidents	Requirements on performing key safety functions:	
	• Reactor shutdown	Same, with enhancement
	• Decay-heat removal	Same, with enhancement
	• Containment building	Substitute high degree of core-damage prevention for containment building
	Limits on core damage from design-basis accidents and containment failure from severe challenges	Same
	Limits on fission-product release from various events	Same
Safety analysis, documentation, and limits	Selection of appropriate design-basis accidents (DBAs) to be considered plus analysis of accommodation of severe challenges	Same, with severe challenges considered in the design for siting determination
	Thorough safety analyses:	Thorough safety analyses verified by prototype testing:
	• Conservative analyses of DBAs	Same
	• Severe-challenge analyses by best-estimate techniques	Same
	• Probabilistic risk assessments	Same

Table 1.4 (Continued)

Factors contributing to safety	Staff approach: LWRs	Staff approach: MHTGR
Safety analysis, documentation, and limits (Con.)	Technical Specifications	Same
	Surveillance and testing programs	Same
Siting	Deterministic source-term analysis based on core-melt accident and offsite-release analysis based on assumption of containment integrity; 10 CFR Part 100 dose determination	Source term and release based on mechanistic analysis of range of severe accidents; 10 CFR Part 100 dose determination
Emergency planning	Preplanned offsite evacuation required	Ad hoc evacuation acceptable if sufficient time available
	Preplanned ingestion-pathway actions required	Same
Operating experience	Much available information and many lessons learned; prototype test will add to experience	Operating experience at Peach Bottom 1, Fort St. Vrain, and in Federal Republic of Germany; some LWR experience applicable; prototype test will add to experience
Human factors	Training	Same
	Licensed operators	Same
	Safety-grade control room	Simplified passive systems and reduced vulnerability to human error compensate for control room and operator function requirements
Sabotage prevention	10 CFR Part 73 requirements	Same; underground location and passive safety features evaluated for protection versus LWR features

Table 1.5 Safety issues related to design selections

Safety issue/design selections	Section in which discussed
Seismic Design	
Seismic instrumentation	3.5.5.A
Thermal and Fluid-Flow Design	
Core-flow distribution (invessel-temperature monitoring*)	4.4.5.A, 4.4.6
Vessel and Heat Removal Systems	
Service levels for conduction-cooldown events	5.2.5.C
Classification of components	5.3.5.A
Steam generator	5.3.5.B
Main circulator cooling water system	5.3.5.C
Issues similar to heat transport system safety issues	5.4.5.A
Safety classification and high reliability*	5.4.5.B
Vessel-temperature monitoring	5.5.5.G
Duct and chimney design*	5.5.5.J
Plant Arrangement	
Location of control room and protection of reactor operators	6.1.2
Reactor Building	
Containment function*	6.2.5.A
Combustible-gas control*	6.2.5.G
Plant Protection, Instrumentation, and Control Systems	
Manual trip and role of the operators	7.2.5.A
Block valve closure interlock system	7.2.5.D
Steam generator dump and isolation valve actuation*	7.2.5.E
Plant protection system status monitoring*	7.2.5.F
Electrical Systems	
Power capacity for operator information needs and actions	8.1.5.A
Power-duration needs and station blackout	8.1.5.B

*Dependent on research findings or further study.

Table 1.5 (Continued)

Safety issue/design selections	Section in which discussed
Helium Purification System	
Safety classification to ensure primary-system depressurization	9.2.5.D
Liquid Nitrogen System	
Importance to instrumentation and dependent systems	9.3.5.B
Reactor Plant Cooling Water System	
Cooling of neutron control assemblies	9.4.5.A
Steam and Water Dump System	
Safety classification*	10.3.5.A
Service Water System	
Safety classification*	10.4.5.A
Role of Operators	
Manual means for reactor trip	13.2.5.A
Postaccident monitoring and communication	13.2.5.C
Accident mitigation and recovery actions	13.2.5.D
Defense-in-depth from control room operators	13.2.5.E

*Dependent on research findings or further study.

Table 1.6 Major safety issues requiring analysis, research, or testing for resolution

Safety issue	Section in which discussed
<u>Fuel Design</u>	
Fuel-performance models*	4.2.5.A
Fuel-performance statistics from laboratory testing*	4.2.5.B
Manufacturing quality*	4.2.5.C
Normal operation fuel performance	4.5.5.E
Ability of fuel to withstand accident-induced temperatures and environments*	4.2.5.G
Effects of fuel composition on performance	4.2.5.H
Effects of exposure to external chemical attack on fuel performance*	4.2.5.I
<u>Nuclear Design</u>	
Calculational uncertainties**	4.3.5.A
Methods and data validation**	4.3.5.B
Reactor shutdown**	4.3.5.C
Steam or water ingress**	4.3.5.D
Reactor-vessel fluence**	4.3.5.E
Decay heat**	4.3.5.F
Qualification and startup testing**	4.3.5.I
<u>Thermal and Fluid-Flow Design</u>	
Core-flow distribution	4.4.5.A
Hot streaks	4.4.5.D
Laminar flow effects	4.4.5.E
<u>Reactor Internals</u>	
Seismic design and fragility data	4.5.5.C
Thermal and fluid mechanical affects	4.5.5.E
Inservice deterioration of materials*	4.5.5.G

*Identified in DOE Reactor Technology Development Plan (RTDP).
**DOE commitment for inclusion in RTDP.

Table 1.6 (Continued)

Safety issue	Section in which discussed
<u>Vessel and Heat Removal Systems</u>	
Chemical attacks on primary-system metals	5.1
<u>Vessel System and Subsystems</u>	
Neutron irradiation*	5.2.5.B
<u>Heat Transport System and Subsystems</u>	
Mechanical failures of main circulator	5.3.5.D
<u>Shutdown Cooling System and Subsystems</u>	
Safety classification and high reliability	5.4.5.B
<u>Reactor Cavity Cooling System</u>	
Heat-transport design, including vessel hot spots, invessel conduction, reactor-vessel-panel emissivities, and effect of water vapor	5.5.5.A
Repair and recovery	5.5.5.B
Modeling conservatisms and sensitivities to uncertainties	5.5.5.C
Reactor-cavity temperatures	5.5.5.D
Duct and chimney design	5.5.5.J
Total failure of reactor cavity cooling system	5.5.5.K
<u>Reactor Building</u>	
Containment function	6.2.5.A
Protection of reactor cavity cooling system	6.2.5.B
Heat transmission to the earth	6.2.5.C
Recovery of reactor cavity cooling system	6.2.5.E
<u>Plant Control, Data and Instrumentation System</u>	
Power-generation stability	7.3.5.A
<u>Radionuclide Design Criteria</u>	
Assumptions used in "back-calculations" of radionuclide design criteria*	11.1.5.A

*Identified in DOE Reactor Technology Development Plan (RTDP).

Table 1.6 (Continued)

Safety issue	Section in which discussed
Radionuclide Design Criteria (Con.)	
Model for "back-calculations"*	11.1.5.B
Role of Operators	
Review plan for advanced control-system technology	13.2.5.F
Task analysis, crew size, and training for operations	13.2.5.G
Prototype-Plant Testing	
Location of prototype plant	14
Safety-related testing	14

*Identified in DOE Reactor Technology Development Plan (RTDP).

2 SITE LOCATION AND DESCRIPTION

No site locations have yet been proposed for the MHTGR. Rather, Chapter 2 of the Preliminary Safety Information Document (PSID) describes the standard site characteristics assumed by DOE for the design of the standard MHTGR power plant. These characteristics are representative of about 85 percent of the potential sites in the United States. Enveloping parameters were identified for geography, demography, meteorology, hydrology, and seismology. Values assumed for the safe-shutdown earthquake and the operating-basis earthquake are 0.3 g and 0.15 g, respectively. The population distribution is assumed to be 500 persons per square mile. The low-population zone and the exclusion area boundary are proposed to be coincident with a radius of 425 meters. Considerations of nearby industrial, transportation, and military facilities will be deferred until an actual site is being studied.

The treatment of standard site characteristics is generally consistent with regulatory guidance provided for determining the suitability of sites for light-water-cooled and high-temperature gas-cooled nuclear power stations. Regulatory guidance is summarized in Appendix A of Regulatory Guide 4.7, "General Site Suitability Criteria for Nuclear Power Stations." The staff believes that the site characteristics considered are consistent with siting at a large number of potential sites. This conclusion does not negate the need for a full-scope determination of site suitability for individual MHTGR construction-permit applications.

The site as well as supporting material presented is typical of a rural, relatively low-population area. Metropolitan or industrial-park siting has not been considered in the NRC review of the MHTGR design. Should more densely populated surroundings be considered, DOE will provide further site details and formally request a staff review that considers such new information.

3 CONFORMANCE WITH CRITERIA AND POLICIES

3.1 Advanced-Reactor Design Criteria

3.1.1 Description of DOE's Approach

DOE's overall philosophy guiding the design of the MHTGR is to produce a safe, economical plant that meets NRC and user requirements (GCRA, 1986) by providing defense-in-depth through pursuit of four goals:

(1) Maintain plant operation.
(2) Maintain plant protection.
(3) Maintain control of radionuclide release.
(4) Maintain emergency preparedness.

With regard to the achievement of goals 1 and 2, measures are to be taken in the design of the MHTGR to minimize defects in the fuel and to purify the primary circuit of any radionuclides that do escape the fuel so that normal operational releases or any accidental releases of primary-circuit activity would be low and worker exposures minimized. These techniques have proved to be effective in operating gas-cooled reactors as demonstrated by measuring releases and worker exposures.

The unique aspect of the MHTGR design, however, is the approach taken to achieve goal 3. To maintain control of radionuclide releases with high assurance, the design of the MHTGR has been guided by the additional philosophy that control of radionuclide releases must be accomplished by retention of radionuclides within the fuel particles with minimal reliance on active design features or operator actions. DOE's requirements for goals 3 and 4 are to control radionuclide releases so reliably that emergency planning does not require provisions for the offsite sheltering or evacuation of the public. DOE developed top-level regulatory criteria (DOE, 1986-2) in order to quantify goal 3, along with the following bases for the selection of the top-level regulatory criteria:

(1) Top-level regulatory criteria should be a necessary and sufficient set of direct statements of acceptable health and safety consequences or risks to individuals or the public.

(2) Top-level regulatory criteria should be independent of reactor type and site.

(3) Top-level regulatory criteria should be quantifiable.

DOE used these bases to select the following top-level regulatory criteria that contain numerically expressed criteria:

(1) Policy Statement of Safety Goals (51 FR 28044):

(a) The risk to an average individual in the vicinity of a nuclear power plant of prompt fatalities that might result from reactor accidents should not exceed one-tenth of one percent (0.1 percent) of the sum of prompt fatality risks resulting from other accidents to which members of the U.S. population are generally exposed.

(b) The risk to the population in the area near a nuclear power plant of cancer fatalities that might result from nuclear power plant operation should not exceed one-tenth of one percent (0.1 percent) of the sum of cancer fatality risks resulting from all other causes.

(2) 10 CFR Part 20 - Standards for Protection Against Radiation:

(a) Section 20.101 - Radiation dose standards for individuals in restricted areas:

Whole-body dose < 3 rem in calendar quarter
Whole-body dose < 5 (N-18) rem lifetime

(b) Section 20.103 - Exposure of individuals to concentrations of radioactive materials in air in restricted areas:

Limits specified in Appendix B, Table I, Column 1

(c) Section 20.105 - Permissible levels of radiation in unrestricted areas:

Whole-body dose < 0.5 rem in calendar year
Whole-body dose < 0.002 rem in any one hour
Whole-body dose < 0.1 rem in any seven consecutive days

(d) Section 20.106 - Radioactivity in effluents to unrestricted areas:

Limits specified in Appendix B, Table II

(3) 10 CFR Part 50, Appendix I, Section II - Guides on design objectives for light-water-cooled nuclear power reactors licensed under 10 CFR Part 50:

(a) Paragraph A

Estimated annual dose from liquid effluents less than 0.003 rem to the whole body or 0.01 rem to any organ

(b) Paragraph B

Estimated annual dose from gaseous effluents less than 0.005 rem to the whole body or 0.015 rem to the skin or any organ

(c) Paragraph C

Estimated annual dose from radioactive material in particulate form in effluents to the atmosphere less than 0.015 rem to any organ

(4) 40 CFR Part 190 - Environmental Radiation Protection Standards for Nuclear Power Operations:

(a) Section 190.10(a) - Annual dose equivalent to a member of the general public from uranium fuel cycle operations (as defined in Section 190.02):

Whole-body dose < 0.025 rem
Thyroid dose < 0.075 rem
Any other organ dose < 0.025 rem

(b) Section 190.10(b) - Total quantity of radioactive materials entering the general environment from the entire uranium fuel cycle, per gigawatt-year of electrical energy produced by the fuel cycle:

Krypton-85 < 50,000 curies
Iodine-129 < 5 millicuries
Plutonium and other alpha-emitting and transuranic nuclides with half-lives greater than 1 year < 0.5 millicuries

(5) 10 CFR Part 100 - Reactor Site Criteria:

Two-hour exclusion area boundary and 30-day low-population zone accident doses less than 25 rem to the whole body and 300 rem to the thyroid

(6) Dose Protective Action Guidelines of the Environmental Protection Agency (EPA, 1980):

Intervention indicated for general population if whole-body dose exceeds 1 to 5 rem or thyroid dose exceeds 5 to 25 rem

The other portions of 10 CFR, CFR implementation guidance, and other regulatory sources were not recommended by DOE as top-level criteria either because they are not numerically expressed or otherwise quantifiable, they are reactor type or site specific, they are not direct statements of acceptable risks or consequences to individual or public health and safety or to the environment, or, under an analytically based, top-down engineering approach, they should more appropriately be assessed at a lower level for their applicability to the standard MHTGR design. This includes the general design criteria (GDC) of 10 CFR Part 50, Appendix A.

In addition to the user requirements and the top-level regulatory criteria, DOE also developed licensing bases for the MHTGR. The licensing bases were derived from top-level regulatory criteria by using a method that compares the results of probabilistic risk assessment with the top-level regulatory criteria (GA, 1986-1, 2, 3). The resultant licensing bases proposed by DOE and given and discussed in Section 3.2 of the Preliminary Safety Information Socument (PSID) consist of three major elements:

(1) Defining a set of licensing-basis events (LBEs) (GA, 1987-3) that are used to demonstrate compliance with the top-level regulatory criteria for

a spectrum of offnormal or accident conditions. LBEs encompass the following three event categories:

(a) Anticipated Operational Occurrences (AOOs): These are families of events expected to occur once or more in the plant lifetime. The families of events selected as AOOs at this stage in the MHTGR design are listed in Table 3.1. These AOO event families are realistically analyzed to demonstrate compliance with 10 CFR Part 50, Appendix I; 10 CFR Part 20; and 40 CFR Part 190 dose limits.

(b) Design-Basis Events (DBEs): These are families of events lower in frequency than AOOs that are not expected to occur in the lifetime of one plant but that might occur in a large population of MHTGRs (approximately 200). The families of events selected as DBEs at this stage in the MHTGR design are listed in Table 3.2. These DBEs are conservatively analyzed to demonstrate compliance of the plant and site with 10 CFR Part 100 dose criteria.

(c) Emergency-Planning-Basis Events (EPBEs): These are families of events lower in frequency than DBEs that are not expected to occur in the lifetime of a large number of MHTGRs. The families of events selected as EPBEs at this stage in the MHTGR design are listed in Table 3.3. EPBEs are postulated in addition to DBEs and realistically analyzed in the MHTGR Probabilistic Risk Assessment for purposes of demonstrating compliance with the emergency-planning criteria.

In developing the remaining elements of the MHTGR licensing bases, the principal design criteria (GA, 1987-1), and the classification of equipment, attention is focused on demonstrating regulatory compliance with the dose limits of 10 CFR Part 100 under accident conditions. Therefore, further consideration of LBEs is concentrated on DBEs and, to a lesser extent, EPBEs.

(2) Defining a set of four principal design criteria that are qualitative statements of the design commitments made to ensure that the dose criteria of 10 CFR Part 100 will be met and, therefore, that public health and safety will be protected under accident conditions. The highest-level principal design criteria are set forth below. DOE also developed 11 additional lower-level criteria in the PSID that are used in the selection of safety-related structures, systems, and components.

(a) The reactor fuel shall be designed, fabricated, and operated so that radionuclides are retained within the fuel to the extent that releases to the primary coolant will not exceed acceptable values.

(b) The vessels and other components that limit or prevent the ingress of air or water shall be designed, fabricated, and operated so that the amount of air or water reacting with the core will not exceed acceptable values.

(c) The reactor shall be designed, fabricated, and operated so that the inherent nuclear feedback characteristics ensure that the reactor thermal power will not exceed acceptable values. Additionally, the reactivity control system(s) shall be designed, fabricated, and operated so that during insertion of reactivity the reactor thermal power will not exceed acceptable values.

(d) The intrinsic dimensions and power densities of the reactor core, internals, vessel, and the passive-cooling pathways from the core to the environment shall be designed, fabricated, and operated so that the fuel temperatures will not exceed acceptable values.

(3) Defining a set of "safety-related" structures, systems, and components (SSC) that make up the set of equipment capable of performing all the functions required to limit releases under accident conditions to those allowed by 10 CFR Part 100 (GA, 1987-2). Table 3.4 provides a list of the DOE-proposed "safety-related" SSC for the MHTGR. To maintain consistency with this approach, the use of other terms traditionally associated in light-water-reactor (LWR) practice with a safety-related designation (for example, seismic Category I, safety class, safety grade) has been avoided by DOE. Under current LWR practice, the application of these terms automatically imposes a prescriptive set of codes and standards, which DOE believes is inconsistent with the MHTGR development of requirements in a top-down fashion. An example of the top-down approach is discussed in Section 7.1 of this report.

3.1.2 Evaluation of DOE's Approach

The staff reviewed those sections of the PSID (1.1, 1.2, 3.1, and 3.2) pertaining to DOE's proposed approach and the references identified above. DOE's approach for the design of structures, systems, and components was compared with the NRC regulatory approach through the use of regulations, regulatory guides, general design criteria, and endorsed codes and standards. The staff concludes that DOE's approach is a systematic and useful approach for designing a nuclear plant. However, it is not an adequate replacement for the application of NRC's regulatory approach to the safety and licensing review. Specifically, the staff found, as a result of review of the MHTGR, that many regulatory criteria (10 CFR) and much Standard Review Plan (NRC report NUREG-0800) guidance are applicable to the MHTGR, and the application of these criteria is necessary to ensure that the MHTGR achieves at least an equivalent level of safety as that of current-generation LWRs. In addition, to ensure consistency with the Commission's policies and protection of the public and the environment at least equivalent to that provided by current-generation LWRs, the staff, in some cases, has proposed licensing criteria to address the unique features and characteristics of the MHTGR design. These criteria are discussed in Section 3.2 of this report. Overall, it is the staff's opinion that the DOE-proposed approach to specifying licensing requirements correlates safety and regulation too closely with probabilistic methodology and to focusing on 10 CFR Part 100 dose guidelines. This results in a process that has the potential for overlooking important conservatisms in the design and removing from regulatory review items important to defense-in-depth, operator protection, and ALARA (as low as is reasonably achievable) dose provisions. Accordingly, in the remainder of this SER, the staff has attempted to identify those requirements and safety classifications it considers necessary to ensure that the MHTGR design provides adequate defense-in-depth and a level of safety at least equivalent to that of a current-generation LWR.

3.2 NRC Review Criteria

The approach and criteria to be applied in the review of the MHTGR are in some cases different from those for conventional LWRs because of the MHTGR's unique features and design characteristics. The major proposed differences in the MHTGR's approach can be summarized as (1) desiring to use a more mechanistic approach in the selection of accidents to be considered in the design and in the calculation of a siting source term; (2) not requiring a conventional containment building in the design; (3) eliminating the need for offsite emergency notification, sheltering, evacuation, and drills; and (4) achieving more flexibility in the approach to standardization. Each of these major differences results from the characteristics of the design that, because of the use of passive reactor shutdown and decay heat removal systems, are claimed by DOE to prevent fuel damage for a wide range of accident conditions, including very unlikely events such as anticipated transients without scram (ATWS), station blackout, and multiple operator errors. Accordingly, the NRC staff has looked at the fundamental technical issues associated with each of these areas and has developed an approach and criteria to address each. The approach utilizes the guidance in the Commission's Advanced Reactor, Safety Goal, Severe Accident, and Standardization Policy Statements as the bases for deriving a set of decision criteria against which the MHTGR can be reviewed. Consistent with the guidance in the Advanced Reactor Policy Statement, the overall goal of the approach and the proposed criteria is to ensure that the MHTGR achieves a level of safety at least equivalent to that of current-generation LWRs, where current-generation LWRs are defined as the evolutionary LWRs - advanced boiling-water reactor (ABWR) and advanced pressurized-water reactor (APWR) - currently undergoing review for final design approval. It is important to note that the proposed criteria allow a tradeoff between plant protection and accident mitigation to achieve an equivalent level of safety as that of LWRs; however, they do not allow elimination of either plant protection or mitigation. In addition to addressing an equivalent level of safety, the proposed criteria require that the MHTGR be evaluated for expected enhancements in reactor safety, as discussed in the Advanced Reactor Policy Statement, and improvements be made when justified on a cost/benefit basis.

The proposed criteria are structured into a set of "general criteria" (Section 3.2.1) that describes the overall framework and principles used by the staff in conducting the review of the MHTGR. It should be noted, however, that these "general criteria" could also be applied in the review of any advanced-reactor design significantly different from the design of current-generation LWRs and as such are stated in general terms. In addition to the "general criteria," the staff also developed a set of "specific licensing criteria" (Section 3.2.2) that implements the general criteria in the following areas:

(1) accident selection
(2) siting source term
(3) containment
(4) offsite emergency planning

Again, these "specific licensing criteria" are written in such a fashion that they could be applied in the review of any advanced-reactor design. The staff in developing these criteria has attempted to address directly the acceptability of the key features and policy issues associated with the MHTGR.

Overall Approach. In the review of the MHTGR, the staff used and built on applicable existing regulations and guidelines for safety developed for application to LWRs and, where necessary, developed additional criteria to address the unique characteristics of the design. 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 non-LWR concepts and issues under review. In making such interpretations, the staff has taken an approach directed toward maintaining limits and criteria at least equivalent to those for LWRs pertaining to (1) quality of design, construction, and operation; (2) release of radiation; (3) defense-in-depth; (4) provisions for conservatism to account for plant-specific uncertainties in the designs; and (5) consistency with the guidance under development for future LWRs for the treatment of severe accidents. Each of these considerations appears in the criteria discussed in this section of the report; however, because of the fundamental importance of the defense-in-depth principle to reactor safety, it is essential that its application for advanced reactors be specifically and separately addressed.

Defense-in-Depth Principle As Applied to Advanced Reactors. Defense-in-depth in nuclear power plant safety regulation is a philosophy that entails the use of various layers of requirements that help to ensure that safety is achieved through multiple, diverse, and complementary means. These layers of requirements address the different stages and aspects of plant safety that can be generally categorized as prevention, protection, mitigation, and emergency planning, and include items such as:

(1) plant design using conservative assumptions, appropriate codes and standards, and quality in design, construction, operation, and maintenance to minimize the potential for accidents

(2) high reliability, redundancy, and/or diversity in components, systems, and structures to adequately respond to and protect the plant and the barriers to radiation release in the event of an accident

(3) mitigative capability to delay and limit the release of fission products to the environment in the event an accident leads to the failure of one or more barriers to radiation release

(4) emergency planning for protecting the public in the event radiation release from the plant exceeds acceptable limits

In general, DOE has attempted to maintain defense-in-depth in the MHTGR design by addressing all the categories listed above. However, the MHTGR designers have approached plant design and the means of maintaining defense-in-depth somewhat differently than the LWR designers. In general, the MHTGR design makes a shift in emphasis from mitigation features to highly reliable protection features. For example, MHTGR designers aim to achieve high reliability and protection through the use of simple and passive decay-heat-removal and reactor-shutdown methods, compared with high reliability through active systems as in LWR designs. These passive protection features are directed toward maintaining fuel integrity, even during very unlikely events. Mitigation is provided in the MHTGR design through different containment systems, through physical phenomena (fission-product retention, plateout, and holdup), and through use of the long-time response of the reactor in accident sequences. This has resulted in a design that proposes to accomplish protection, mitigation, and emergency planning in ways different from those used for LWRs, and thus the issues discussed in Section 3.2.2 are raised.

In the development of the criteria discussed in the remaining part of this SER, requirements were included to ensure that each of the four categories of defense-in-depth listed above was addressed in the MHTGR design consistent with its unique characteristics, but with the objective of providing at least equivalent protection to the public when the defense-in-depth provisions are considered as a whole. In summary, the criteria relative to the accident-prevention aspects of defense-in-depth for the MHTGR are intended to require at least equivalent accident-prevention capabilities as those required for current-generation LWRs. The criteria for the protection and mitigation aspects of defense-in-depth are intended to provide equivalent protection to the public and environment against the release of radiation as for LWRs, when viewed together (that is, some trade-off between protection and mitigation is allowed, such as the use of highly reliable passive plant-protection features versus a traditional containment building). The criteria for emergency planning are intended to provide an equivalent level of protection in consideration of the characteristics of the MHTGR.

It should be noted that the staff-proposed criteria include requirements for independent and diverse means to accomplish the main safety functions (reactor shutdown and decay-heat removal) and multiple barriers to prevent the release of radioactive material. It is the staff's judgment that reliance on a single system or plant feature to accomplish these important safety functions (even a highly reliable passive system) is not justifiable in light of the importance of these functions to the protection of public health and safety and, in view of the difficulty of predicting the failure-mode possibilities, in a unique design.

In developing the criteria proposed for use in assessing the key issues, a set of general criteria was developed that describes the approach and framework applied by the staff in the review of the MHTGR and could be applied in the review of any reactor significantly different from current-generation LWRs. These general criteria are discussed first (Section 3.2.1). Second, specific criteria were developed to implement the general criteria for each of the four key issues associated with the MHTGR. It should be emphasized that the proposed criteria were developed on the basis of technical considerations only and were directed toward ensuring an equivalent level of safety as that of current-generation LWRs, as well as requiring that the MHTGR be evaluated for cost-effective safety enhancements. In addition, it should be noted that the criteria were developed in consideration of the long-range goal of DOE to certify the MHTGR design. Since design certification is its ultimate goal and the plans and supporting research and development proposed by DOE for the MHTGR are directed toward certification, the staff's proposed criteria were developed from the perspective of what is required to support design certification.

3.2.1 General Criteria

The following general criteria represent a framework and approach for guiding staff review of the MHTGR. It is from these general criteria that the specific criteria to address the key issues of accident selection, source-term containment, and emergency planning were derived. The general criteria are a combination of

(1) criteria that must be met to ensure at least an equivalent level of safety as that of LWRs, which is the level of safety considered to be adequate protection

(2) criteria associated with enhanced safety

3.2.1.1 Criteria Directed Toward Ensuring at Least an Equivalent Level of Safety as That of LWRs

(1) In the design and review of the MHTGR, the designers and staff shall utilize applicable existing rules and regulations, as interpreted for advanced-reactor concepts. This involves a process similar to that used in the review of the Clinch River Breeder Reactor, whereby the LWR Standard Review Plan (NUREG-0800), general design criteria (GDC), and other regulations were reviewed for their applicability and revised and supplemented, as necessary, to account for the differences and unique attributes of the design as compared with LWRs. The process was carried out by the staff on a conceptual basis for the GDC listed in Table 3.5. For pertinent 10 CFR Part 50 design rules, the review status is given in Table 3.6. Final determination of the applicability of these GDC and other specific criteria will necessarily be made at a later design stage. At the preliminary standard safety analysis report (PSSAR) stage, a summary matrix will be made showing how each regulation and GDC is met, surpassed, or deviated from.

The following major exceptions to existing rules and regulations were proposed at this review stage:*

(a) Permit calculation of the siting source term based on mechanistic analysis instead of the large nonmechanistic source term applied to LWRs (that is, the AEC report TID-14844 (AEC, 1962) source term used in the 10 CFR Part 100 siting determination).*

(b) Permit the containment function to be performed in a fashion that is different from that for LWRs.*

(c) Permit offsite emergency planning to be modified to reflect plant safety characteristics.*

Specific criteria developed for substitution in these three areas are discussed in Section 3.2.2.

(2) The advanced-reactor concepts shall comply with the intent of the severe-accident requirements, which are being formulated for LWRs, as follows:

(a) Meet the four procedural criteria for new plants stated in the Commission's Severe Accident Policy Statement.

(b) Identify important severe events to be considered in the design (design dependent).

*Indicates statement is particularly sensitive to change by evaluation of forthcoming DOE information.

(c) Evaluate design features incorporated to prevent severe accidents (design dependent).

(d) Evaluate design features provided for mitigation and accident management (design dependent).

(3) The advanced-reactor concepts must show fission-product (FP)-retention capability at least equivalent to that of LWRs; that is, for equivalent classes of events, criteria associated with FP release (fuel-damage limits, primary-system integrity, and offsite dose limits) from advanced reactors should require the same or better fission-product retention than for LWRs.

(4) The advanced-reactor concepts shall maintain the defense-in-depth concept; however, in its application, consideration may be given to the unique safety characteristics of the advanced plants.* Some tradeoff between prevention and mitigation is acceptable. Defense-in-depth in performing key safety functions must be maintained equivalent to that of LWRs by requiring:

(a) Two diverse, independent means of reactor shutdown, each of which is capable of shutting down the reactor in the event of a single failure of active components and without dependence on support systems (electric power, instrument air, etc.). One of the systems must be capable of bringing the plant to cold shutdown indefinitely. The other system must be capable of bringing the plant to hot shutdown for an extended period of time.

(b) Two diverse, independent means of decay-heat removal, each of which is capable of removing decay heat in the event of a single failure of active components.

(c) Multiple barriers to fission-product release.

(5) To account for the reduced experience, as compared with LWRs, designs that utilize new or innovative features to perform their safety functions must:

(a) Demonstrate before design certification, via testing on the first-of-a-kind or prototype plant, that reasonable assurance will exist that these features will prevent or accommodate accidents. Specific details of plant testing can be determined on a case-by-case basis (based on review of the plant-specific safety analysis, probabilistic risk assessment [PRA], etc.) but, in general, should include sufficient scope of testing on a full-size reactor module to demonstrate the performance of new and innovative safety features over the range of accidents that must be considered in the design.

(b) Develop additional quality assurance (QA), inspection, surveillance, and inservice testing techniques and programs to ensure that the quality and performance of the new and innovative safety features are maintained within acceptable limits over the life of the plant.

*Indicates statement is particularly sensitive to change by evaluation of forthcoming DOE information.

3.2.1.2 Criteria Associated With Assessment of Enhanced Safety

(1) Applicants must assess and document enhanced safety characteristics and margins, such as:

  (a) long response time
  (b) reduced potential for operator error
  (c) capability to retain fission products
  (d) highly reliable safety systems (passive and inherent characteristics)
  (e) simplification (systems and analyses)

(2) Potential improvements in safety are to be considered when the margins are small or when large improvements in safety can be realized with reasonable cost.

(3) Where enhanced safety and margins are used to reduce uncertainty or to affect the design and operation of the facility, these enhancements must be demonstrated via testing on the first-of-a-kind or prototype plant. Specific details of plant testing can be determined on a case-by-case basis.

3.2.2 Specific Licensing Criteria

Within the framework of the general criteria, more specific criteria that implement the general criteria are provided for each of the four key licensing issues. These specific criteria are discussed in the following sections.

3.2.2.1 Accident Selection

Selection of a spectrum of accidents that must be considered in the design, beyond the traditional LWR design-basis-accident (DBA) envelope, is thought to be necessary for advanced reactors. Consideration of such a spectrum of accidents will (1) ensure that advanced designs comply with the Commission's Safety Goal and Severe Accident Policy Statements, (2) provide a sufficient test of the capability of the design to allow use of mechanistic source terms for siting determinations and for decisions regarding containment and emergency evacuation plans, and (3) ensure that the shift in emphasis in defense-in-depth from accident mitigation to accident protection, as compared with LWRs, does in fact provide designs with safety at least equivalent to that of current-generation LWRs. Therefore, it was proposed that a set of event categories be defined that corresponds to events that must be used for design, siting, and emergency planning. Events to be included in each category would be selected deterministically, supplemented by insights gained from a PRA. The events selected could then be used as a basis for calculating source terms, evaluating the safety characteristics of the proposed designs, and assessing the adequacy of containment systems and offsite emergency planning. The following are the staff-proposed event categories and their associated descriptions:

<u>Event Category I</u>. Events in category I (EC-I) would be equivalent to the current anticipated operational occurrence (AOO) class of events considered for LWRs. The frequency range for these events goes down to approximately 10^{-2} per plant-year, which corresponds to the frequency of events that may be expected to occur one or more times during the life of the plant. These events would be analyzed in a manner similar to that used for LWRs to demonstrate compliance

with Appendix I of 10 CFR Part 50 and 40 CFR Part 190. The events proposed by DOE corresponding to EC-I are listed in Table 3.1 and are acceptable to the staff at this stage of the review.

Event Category II. Events in category II (EC-II) would be equivalent to the current DBA category for LWRs and would be selected in a manner consistent with that for selection of an LWR DBA envelope. Specifically, events in EC-II would

(1) Be selected by using traditional engineering judgment, complemented by PRA methods that would include internal events down to a frequency of approximately 10^{-4} per year, a value based on ensuring that any event expected to occur over the lifetime of a population of reactors is included.

(2) Include a traditional selection of external events.

(3) Be subject to single-failure criteria and other traditional conservatisms, with no credit for non-safety-grade equipment, etc. Events in this category would require conservative analysis, as is currently done for LWRs.

The events proposed by DOE corresponding to EC-II are listed in Table 3.2 and are acceptable to the staff at this stage of the review.

Event Category III. Events in category III (EC-III) would correspond to those severe events beyond the traditional DBA envelope that should be used by designers in establishing the design bases. The staff believes that the identification and use of such an event category is consistent with the Commission's Severe Accident Policy Statement and is justified for advanced reactors, particularly those proposing the use of a mechanistic calculation of source terms and a shift in emphasis from accident mitigation to plant protection. The events in this category would be selected using engineering judgment, complemented by PRA. This is consistent with the guidance provided in the Commission's Safety Goal and Severe Accident Policy Statements, which encourages the use of PRA methods to supplement engineering judgment and deterministic (nonmechanistic) analyses. Specifically, events in EC-III would:

(1) Include internal events (less likely initiating events plus multiple-failure events) down to a frequency of approximately 10^{-7} per year; 10^{-7} per year is based on ensuring that the cumulative effect of events below 10^{-6} per year is considered in assessing compliance with the Commission's proposed performance guideline of less than a 10^{-6} per year frequency of a large release of radioactive material to the environment. External events beyond those included in EC-II would be included consistent with their application to future LWRs. Such events are currently being developed as part of the implementation of the Commission's severe accident policy. The events proposed by DOE for emergency planning, as listed in Table 3.3, are considered by the staff to fall into the EC-III category.

(2) Include, using engineering judgment, additional bounding events to account for plant-specific uncertainties. Bounding events for the MHTGR are listed in Table 3.7.

In selecting the events to be included in EC-III, the design would be specifically reviewed to identify those events with the potential of a large release, core melt (or equivalent), or reactivity excursion to ensure that adequate prevention or protection is provided before these events could be excluded from this category. EC-III events could be analyzed on a best-estimate basis.

Event Category IV. Events in category IV (EC-IV) would be used in the assessment of the need for offsite emergency planning. EC-IV includes internal events of frequency similar to the frequency of those events considered in the basis for the emergency planning zones and requirements for LWRs (described in NRC report NUREG-0396, "Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Nuclear Power Plants," December 1978). These events would be analyzed in a PRA and would be used as described in Section 3.2.2.4. DOE's proposed emergency-planning-basis events listed in Table 3.3 do not cover this range. PRA results will be used to evaluate events in EC-IV.

Event Frequencies

The staff recognizes that large uncertainties may exist in PRA results, especially in the lower frequency ranges. Therefore, in selecting and analyzing the events, consideration must be given to the treatment of uncertainties. Accordingly, where the event categories include in their definition a frequency value, this frequency value is intended to be a guideline only and is not to be considered a rigid limit for which compliance must be rigorously demonstrated.

Application to Modular Reactor Designs

In analyzing each event, a determination must be made as to whether it applies to all reactor modules simultaneously or to one module only. In addition, in determining the events to be included in EC-I through EC-IV and in assessing the risk from a plant (where a plant consists of more than one module), the probability of certain events occurring must be increased to account for the multiple modules.

3.2.2.2 Siting-Source-Term Calculation and Use*

The staff believes source terms can be developed for advanced reactors based on mechanistic analysis, provided that (1) those source terms are used in conjunction with dose guidelines consistent with those applied to LWRs, (2) the events considered in the mechanistic analysis are selected to bound credible severe accidents and design-dependent uncertainties, and (3) the performance of the reactor and fuel under normal and offnormal conditions is sufficiently well understood to permit mechanistic analysis. This will provide a realistic estimate of source terms and give advanced-reactor designers incentive to develop designs that minimize releases.

Calculation. The criteria proposed for application in the calculation of a mechanistic siting source term are listed below:

(1) Using the EC-II spectrum, perform a conservative evaluation of EC-II scenarios and calculate the source.

*Indicates statement is particularly sensitive to change by evaluation of forthcoming DOE information.

(2) Using the EC-III spectrum, perform a best-estimate evaluation of EC-III scenarios and calculate the source.

(3) Ensure that sufficient data exist (through a research and development program and/or prototype testing) on reactor and fuel performance under EC-II and EC-III conditions to provide adequate confidence in the mechanistic analysis methods used.

(4) Ensure that none of the EC-II and EC-III scenarios are on a threshold, where a slight change in assumptions or uncertainty can cause an unacceptable change in the source.

Use. To allow the use of mechanistic analysis for siting-source-term selection, the following dose guidelines would apply for the siting determination:

Event category	Dose guidelines	Meteorology
EC-II	10% of 10 CFR Part 100 values	Conservative
EC-III	10 CFR Part 100 values	Conservative

The dose guideline specified for EC-II is based on maintaining an equivalent dose guideline as that for LWRs, where mechanistically calculated source terms are used; that is, where the LWR Standard Review Plan (NUREG-0800) allows the use of mechanistically calculated source terms in analyzing accidents, it specifies that the offsite dose must be a small fraction of 10 CFR Part 100 guidelines and is generally interpreted as 10 to 25 percent of the 10 CFR Part 100 dose guidelines. The dose guideline specified for EC-III is based on applying the same siting dose guideline as is applied to LWRs (10 CFR Part 100) to those events being analyzed in place of the traditional nonmechanistic LWR source term; that is, EC-III events include those severe events that, in an LWR, have traditionally been predicted to result in a core melt, and that, for LWRs, lead to the establishment of the nonmechanistic AEC report TID-14844 (AEC, 1962) source term.

These proposed criteria for the siting-source-term calculation and dose guidelines would be used in conjunction with the traditional assessment of site suitability using the guidelines of Regulatory Guide 4.7, "General Site Suitability Criteria for Nuclear Power Stations," for factors such as population distribution and meteorology. The criteria are not intended to modify any of the other NRC siting guidelines described in Regulatory Guide 4.7.

3.2.2.3 Containment Adequacy*

The staff recognizes that a design without a conventional containment building represents a significant departure from past practice on LWRs and that under certain situations LWR containment buildings have been effective components of the defense-in-depth approach. Therefore, designs that deviate from such practice need to be reviewed to ensure that an equivalent level of safety as that of current-generation LWRs is maintained and that uncertainties in design and

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performance are properly accounted for. The staff believes that such designs are possible, although the ultimate acceptance of such designs will require extensive review, testing, and demonstration. Accordingly, the staff proposes criteria to be met in order to certify a reactor design without a containment building with the understanding that in reviewing a design against these criteria, a large burden will rest with the applicant to demonstrate compliance, particularly in view of the uncertainties associated with a new design.

Specifically, the following are proposed criteria that advanced-reactor designers must meet for NRC certification of a design without a containment building:

(1) The design should contain multiple barriers to radiation release that limit radiation release for EC-I, EC-II, and EC-III events at least equivalent to that of current-generation LWRs. The limits that should be met are

  (a) 10 CFR Part 50, Appendix I, and 40 CFR Part 190 limits for normal operating conditions, including events in EC-I

  (b) 10 percent of 10 CFR Part 100 and the 10 CFR Part 100 dose guidelines, with conservative meteorology, for the EC-II and EC-III events, respectively, as described in Section 3.2.2.2

(2) The fission-product-retention capability of the design must be demonstrated via a testing program utilizing a full-size prototype plant consisting of at least one reactor module and the associated systems, structures, and components necessary to demonstrate safety. Such testing should be done at an isolated site, such as the National Reactor Testing Station, and the prototype plant should conform to the same regulations and standards as the design to be certified. The testing program should generate plant performance data sufficient to validate safety-analysis analytical tools over an extensive range of operating and accident conditions considered in the design (EC-I, -II, and -III), including an assessment of the response of the plant safety features over those conditions that may vary over the life of the plant, such as fuel burnup.

(3) Different emphasis and types of QA, surveillance, inservice inspection, and inservice testing over and above that traditionally employed on LWRs should be provided, as necessary, to ensure that the new and innovative systems, structures, and components that contribute to performing the containment function are, in fact, built, operated, and maintained over the life of the plant in a fashion commensurate with their safety functions. For example, the MHTGR fuel quality may require special attention because of its role in limiting the release of fission products.

(4) Protection of safety-related systems, structures, and components from sabotage and external events should be provided that is at least equivalent to that for current-generation LWRs.

(5) The design should have specific measures to ensure that no core-melt accidents, accidents with significant positive reactivity feedback, or

other accidents with the potential of a large radiation release, such as graphite fires, are in the EC-I, EC-II, or EC-III spectrum.

(6) An assessment of the potential improvement in safety if a containment building were added would have to be made. Judgment would then be used to determine the need for a containment building based on the cost and change in risk.

These criteria are intended to maintain at least the same level of protection of the public and environment, by specifying equivalent dose guidelines and protection, as is provided by current-generation LWRs. In addition, for acceptance of a design without a containment building, these criteria would require demonstration via a full-size prototype test at an isolated site of the fission-product-retention capability of the design. Requiring such demonstration testing is considered necessary to compensate for removal of the traditional (and testable) containment building. Such testing will help ensure that licensed plants of that design have adequate fission-product retention. In fact, the potential of these advanced designs to prevent core damage over an extensive range of low-probability events allows such integrated full-scale testing to be done, whereas the testing of the response of containment buildings (for those designs that utilize containment buildings) to low-probability events is usually limited to less than completely prototypic conditions. These criteria will allow designs that propose to withstand severe or bounding events without the need for a containment building (with due consideration for uncertainties) to be licensed and certified.

3.2.2.4 Offsite Emergency Planning*

Currently, offsite protective actions are recommended when a situation occurs that could lead to offsite doses in excess of the protective action guidelines (PAGs), which are 1 to 5 rem to the whole body and 5 to 25 rem to the thyroid. At the lower projected dose, protective actions should be considered. At the higher projected dose, protective actions are warranted. A dose that has already been accumulated before the decision on whether to take protective actions is not considered to be part of this planning decision. In the past, the Commission has not required offsite emergency planning in those situations where the lower-level PAGs were not expected to be exceeded. For example, emergency planning for research reactors is restricted to the area around the reactor where the lower-level PAGs are expected to be exceeded. This is usually within the owner-controlled area. For fuel-cycle facilities, the proposed rule on emergency preparedness exempts those facilities where the lower-level PAGs will not be reached outside the owner-controlled areas. Therefore, there is a precedent for not requiring offsite emergency planning, beyond simple notification, where warranted by operation. Response of certain offsite agencies into the owner-controlled area (for example, police, fire, and medical personnel) is traditionally considered part of the onsite planning.

The staff believes that emergency-planning requirements for advanced reactors should be based on the characteristics of the designs. This principle is similar

*Indicates statement is particularly sensitive to change by evaluation of forthcoming DOE information.

to that in the emergency planning rule (10 CFR 50.47), which states that the size of the emergency planning zone for HTGRs can be determined on a case-by-case basis. In addition, the power level of each advanced-reactor module is much smaller than that of a conventional LWR and, based on size alone, some reduction in the radius of the emergency planning zone may be warranted similar to what has been done for the existing small-size LWRs. In addition to these considerations, it is the staff's judgment that a plant's ability to prevent significant releases of radioactive material (particularly the prevention of release by core melt) and to provide long times before releases for all but the most remotely probable events should also be reflected in any emergency-planning requirements. Accordingly, the staff proposes criteria that consider such ability, consistent with evaluating a range of events similar to those evaluated for LWRs.

Specifically, the staff proposes the following criteria as guidelines for the advanced-reactor designs in order for NRC to accept the DOE proposal of no traditional offsite emergency planning (other than simple notification). While an offsite emergency plan would still be required, such a plan would not have to include early notification, detailed evacuation planning, and provisions for exercising the plan if

(1) the lower-level PAGs were not predicted to be exceeded at the site boundary within the first 36 hours following any event in categories EC-I, -II, and -III

(2) a PRA for the plant, which included at least all events in categories EC-I through EC-IV, indicated that the cumulative mean value for the frequency of exceeding the lower-level PAGs at the site boundary within the first 36 hours did not exceed approximately 10^{-6} per year

These criteria give credit for designs that provide long times before significant radiation release. For designs such as these, the staff believes that because sufficient time is available, prompt notification of offsite authorities will permit effective evacuation on an ad hoc basis.

3.2.3 Standardization Criteria

All three DOE-sponsored advanced-reactor programs have as their stated objectives the development of a standardized plant design that would be submitted to NRC for design certification. It is expected that the MHTGR PSSAR will reflect and the design will be in accord with the rulemaking, as finalized, on standard design certification (10 CFR Part 52) recently proposed (53 FR 32060). It is currently the intent of 10 CFR Part 52 to address the standardization criteria associated with advanced designs, including the MHTGR, by addressing the following standardization issues:

(1) scope and level of detail of design to be standardized
(2) plant options (number of reactor modules) to be standardized
(3) prototype testing

These criteria are intended to ensure that before a design certification is granted for the design of any plant that is significantly different from one that has been built and operated, high confidence in the performance of the safety features of that design is demonstrated.

3.3 Safety Classification of Structures, Systems, and Components

DOE's approach to defining "safety-related" structures, systems, and components (SSC) is summarized in Section 3.1 of this report, and a list of DOE's proposed "safety-related" SSC is provided in Table 3.4. NRC finds DOE's definition of "safety related" unacceptable for the review of the MHTGR because of its limited focus on 10 CFR Part 100 dose guidelines. The staff has relied on the definition of safety related set forth in 10 CFR 50.49(b) for its review of the MHTGR. The staff believes this definition should be used by DOE in the future development of the MHTGR. Table 3.4 is not fully acceptable to NRC. The staff's positions on design selections that include quality classification upgrades for systems and components are listed in Table 1.5 in line with the NRC definition for safety related and its safety analyses of the MHTGR.

In response to NRC Comment G.3-1, DOE stated that SSC that are important to safety are equivalent to the SSC that are classified as "safety related" in the PSID. NRC finds DOE's position to be unacceptable. The NRC position on the use of the terms "important to safety" and "safety related" are set forth in NRC Generic Letter 84-01, dated January 5, 1984. The staff stated in that generic letter that the two terms are not equivalent and that NRC's regulatory jurisdiction involving a safety matter is not controlled by the use of these terms.

3.4 Design of Structures

3.4.1 Safety Objective

This section pertains to the review of the four safety-related structures described in PSID Chapter 6, "Plant Arrangement, Reactor Building and Containment Design Options." The buildings are the reactor building, the two reactor auxiliary buildings, and the reactor service building. The safety objective for these buildings as stated by DOE is to ensure, with a high level of confidence, that the systems or components housed within will maintain their 10 CFR Part 100-related radionuclide control function under design-basis conditions.

3.4.2 Scope of Review

The staff review focused on the identification and acceptability of design criteria, broad concepts of design procedures, relevant LWR practices, and the potential of the design to meet acceptable standards. No independent calculations were performed by the staff or its contractors for design of structures.

3.4.3 Review and Design Criteria

The principal design criteria considered by the staff in reviewing designs of structures are 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities"; GDC 1, "Quality standards and records"; GDC 2, "Design bases for protection against natural phenomena"; and GDC 4, "Environmental and dynamic effects design bases."

DOE has also made a commitment to meet the intent of the following regulatory guides for design of structures:

1.60 Design Response Spectra for Seismic Design of Nuclear Power Plants (Rev. 1, December 1973)

1.61 Damping Values for Seismic Analysis of Nuclear Power Plants (October 1973)

1.76 Design Basis Tornado for Nuclear Power Plants (April 1974)

1.92 Combining Modal Responses and Spatial Components in Seismic Response Analysis (Rev. 1, February 1976)

1.122 Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components (Rev. 1, February 1978)

Conformance with the identified criteria, codes, specifications, and standards constitutes an acceptable basis for satisfying the applicable requirements of GDC 2 and 4.

3.4.4 Research and Development

No special research needs for design of MHTGR structures to meet the specified design criteria have been identified at this time.

3.4.5 Safety Issues

A. Wind and Tornado Loadings

For wind loads, an envelope of conditions is defined in Section 2.6 of the PSID. As sites are chosen, local wind conditions will be analyzed to ensure that they fall within the design envelope. DOE has selected four tornadoes from Regulatory Guide 1.76, "Design Basis Tornado for Nuclear Power Plants," for Region I as the design-basis tornadoes (DBTs). These are the maximum tornadoes, and when results of probabilistic risk analyses are known, different DBTs may be proposed if the analyses support them. The staff will have to review the appropriate document when submitted.

B. Missile Protection

The location below grade of major portions of the MHTGR systems and components provides inherent protection against many externally generated missiles. Tornado missiles used for design of the MHTGR are Spectrum II, as described in the Standard Review Plan (SRP) (NUREG-0800), Section 3.5.1.4, "Missiles Generated by Natural Phenomena," for Region I tornadoes. Internally generated missiles and their mechanistic and probabilistic analyses will be considered at a later time. A turbine-missile strike is a very low probability because the turbine-generator is arranged so that the planes of rotation of the turbine disks do not intersect any structures, systems, or components required to function to meet 10 CFR Part 100 limits.

C. Barrier-Design Procedures

Missile-resistant barriers are designed to withstand and absorb missile impact loads without being fully penetrated. The overall structural response was also evaluated to ensure structural integrity under missile impact loads. Minimum thicknesses for concrete missile barriers are consistent with the provisions of the SRP, Section 3.5.3, Table 1, "Minimum Acceptable Barrier Thickness Requirements for Local Damage Prediction Against Tornado Generated Missiles." Other barrier-design procedures are consistent with current LWR practice.

D. Loads and Loading Combinations

DOE suggests that as a result of probabilistic analyses, different loads and load combinations may be specified. The staff will have to review such specifications in detail, since this is a deviation from standard LWR practice.

The loading combinations given in Table 3.8-1 of the PSID are not consistent with SRP Section 3.8.4 for seismic Category I structures other than containment buildings. DOE has been informed of this discrepancy and has made a commitment to resolve this matter at a later design stage. In addition, equations 7 and 8 in PSID Table 3.8-1 for concrete structures and equations 7 and 8 on PSID page 3.8-6, Amendment 1, for steel structures decouple seismic and pipe-rupture loads, in direct conflict with SRP acceptance criteria.

The recent amendment to GDC 4 allows application of leak-before-break (LBB) technology to gas-cooled reactors and would be the preferred approach in lieu of decoupling.

E. Design and Analysis Procedures

The general concept for design and analysis of structures is consistent with LWR practice. A three-dimensional model with linear spring elements at each node in three orthogonal directions is used based on half-space theory. Thermal stresses are also considered. The design procedures are in accordance with the applicable portions of the following codes and standards:

(1) American Concrete Institute, ACI-349, "Code Requirements for Nuclear Safety Related Concrete Structures"

(2) American Institute of Steel Construction, AISC-S326, "Specification for the Design, Fabrication and Erection of Structural Steel for Buildings"

(3) American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section III, Division 1, "Nuclear Power Plant Components"

F. Foundation Design and Stability Investigation

The criteria to be used in the design of foundations of structures and investigations of stability against overturning, sliding, and flotation are consistent with the requirements of SRP Section 3.8.5, "Foundations." Stability factors of safety given in PSID Table 3.8-3 are acceptable, since they are consistent with LWR practice and SRP requirements. Conformance with these criteria constitutes an acceptable basis for satisfying, in part, the requirements of GDC 2 and 4.

G. Testing and Inservice Inspection Requirements

MHTGR safety-grade structures should conform to the quality assurance requirements stated in 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants."

3.4.6 Conclusions

The staff concludes that the use of criteria defined by applicable codes, standards, and specifications; the loads and load combinations; the design and analysis procedures; the materials; the quality control and special construction techniques; and the testing and inservice surveillance requirements specified in the PSID provide reasonable assurance that, in the event of winds, tornadoes, earthquakes, and various postulated accidents occurring within the structures, the structures will withstand the specified design conditions without impairment of their structural integrity and the performance of their safety functions, subject to the resolution of open items mentioned in Section 3.4.5 for wind and tornado loadings, missile protection, and loads and loading combinations.

3.5 Seismic Design

3.5.1 Safety Objective

The objective of the seismic design is to ensure the suitable functioning of structures, systems, and components (SSC) that are safety related for the same degree of seismic hazard as postulated for LWRs.

3.5.2 Scope of Review

The staff review followed the current LWR seismic analysis and design practices where applicable. The review deferred evaluation of certain unique features of the MHTGR to a later design stage. These features are (1) the reactor vessel system including the support system, (2) the core and reactor internals, and (3) the reactor cavity cooling system. The staff deferred review of these items because of the lack of present resources and its belief that the state of the art in seismic design and seismic analysis is sufficient to predict that the MHTGR will meet Category I seismic criteria for all safety-grade structures, systems, and components. No independent calculations were performed by the staff or its contractors relative to seismic analyses and design.

3.5.3 Review and Design Criteria

The principal design criteria considered by the staff in its review of the seismic design were GDC 2 and 10 CFR Part 100, Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants."

DOE has also made a commitment to meet the intent of the following regulatory guides for seismic design:

1.12 Instrumentation for Earthquakes (Rev. 1, April 1974)

1.60 Design Response Spectra for Seismic Design of Nuclear Power Plants (Rev. 1, December 1973)

1.61 Damping Values for Seismic Analysis of Nuclear Power Plants (October 1973)

1.92 Combining Modal Responses and Spatial Components in Seismic Response Analysis (Rev. 1, February 1976)

1.122 Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components (Rev. 1, February 1978)

1.142 Safety Related Concrete Structures for Nuclear Power Plants (Other Than Reactor Vessels and Containments) (Rev. 1, November 1981)

With respect to applicability of LWR generic safety issues, DOE has responded in the case of Unresolved Safety Issue (USI) A-40, "Seismic Design Criteria," with an approach consistent with the proposed resolution of this issue. Hence no additional criteria for seismic design are expected to be imposed.

3.5.4 Research and Development

Research and development for features unique to the MHTGR with repect to seismic design and analysis will be discussed with DOE at a later review stage.

3.5.5 Safety Issues

A. Seismic Input

To develop a standard plant design that could be sited on 85 percent of the prospective U.S. sites, over 100 sites were surveyed by DOE, and appropriate seismic parameters were developed and incorporated into the seismic analysis. DOE has made a commitment to perform a site-specific seismic analysis when a specific site is identified. Floor-response spectra (FRS) will be prepared and compared with the MHTGR FRS to ensure that the site-specific FRS are bounded.

B. Damping Values

The damping values used are those provided in Regulatory Guide 1.61, "Damping Values for Seismic Analysis of Nuclear Power Plants," with some exceptions for steam generator tube bundles and the reactor core. DOE claims that these exceptions are justified by dynamic test results for similar configurations. The staff will need to review the test data that support the proposed high damping values for the steam generator tube bundle and the reactor core.

C. Time-History Analyses

Time-history analyses were performed in a manner consistent with the methodology of Regulatory Guide 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants," with input motion applied at the ground surface rather than at the foundation level, as required by the current Standard Review Plan (SRP). This exception was taken because input of seismic motion at the grade level rather than the foundation level was the consensus of the attendees at the NRC Workshop on Soil-Structure Interaction on June 16-18, 1986, in Bethesda, Maryland. The staff is proposing this change in the upcoming revision to SRP Section 3.7.1.

D. Development of Floor-Response Spectra

The computer code SASSI was used to get floor-response spectra at selected nodes in the mathematical model. The floor-response spectra were converted into design-response spectra by a smoothing and peak-broadening process consistent with the methodology in Regulatory Guide 1.122, "Development of of Floor Response Spectra for Seismic Design of Floor-Supported Equipment or Components," with the exception of plus and minus 10-percent peak broadening. This should be plus and minus 15 percent unless special studies are undertaken.

E. Interaction of Structures

Structures that are not required to ensure proper functioning of systems or components during the safe-shutdown earthquake (SSE) but are connected to or adjacent to Category I structures were analyzed for SSE loadings to ensure their integrity and the functionality of other components.

F. Torsional Effects

The approach used by DOE was consistent with LWR design practice; that is, either a three-dimensional finite-element analysis was used or torsional effects were considered by using static factors. The accidental torsional effects were not accounted for. The SRP requires an eccentricity of plus or minus 5 percent to account for uncertainties, and this should be included in the design basis.

G. Reactor Building

Assessment of the structural adequacy of the reactor building was performed by using ACI-349 with the modifications described in Section 3.8 of the PSID to make it consistent with Regulatory Guide 1.142, "Safety-Related Concrete Structures for Nuclear Power Plants (Other Than Reactor Vessels and Containments)."

H. System Seismic Analysis

The system or component will be analyzed by DOE using the response spectra or time-history method. Significant modes of mathematical model are used which are determined so that inclusion of additional modes will not result in more than a 10-percent increase in total response. Individual modes are combined by the square-root-of-the-sum-of-the-squares (SRSS) method, except for closely spaced modes where the absolute sum method is used. All the three-dimensional responses are combined by SRSS of maximum values for each of the three directions. This is consistent with Regulatory Guide 1.92, "Combining Modal Responses and Spatial Components in Seismic Response Analysis."

Seismic qualification of electrical equipment and components is performed on a basis consistent with the provisions of Institute of Electrical and Electronics Engineers (IEEE) Standard 344-1975. Electrical raceway analysis is comparable to methods used in recent LWR final safety analysis reports (FSARs).

For reactor core and core supports, the stress criteria consistent with the ASME Code, Section III, Division 1, Subsection NG, for steel structures and Section III, Division 2, for core support graphite are used. The Division 2 code has not yet been approved by the staff, as discussed in Section 4.5. For fuel and reflector element graphite, stress criteria based on probabilistic considerations are currently being developed. The staff will review these criteria when they are available.

I. Seismic Instrumentation

The DOE-proposed seismic instrumentation program is consistent with Regulatory Guide 1.12, Revision 1, "Instrumentation for Earthquakes," with certain exceptions noted in the PSID. This system achieves the intent of Regulatory Guide 1.12, Revision 1. However, a response spectrum analyzer cannot replace response spectrum recorders. The analyzer processes information, while the recorder collects information. To eliminate the response spectrum recorders would lead to a reduction in seismic instrumentation and in the reliability and diversity of the seismic instrumentation at the plant. Response spectrum recorders are passive devices, whereas accelerometers require a power supply. Although a response spectrum analyzer is useful, it is not a replacement for a response spectrum recorder.

3.5.6 Conclusions

The staff concludes that the procedures that will be utilized for seismic design of structures and components are generally acceptable. The use of these procedures provides reasonable assurance that if a design-basis earthquake should strike seismic Category I structures, the structural integrity of structures, systems, and components will not be impaired.

Items that need to be resolved at a later review have been identified in Section 3.5.5 for damping values, development of floor-response spectra, torsional effects, and seismic instrumentation. The staff also believes that adequate seismic design can be achieved at a later design stage for the unique MHTGR features identified in Section 3.5.2. The staff also believes that for structural graphite the most important needs have been identified. Resolution of these items is not necessary to assess the feasibility and applicability of the MHTGR concept.

Table 3.1 Anticipated operational occurrences (AOOs)

Number	Occurrence
AOO-1	Main-loop transient with forced core cooling
AOO-2	Loss of main and shutdown cooling loops
AOO-3	Control-rod-group withdrawal with control rod trip
AOO-4	Small steam generator leak
AOO-5	Small primary-coolant leak

Source: DOE, 1986-3

Table 3.2 Design-basis events (DBEs)

Number	Event
DBE-1	Loss of heat transport system (HTS) and shutdown cooling system (SCS) cooling
DBE-2	HTS transient without control rod trip
DBE-3	Control-rod withdrawal without HTS cooling
DBE-4	Control-rod withdrawal without HTS and SCS cooling
DBE-5	Earthquake
DBE-6	Moisture inleakage
DBE-7	Moisture inleakage without SCS cooling
DBE-8	Moisture inleakage with moisture-monitor failure
DBE-9	Moisture inleakage with steam-generator-dump failure
DBE-10	Primary-coolant leak
DBE-11	Primary-coolant leak without HTS and SCS cooling

Source: DOE, 1986-3

Table 3.3 Emergency-planning-basis events (EPBEs)

Number	Event
EPBE-1	Moisture inleakage with delayed steam generator isolation and without forced cooling
EPBE-2	Moisture inleakage with delayed steam generator isolation
EPBE-3	Primary-coolant leak in all four modules with neither forced cooling nor helium purification system pumpdown

Source: DOE, 1986-3

Table 3.4 DOE-proposed "safety-related" structures, systems, and components

System	Associated 10 CFR Part 100-related function
Reactor System	
Neutron control - Control rod drive systems - Reserve shutdown control equipment (RSCE) mechanical trip system - Exvessel neutron detection equipment	Control heat generation
Reactor internals - Reflector elements - Lateral restraint assembly - Core-support floor - Upper plenum thermal protection structure	Remove core heat, control heat generation
Reactor core	
- Graphite elements	- Remove core heat
- Fuel rods	- Remove core heat
- Coated particles	- Retain radionuclides in fuel
- Control rods (outer)	- Control heat generation
- RSCE material	- Control heat generation
Vessel System	
Reactor vessel Steam generator vessel Vessel supports Crossduct Pressure-relief piping and valves Steam generator isolation valves	Remove core heat, control chemical attack
Reactor Cavity Cooling System	
Ducting Heat transfer panels Plenum structures Intake and exhaust structures	Remove core heat
Plant Protection and Instrumentation Systems	
Safety protection - Reactor trip circuits, release of control rods - RSCE trip circuits, release of boron balls	Control heat generation

Table 3.4 (Continued)

System	Associated 10 CFR Part 100-related function
Electrical Group	
Class 1E Systems	
- Uninterruptible ac power supply	Control heat generation
Rectifiers	
Inverters	
Distribution equipment	
- DC power system	
Station batteries	
Distribution equipment	
Building, Structures, and Building Services	
Reactor building silo	Remove core heat

Table 3.5 Light-water-reactor general design criteria (GDC) that apply without modification, apply with modification, or do not apply to the MHTGR

GDC	Title	Apply without modification	Apply with modification	Do not apply
1	Quality Standards and Records	X		
2	Design Bases for Protection Against Natural Phenomena	X		
3	Fire Protection	X		
4	Environmental and Dynamic Effects Design Bases	X		
5	Sharing of Structures, Systems, and Components	X		
10	Reactor Design	X		
11	Reactor Inherent Protection	X		
12	Suppression of Reactor Power Oscillations	X		
13	Instrumentation and Control	X		
14	Reactor Coolant Pressure Boundary	X		
15	Reactor Coolant System Design	X		
18	Inspection and Testing of Electric Power Systems	X		
19	Control Room	X		
20	Protection System Functions	X		
21	Protection System Reliability and Testability	X		
22	Protection System Independence	X		
23	Protection System Failure Modes	X		
24	Separation of Protection and Control Systems	X		
25	Protection System Requirements for Reactivity Control Malfunctions	X		
26	Reactivity Control System Redundancy and Capability	X		
27	Combined Reactivity Control Systems Capability	X		
28	Reactivity Limits	X		
29	Protection Against Anticipated Operational Occurrences	X		
30	Quality of Reactor Coolant Pressure Boundary	X		
31	Fracture Prevention of Reactor Coolant Pressure Boundary	X		
32	Inspection of Reactor Coolant Pressure Boundary	X		
44	Cooling Water	X		
45	Inspection of Cooling Water System	X		
46	Testing of Cooling Water System	X		
60	Control of Releases of Radioactive Materials to the Environment	X		
61	Fuel Storage and Handling and Radioactivity Control	X		
62	Prevention of Criticality in Fuel Storage and Handling	X		
63	Monitoring Fuel and Waste Storage	X		

Table 3.5 (Continued)

GDC	Title	Apply without modification	Apply with modification	Do not apply
16	Containment Design		X*	
17	Electric Power Systems		X	
34	Residual Heat Removal		X	
36	Inspection of Emergency Core Cooling System		X	
37	Testing of Emergency Core Cooling System		X	
64	Monitoring Radioactivity Releases		X	
33	Reactor Coolant Makeup			X
35	Emergency Core Cooling			X
38	Containment Heat Removal			X
39	Inspection of Containment Heat System			X*
40	Testing of Containment Heat Removal System			X*
41	Containment Atmosphere Cleanup			X*
42	Inspection of Containment Atmosphere Cleanup Systems			X*
43	Testing of Containment Atmosphere Cleanup System			X*
50	Containment Design Basis			X*
51	Fracture Prevention of Containment Pressure Boundary			X*
52	Capability for Containment Leakage Rate Testing			X*
53	Provisions for Containment Testing and Inspection			X*
54	Systems Penetrating Containment			X*
55	Reactor Coolant Pressure Boundary Penetrating Containment			X*
56	Primary Containment Isolation			X*
57	Closed System Isolation Valves			X*

*Indicates statement is particularly sensitive to change by evaluation of forthcoming DOE information.

Table 3.6 Review status of conformance with pertinent 10 CFR Part 50 design rules

10 CFR section	Title	Review status
50.34a	Design objectives for equipment to control releases of radioactive material in effluents - nuclear power reactors	Deferred to preliminary standard safety analysis report (PSSAR) stage of review.
50.44	Standards for combustible gas control system in light-water power reactors	Discussed in Sections 6.2.5.G and 15.2.5.1. Details deferred to PSSAR stage of review.
50.47	Emergency plans	Discussed in Sections 3.2.2.4 and 13.1.
50.48	Fire protection	Discussed in Section 9.6. Details deferred to PSSAR stage of review.
50.49	Environmental qualification of electrical equipment important to safety for nuclear power plants	Concern identified in Section 10.1.5.B. Details deferred to PSSAR stage of review.
50.55	Codes and standards	Discussed in Chapters 3, 4, 5, 7, and 8.
50.60	Acceptance criteria for fracture prevention measures for light-water nuclear power reactors for normal operation	Discussed in Section 5.2.
50.62	Requirements for reduction of risk from anticipated transient without scram (ATWS) events for light-water-cooled nuclear power plants	Discussed in Sections 4.3 and 15.2 and Chapter 14.

Table 3.7 Bounding events (BEs) for the MHTGR

Number	Event
BE-1	Inadvertent withdrawal of all control rods, without reactor trip for 36 hours (one module): (1) reactor system pressurized, with forced cooling available (2) reactor system pressurized, with reactor cavity cooling system (RCCS) cooling only (3) reactor system depressurized, with RCCS cooling only
BE-2	Station blackout (all modules) for 36 hours: (1) reactor system pressurized (2) reactor system depressurized
BE-3	Loss of forced cooling plus RCCS cooling for 36 hours (one module): (1) reactor system pressurized, RCCS 25 percent unblocked after 36 hours (2) reactor system depressurized, RCCS 25 percent unblocked after 36 hours
BE-4	Rupture of justifiable number of steam generator tubes with failure to isolate or dump steam generator: (1) reactor system depressurized, with forced-circulation cooling maintained (2) reactor system depressurized, without forced-circulation cooling
BE-5	Rapid depressurization (one module). Double-ended guillotine break of crossduct with failure to trip (assume RCCS failed for 36 hours and 25 percent unblocked thereafter). Partial control-rod insertion after 36 hours.
BE-6	Severe external events consistent with those imposed on light-water reactors.

4 REACTOR

4.1 System Characteristics

Figure 4.1 is a cut-away drawing showing the entire reactor primary system, including the major components of the reactor and the heat removal systems. The reactor core is 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 crossduct vessel, contains the steam generator and the other components of the heat transport system (HTS), which includes the main helium circulator and a helium-flow shutoff valve. As may be seen from Figure 4.1, the reactor vessel is above and off to the side of the steam generator vessel to negate natural circulation cooling of the core. This design reduces the possibility of ingress of steam or water to the core in the event of steam generator tube failures provided the expected trip of the main helium circulator has been achieved, and it 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) design is described in the Preliminary Safety Information Document (PSID), Section 4.2. It consists of hexagonal, prismatic, block-type graphite fuel and reflector elements, metallic upper plenum elements, startup sources, and reactivity control material. The active core is formed by the hexagonal fuel elements stacked in columns of 10 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, as shown in Figure 4.2. Each fuel element is 0.793 meter (31.2 inches) high by 36 centimeters (14.2 inches) across flats and contains blind holes for fuel compact rods and full-length channels for helium-coolant flow, as shown in Figure 4.3. Corner holes 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 are supported from below by the graphite core support structure described in Section 4.5.1, and their lateral motion is limited at the top by close-fitting keyed connections provided by the upper-plenum elements.

The fuel elements are of two types - the "standard" element pictured in Figure 4.3 and a similar "reserve shutdown" element, which provides for the insertion of pellets of boron carbide absorber material in a graphite matrix, as described in Section 4.3.1. Similarly sized and replaceable graphite reflector blocks surround the active core annulus. These are also 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 are 0.625 inch in diameter. These coolant holes are interspersed among the 0.50-inch-diameter fuel "compacts" or "rods" in the fuel elements. The design provides for efficient 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 at Fort St. Vrain. The normal transit time of the helium from the top to the bottom of the core at full flow is 0.3 second.

The annular core configuration was selected, in combination with a core average power density of 5.91 MW per cubic meter, to achieve a thermal reactor power rating of 350 MWt and to permit passive core-heat removal while maintaining the maximum fuel temperature below about 1600°C (2912°F) during certain categories of events postulated in the safety analysis described in Section 15.2. 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 a 6.55-meter (258.0-inch) inner diameter. These dimensions allow for a lateral restraint structure between the reflector and vessel that 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 ensuring axial power stability to xenon transients over the entire burnup cycle.

Core reactivity control is achieved by a combination of the fixed LBP in the fuel blocks, movable poison, and a negative temperature coefficient. The movable poison is in the form of metallic-clad, boron carbide control rods and boronated pellets that are part of the neutron control subsystem, described in Section 4.3. Figure 4.1 shows the top head refueling penetrations that house the top entry, gravity-driven control rod assemblies that insert control rods into both the inner and outer reflector regions. No control rods enter the core directly.

Figure 4.4 is a simplified flow diagram showing the normal flow path for the primary coolant and a schematic design for the secondary system. Forced-convection cooling, under normal and shutdown conditions, is provided to the reactor by the main circulator (MC) of the heat transport system (HTS), which is described in Section 5.3, or under shutdown conditions only by the shutdown cooling system (SCS), which is located within the reactor vessel below the core, as described in Section 5.4. For normal conditions the core is cooled by helium leaving the MC at a temperature of 260°C (497°F) and a pressure of 64 bars (925 psia). The helium passes through the outer annulus of the crossduct vessel, up the outer annulus between the core barrel and vessel in rectangular ducts, and then into the upper plenum of the reactor pressure vessel. The coolant then flows 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 begins 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 crossduct vessel. The mixed core outlet temperature is 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 "bypass flow," is through small gaps between the center and side reflector blocks and through other miscellaneous channels and gaps within the core barrel.

4.2 Fuel Design

4.2.1 Design Description and Safety Objectives

The MHTGR uses a low-enriched uranium and thorium fuel that has an initial cycle length of 1.9 years. Subsequent burnup cycles are 3.3 years, with one-half 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 (heavy) 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 distribution is tailored by fuel zoning. Each fuel zone is loaded with different fissile and fertile concentrations to provide heavier concentrations of fissile material (uranium-235) in the higher power zones, while keeping the total core and reload fuel loadings unchanged. In the current zoning scheme, there are three radial and three axial zones. The three axial zones consist of layers that are five, three, and two fuel elements high in the top, middle, and bottom zones, respectively. The 3 radial zones correspond to the 3 annular rings of fuel elements; that is, 18, 24, and 24 columns of fuel elements per ring, as shown in Figure 4.2. 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 ringwise relative power densities of 0.87, 1.00, and 1.10 are achieved and maintained over most of the operating cycle. The axial power fractions are 0.65 for the top zone, 0.25 for the middle zone, and 0.10 for the bottom zone. These power distributions ensure that a maximum fuel temperature of 1250°C (2280°F) is not exceeded during normal operation.

The MHTGR fuel-particle, fuel-element, and core designs were 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 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. As shown in Figures 4.5 and 4.6, 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 as TRISO. The fissile and fertile coated particles are blended and bonded together with a carbonaceous binder into fuel "rods" or "compacts." Rods are inserted into fuel holes drilled through the graphite fuel blocks, as shown in Figure 4.6.

For all the safety-analysis events described in Chapter 15, the fuel is designed to retain radionuclides within fuel-particle coatings under all postulated conditions. Offsite doses for these events are based mainly on the prediction that the only radionuclides released are those that escape the fuel kernel barriers during normal operation. Doses are calculated for depressurization events from the release of the radionuclide inventory circulating with the coolant and the "liftoff" of radionuclides "plated out" in the primary systems. The mechanics for this source term are described in Section 11.1 and its applications are presented in Chapter 15. In order to meet the containment functional objectives for the fuel, the MHTGR fuel must meet a manufacturing product specification on quality that calls for an equivalent fraction of unprotected particles due to silicon carbide coating defects and heavy-metal contamination that is less than 6×10^{-5}. This quality specification is the equivalent of 6 failed fuel particles per 100,000 particles fabricated - 5 from silicon carbide defects and 1 equivalent from heavy-metal (uranium) contamination outside the silicon carbide layer. This as-manufactured quality level has been attained in laboratories in the United States and the Federal Republic of Germany (FRG) but not on production fuels. The corresponding Fort St. Vrain quality level is 9×10^{-3}.

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, which is 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 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.

A descriptive diagram of the essential elements of the TRISO coatings and the interrelations between manufacturing defects and failure modes is shown in Figure 4.5, which is based on DOE's developing fuel-failure model (Neylan, 1988-1). This figure illustrates some of the particle-failure mechanisms that are being considered by DOE in its model. These are pressure-induced failures, failures due to manufacturing defects, failures of the OPyC layer induced by irradiation, and failures of the SiC layer at elevated temperature by internal fission-product corrosion or thermal decomposition. The numbers given on the figure refer to the type and cause of the failures. Pressure-induced failures are stated by DOE to be negligibly small except when the buffer or OPyc layers are missing. The manufacturing process is seen from Figure 4.5 to cause defects in 25 particles per 100,000 that consist of missing or defective layers of buffer, OPyC, SiC, IPyC, or heavy-metal contamination outside the SiC layer.

Of these 25, 19 are classified as nonreleasing defects and 6 are considered releasing. These six consist of five particles with missing SiC layers and one equivalent from heavy-metal contamination exterior to the SiC layer. Normal irradiation results in the five initially nonreleasing particles with a missing buffer layer exposing five more kernels by pressure-induced failure and one of the four initially missing IPyC layers causing an additional release failure by corrosion-induced SiC failure. This results in a total of 12 releasing-type particles at the end of the fuel-element residence time of 3.3 years. If the core is subjected to an elevated temperature, such as could be caused by a conduction-cooldown event, additional SiC failures would increase the total releasing failures to 14. In summary, missing or defective SiC layers lead to direct release of radionuclides and are the dominant release mechanism during normal operation. It is important to emphasize that the numbers not footnoted in Figure 4.5 refer to the approximate planned product specifications in terms of the allowable numbers of detective particles, releasing or nonreleasing, per 100,000 manufactured particles. These specifications control the inventory of radionuclides released to the primary system during normal operation and are central to acceptance of the proposed mechanistic source term. DOE acknowledges that a very high level of manufacturing quality control will be needed to achieve these specifications. Fuel failures by chemical reactions and decomposition begin to occur at temperatures above the temperature maximums estimated in the safety analyses (about 1600°C) and, while considered in the safety analysis, DOE has proposed that such failures can be largely excluded from the source term proposed for the MHTGR.

Figure 4.5 does not show the fuel-failure mode by hydrolysis of the fuel kernel for those particles that have a missing silicon carbide layer or failure modes that could be caused by steam, water, or air ingress or other chemical attacks. This augmented release of fission products would occur if the helium coolant should contain significant trace levels of moisture or induction of steam, such as would follow a steam generator tube failure at operating temperatures.

Failure and performance models for fuel particles, other than from manufacturing defects, were developed from past fuel testing and are being used to assess the adequacy of the MHTGR fuel-particle design for both normal and transient conditions, including failures of adequately manufactured particles. The dominant failure mechanism for temperature conditions exceeding 2000°C is now thought to be due to thermal decomposition of the silicon carbide coating layer into its elemental components. In the 1600°C temperature range, failure is believed to be caused by chemical interaction between the fission products and the silicon carbide layer, but at rates substantially less than by decomposition at higher temperatures. Earlier models considered pressure-induced-type failures from internal gas pressure and by kernel migration caused by carbon mass transport in the presence of a thermal gradient. Kernel migration is the fuel-failure mode addressed in the Fort St. Vrain technical specification safety limit. Both the pressure-induced and kernel-migration failure modes are now expected to have negligible effects because of design changes incorporated into the MHTGR reference fuel particles. The comprehensive failure and performance model that is being developed by DOE will take into account all the contributing conditions for failure for both normal and abnormal conditions. These include parameters such as temperature, time at temperature, operating history, kernel composition and density, fast fluence, burnup, internal and external chemical attack and interactions, critical steps in the manufacturing process, manufacturing deficiencies, and time to failure.

4.2.2 Scope of Review

The staff and a contractor (Oak Ridge National Laboratory) reviewed Section 4.2 of the PSID, additional information provided in PSID Amendment 9, relevant portions of the Probabilistic Risk Assessment, Section 6 of the DOE Regulatory Technology Development Plan (RTDP), and a reference supplied by DOE entitled "US/FRG Accident Condition Fuel Performance Models," HTGR-85-107 (GA, 1985). The staff was guided in its review by NRC report NUREG-0111, "Evaluation of High-Temperature Gas-Cooled Reactor Fuel Particle Coating Failure Models and Data." Although this study was performed some years ago, it reported many of the concerns facing the current review. At a later review stage, the staff will review in detail both past and future fuel development programs. The review will emphasize statistical aspects in recognition that the MHTGR will contain about 10 billion fuel particles in each core. The review did not include a separate review of the fertile particles that contain thorium. At a later review stage the fertile-particle design and its supporting research program will be evaluated in a manner similar to that performed here for the fissile particles, although the contribution to the fission-product inventory in the core is far less than from the fissile particles.

4.2.3 Review and Design Criteria

In the PSID, DOE presented its "10 CFR 100 Design Criteria," which were judged to be too general and not consistent with the need to present complete information regarding the compliance of the fuel with licensing requirements. In general, the MHTGR fuel-performance requirements should be specified as a set of acceptance criteria that are developed to be at least the functional equivalent of the LWR fuel-acceptance criteria in accordance with Section 4.2 of the Standard Review Plan (SRP) (NUREG-0800). Specifically, the performance of the fuel during core-heatup events should be demonstrated by the use of appropriate experimental data with defensible uncertainty estimates. In order to develop these criteria, the fuel must meet the radionuclide design criteria discussed in Section 11.1. General guidance may be found in the proposed revisions to 10 CFR 50.46 (52 FR 6339, "Emergency Core Cooling Systems; Revisions to Acceptance Criteria") and in Section II, Appendix K, of 10 CFR Part 50. The fuel performance should be demonstrated by test and analysis to preclude core damage during normal operation (1) in accordance with the intent of General Design Criterion (GDC) 10 ("Reactor design") during reactivity transients, (2) in accordance with the intent of GDC 27 ("Combined reactivity control systems capability"), and (3) during core heatup with long-term active or passive emergency and abnormal core cooling in accordance with the intent of GDC 35 ("Emergency core cooling"). Specific acceptance criteria for the performance of fuel particles, fuel-particle coatings, fuel rods, and fuel-element blocks during normal operation and accident conditions should be defined mechanistically and quantitatively (with uncertainties) for the MHTGR using the format of SRP Section 4.2 as a general guide. In response to Comment G.3-1, DOE has made a commitment to meet the intent of the relevant GDCs with certain exceptions that clearly relate to LWRs.

4.2.4 Research and Development

The RTDP presented, in Section 6, the proposed technology development needs (TDNs) for fuel and fission-product transport. Priorities of the TDNs were established according to the uncertainty and inadequacy of existing data and the anticipated importance of new data.

A total of 19 TDNs are identified in the RTDP. Those dealing with fuel development and performance are

6-10 Validation of Design Methods for Fission Gas Release

6-11 Validation of Design Methods for Fission Metal Release

6-12 Fuel Irradiation Proof Test

6-13 Fuel Compact Process Development

6-14 Fission Product Diffusivities in Particle Coatings

6-15 Performance Models for Defective Fuel Particles

6-16 Validation of Fuel Performance Models Under Normal Operating Conditions

6-17 Validation of Fuel Performance Models Under Core Core Conduction Cooldown Conditions

6-18 Particle Coating Process Development

6-19 Fission Gas Release From Core Materials

Those TDNs dealing with fission-product plateout and liftoff in the primary coolant system and transport to the environs through the reactor building are identified in Section 11.1.4.

The adequacy of the RTDP for fuel development is an essential requirement for staff acceptance of the MHTGR concept, since the MHTGR proposes to forego some of the traditional requirements for defense-in-depth and would use mechanistic analyses for all safety assessments.* To enable the staff to evaluate the MHTGR design as proposed, the completeness and adequacy of Section 6 of the RTDP must be improved, and it must appropriately consider the safety issues described below. Thus the RTDP must be revised to demonstrate that a coherent and proven correlation exists between the fuel design's safety-related capabilities and all the possible and postulated conditions the fuel may be exposed to. The revised plan must demonstrate explicitly the correlations between the reference fuel design of the MHTGR and its response to the postulated events described in Chapter 15. In Amendment 9 to the PSID, DOE stated that it is expanding and revising the fuel portion of the RTDP as the design progresses and the results of the program warrant. This revision should also provide a description of the manufacturing process including the quality control procedures so that the staff can evaluate the likelihood of achieving the requried product specifications.

*Indicates statement is particularly sensitive to change by evaluation of forthcoming DOE information.

Other areas were identified by the staff as deserving special attention, since they are not yet covered in sufficient detail in the RTDP. The area of statistical uncertainty is important in the design, analysis, testing, and evaluation phases. The area of independent review, as part of the quality assurance (QA) program, needs to be addressed in more detail. Further efforts to reference other independent, parallel work are also encouraged. The fact that no detailed QA plans were made available to the staff was also of concern, since such matters need early attention in this type of program. Finally, it is the staff's impression that throughout the entire fuel program, there is an implicit assumption of "total" success and a lack of contingency planning.

4.2.5 Safety Issues

The staff's primary concerns are in the areas of fuel-failure mechanisms, fuel-manufacturing quality, and the statistical aspects of manufacturing, performance, and research. The underlying issue is whether available data obtained from laboratory-produced fuel or experience gained from operating reactors provide sufficient confidence that the research program will meet its goal of developing fuel that will meet the HTGR performance requirements. It is essential that the final data offered in support of the MHTGR concept be obtained on the reference fuel design in test environments that include the conditions given in Section 15.2. The analysis and resulting correlations should not be based on an interpretation or extrapolation of earlier fuel designs.

A. Fuel-Performance Models

The document entitled "US/FRG Accident Condition Fuel Performance Models," HTGR-85-107 (GA, 1985), became the principal focus of the staff's review of the fuel-particle model. The staff made specific comments to DOE on this report, to which DOE responded in Amendment 9 to the PSID. While these responses were generally helpful in describing the fuel-performance model, the following specific comments remain:

(1) At the present time, the NRC staff considers the report to be the best available document in support of a fuel that is to be significantly more advanced than and superior to older fuels in terms of radionuclide retention under the normal and abnormal operating conditions of the MHTGR. A discussion should be developed in a revised or subsequent document to show that the reference fuel will be systematically and consistently appropriate for the MHTGR concept. The advancements in the fuel are to be due in large part to better manufacturing techniques that result in fewer defects and (statistical) variations in the particle and TRISO-coating characteristics. 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. This will assure the staff that at least some of the wealth of test data available on the older fuels will be transferrable to the MHTGR fuel-performance derivations.

(2) There are two basic assumptions underlying the data presented in the report: (a) cesium release is a direct indicator of silicon carbide coating failure, and (b) the delay in the release of krypton after silicon carbide failure is due to a diffusive transport mechanism in the remaining intact outer layer of pyrolytic carbon. The DOE response to the staff's comment about these assumptions in Amendment 9 notes that several independent experiments

have confirmed their validity. Since these are crucial points in the understanding of fuel-performance data, these experiments should be referenced and documented by the preliminary standard safety analysis report (PSSAR).

(3) Almost all the experiments cited for high-temperature failure conditions are out-of-reactor, simulated heating tests. The RTDP should provide for systematic testing to determine whether any synergistic effects due to radiation and high temperatures are present that would affect the (very small) fuel-failure fraction predictions. DOE noted in response to Comment 4-45.C that no synergistic effects have yet been observed, but it is not clear that they would not exist at the level of failure probabilities for MHTGR fuel. Also, the significance (or lack thereof) of any differences in particle internal pressures for in-pile operation and out-of-pile tests should be described and documented. For example, although essentially all the total gases present during normal operation are due to stable fission products, it is necessary to confirm that the particle internal pressure is not reduced because of diffusion of these gases between the time of irradiation and the out-of-pile heatup testing.

B. Fuel-Performance Statistics From Laboratory Testing

It is necessary to demonstrate the achievement of the statistically low failure probabilities at a satisfactory confidence level in the face of a multitude of affecting parameters. This will require a rigorous research and development program that complies with a systematic statistical approach commensurate with the number of parameters and the required accuracy. This is necessary to enable the staff to accept the fuel-quality level proposed. This concern was noted previously in NRC report NUREG-0111. In PSID Amendment 9, DOE agreed to provide further validation of statistical correlations with test data derived from the RDTP. Specifically, there are two areas in which statistical information on the production of fuel and its performance needs to be supported by laboratory-scale testing.

(1) The statistical validity of experimental evidence is a well-established branch of statistics, and the RTDP does not currently explain how quality assurance will be applied to this observation. In response to Comment 4-52, DOE stated that the RTDP will be modified to explain in more detail how compliance with 10 CFR Part 50, Appendix B, is to be achieved.

(2) The means for achieving 95- and 50-percent confidence levels need to be confirmed, and the associated Weibull probability distribution should be validated. While the staff agrees that the Weibull distribution appears to give excellent fits of the failure data, the parameters used in the correlations should be carefully verified to ensure that confirmation data exist for all combinations or parameter ranges applicable to the normal and accident-condition circumstances to which the model is to be applied. The performance testing program needs to demonstrate that the 5- and 50-percent levels of "nonconfidence" do not result in exceeding the stated performance requirements. This needs to be done in such a manner that it will satisfy the intent of SRP Section 4.2. In response to Comment 4-49.2, DOE partially addressed this concern, but for resolution it will be necessary for the staff to review at a later design stage DOE's planned separate document that will describe the fuel quality assurance and control program.

C. Manufacturing Quality

Statistical quality control and assurance plans for fuel manufacture, including acceptance criteria, need to be considered in the RTDP so that there will be assurance that the actual reference fuel is of the specified quality and will perform as predicted. In particular, the quality control program must contain a fuel-particle and fuel-compact sampling scheme and inspection technique that reflect the allowable defect rate. Although DOE proposes to set up the fuel manufacturing quality assurance/quality control (QA/QC) document separately from the RTDP, the verification of the QA/QC plan is expected to require empirical confirmation within the scope of the RTDP. Manufacturing defects are accounted for (statistically) in the fuel-performance model, as noted in the DOE response to Comment 4-45.G.

D. Weak Fuel Particles

The staff considers "weak" fuel particles to be those particles that do not fail like defective particles under normal operation (which gives evidence of the amount of defective fuel) but would fail unexpectedly under postulated transient conditions. For this reason the staff requires that the particle-failure model and the manufacturing specifications be developed to account for particle weaknesses and that the accompanying quality assurance program be capable of their detection with the same reliability as for fully defective particles. In response to Comment 4-53, DOE stated that the fuel-performance models account for a distribution that includes the weak particles as the "tails" of the distribution curve. 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 staff finds this approach acceptable at this review stage.

E. Normal Operation Fuel Performance

The present data represent primarily separate effects obtained in experiments. The correlations for in-place fuel must be firmly established, since low failure probabilities and high accuracies are required. The RTDP recognizes the need for integrated proof testing to indicate any weakness in the fuel integrity by long-term exposures at normal operating conditions. There is, however, no commitment to perform such tests with full-scale elements under normal conditions. DOE has, in response to Comment 4-54, argued against the need for such testing, noting that the F-30 capsule irradiated in the General Electric Test Reactor provided proof (later confirmed) of Fort St. Vrain's fuel performance, and hence the MHTGR-related capsule tests should do likewise. While this provides a helpful precedent, the fact that the fuel-performance requirements for the MHTGR are so much more stringent makes this type of extrapolation less credible. The staff believes that a long-term arrangement should be pursued to use the Fort St. Vrain reactor for this purpose, coupled with an appropriate postirradiation examination program. If the Fort St. Vrain facility is unavailable for this purpose, the need for such proof testing could possibly be met by other means. In any case, it would be necessary for DOE to fully demonstrate that any program developed would yield the equivalent in overall confidence that could be provided by Fort St. Vrain reactor tests.

DOE has acquired data from irradiation tests that provide conservative simulations of temperature gradients in the TRISO fuel particles. The fuel power densities achieved in these tests were considerably higher than those expected in normal operation to accelerate the irradiation process. This difference led to particle center temperatures as much as 250C° higher than surface temperatures, in contrast to the less than 10C° differences expected normally. Also, in the tests of fuel compacts, the compact centerline temperatures were much higher than the surface temperatures. In both instances, the surface temperatures were taken as the reference fuel exposure temperatures. While the staff notes the potential conservatism, there remains the added concern that use of data with several hundred degree systematic errors in the exposure temperatures may lead to masking other significant effects when other important fuel-failure model parameters are factored in. Furthermore, the accelerated irradiation may or may not produce accurate or conservative failure-model data, so the rate effects should be investigated systematically. For these reasons, it will be necessary for DOE to demonstrate at a later review stage that the same degree of confidence can be gained from the accelerated tests as from tests under normal conditions.

F. Fuel-Block Cracking Under Thermal Stress

Cracks in Fort St. Vrain fuel blocks due to unanticipated thermal stress were judged by the staff to be acceptable for that reactor because the cracks were seen as not affecting safety performance. Although calculations of stress-to-strength limits have been made for the MHTGR fuel which indicate a margin of safety against cracking, the PSID states the fuel design is not intended to preclude limited cracking. Cracking "modes" or types that could affect operation (cooling functions) or fuel handling are not expected, since the predicted stresses in the elements are a relatively small fraction of the strength. DOE has also noted that the graphite to be used in the MHTGR fuel elements is of a better grade (stronger) than that used at Fort St. Vrain. In response to the staff's concerns, DOE reworded its intentions about the design as follows:

> The fuel design criteria is intended to permit the probability of limited cracking even though expected stresses are limited to values less than the mean strength of the fuel element graphite. The probability of cracking is held to a low enough value such that the probability of functional damage to the fuel elements is within the risk allotment of the reactor core.

The staff has deferred judgment on the acceptability of fuel-block cracking in the MHTGR pending further experience with Fort St. Vrain.

G. Ability of Fuel To Withstand Accident-Induced Temperatures and Environments

DOE contractor report HTGR-85-107 (GA, 1985) notes, and the NRC staff agrees, that the fuel-failure model needs further evaluation with fuels of current design, particularly in the temperature range from 1200 to 1800°C, for which the amount of relevant testing data is limited. It is the staff's position that this need for additional testing is recognized in the RTDP but that the service conditions and testing approaches are not adequate with respect to the safety analyses and the need for defense-in-depth confirmation. In this respect, the staff notes that the range 1200 to 1800°C is the most important temperature range for the acceptance of the MHTGR, since this covers the full temperature range the fuel is predicted to experience during the anticipated and most of the worst-case safety-analysis scenarios. In response to Comment 15-6, however, DOE stated that near 2000°C, the onset of a rapid increase in the thermal decomposition of the silicon carbide coating is seen. Hence the staff believes that adequate data up to this decomposition temperature are needed to gain sufficient knowledge of failure mechanisms and margins. In response to Comment 4-45A, DOE stated that testing of older fuels with silicon carbide coatings deposited under the same conditions as those proposed for the MHTGR fuel has demonstrated the physical mechanisms that occur up to 2500°C and that no discontinuities in phenomena exist near 1800°C. The staff concluded, however, that this issue is particularly important for the new MHTGR fuel, where subtle changes in fuel design or manufacturing processes may influence performance.

Testing up to 2000°C should also provide data for fuel performance in both moist and oxidizing conditions. DOE has noted that all fuel-performance tests cited were run in a dry helium environment. Although DOE currently does not expect moisture or air environments during the heatup tests to affect the results, because the fuel-failure requirements are so stringent, performance in moist and air environments should be confirmed experimentally.

H. Effects of Fuel Composition on Performance

Because the failure mode of the silicon carbide layer at around 1600°C is stated to be internal chemical interaction, the fuel-failure model and experimental data must include fuel composition as an explicit parameter, and the effect of composition changes must be considered over the irradiation lifetime. Most of the data available relate to highly enriched fuel, while the MHTGR will utilize a fuel of 19.9 percent enrichment that will eventually contain significant quantities of plutonium. Although the limited data available on low-enriched uranium fuel indicate little difference between failure behavior for the high-enriched uranium fuel and low-enriched uranium fuel (with an expected 2-percent maximum plutonium content), this indication needs to be confirmed for oxide and carbide fuel mixtures. The RTDP states that the various fuel parameters are to be covered, but it implies that "representative sample" testing will suffice for the overall proof. The staff does not believe such an approach can be justified because there is need to cover, in a statistically valid way, the entire range of parameters and their combinations. The staff expects that a matrix of these validations will be planned and executed. DOE has made a commitment to tests of representative fuels at a full range of conditions in Amendment 9.

I. Effects of Exposure to External Chemical Attack on Fuel Performance

The "service conditions" listed in the RTDP with respect to the effects, or lack thereof, of chemicals to which the fuel may be externally exposed should be fully described and consistent with safety-analysis parameters. The chemicals to be considered must include, at the least, water vapor, oxygen, and nitrogen and those potentially arising from synergistic effects of trace chemicals and radiation. In consideration of long-term normal exposure, the quantities of impurities that are either necessary or must be avoided need to be determined. The planned experiments should take into account the full range of exposures to chemical attacks that can be derived from long-term normal operation, including anticipated operational occurrences, followed by the appropriate safety-analysis events described in Chapter 15. In response to Comment 4-51, DOE confirmed that such parameters will be tested under accelerated conditions it believes to be more extreme than normal conditions. The staff will defer judgment on this approach until more test results are available for reference fuel. At a later review stage, DOE will be expected to identify the ranges of impurities that must be kept from entering the system (for example, sodium hydroxide and chlorine).

J. Applicability of FRG Data

DOE has a long, established relationship with the Federal Republic of Germany (FRG) with respect to HTGR fuel development. It is important that the fuel-performance model be consistent with and confirmed by FRG data. The revised RTDP should augment the discussions of related FRG work and clarify both DOE's degree of dependence on the FRG program and the adherence of the FRG program to acceptable standards of quality assurance and documentation.

4.2.6 Conclusions

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 analysis,* but notes that this conclusion is dependent on the successful outcome of a research program that must be augmented beyond that currently described. Satisfactory resolution of the safety issues identified herein, together with completion of a satisfactorily augmented research program plan, will be necessary.

The staff notes that actual fuel performance at Fort St. Vrain and in the FRG reactors, together with reported laboratory and in-pile tests, gives promise that the performance objectives can eventually be demonstrated. Until substantial further work is done, however, the staff reserves final judgment on the acceptance of the goal of the fuel-coating barriers serving the containment function as proposed by DOE and considered by the staff in the Chapter 15 safety analyses.

4.3 Nuclear Design

The nuclear design of the MHTGR was reviewed with respect to the reactor core subsystem (RCSS) described in portions of Section 4.2 of the PSID and the neutron control subsystem (NCSS) described in PSID Section 4.3. The design and safety objectives for the NCSS are stated below, while previous sections of this chapter provide descriptions of the reactor core. The sections following on criteria, research and development, and safety issues discuss the reactor physics aspects of the nuclear designs included in both the RCSS and the NCSS.

4.3.1 Design Description and Safety Objectives

The NCSS serves the nuclear design objectives of the MHTGR by monitoring and controlling the neutron generation rate in the core, functions to control direct radiation exposure to operating personnel, and serves the fuel-handling system,

*Indicates statement is particularly sensitive to change by evaluation of forthcoming DOE information.

as described in Section 9.1. Although the core configuration is significantly different from that of 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 6 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 two 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 are clustered in each of the 12 outer neutron control assemblies (ONCAs). All these assemblies penetrate and are housed in the reactor-vessel upper head, as shown in Figure 4.1. Twelve symmetrically located columns of reserve shutdown fuel elements adjacent to the inner reflector contain an offcenter, 3.75-inch-diameter hole to allow the insertion of borated graphite pellets by actuation of the reserve shutdown control equipment (RSCE), if needed to ensure shutdown. The RSCE is part of the six INCAs and contains the pellets and release mechanisms in two hoppers within each INCA. Reactivity is also controlled by lumped burnable poison (LBP) rods in the corners of fuel elements, as previously described. Nuclear instrumentation consists of six exvessel neutron detector assemblies, three startup detector assemblies, and five invessel 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 subsystem, the investment protection subsystem, or the operator to bring the reactor ultimately to cold shutdown (that is, a refueling temperature of 192°C). The NCSS also has 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 are 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 derives from the negative reactivity input characteristic 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 stated by DOE to result in a power level somewhat less than 1 percent of full power with the reactor not in a sustained subcritical state. Rather, the reactor oscillates about a power level sufficient to maintain a temperature level such that the Doppler coefficient will bring the reactor to less than 1 percent power.

The equipment items DOE proposes to classify as safety related are (1) the control assembly structures, (2) the mechanical components of the control rod drives, (3) all portions of the RSCE, except indicators of condition, and (4) those portions of the exvessel detector assemblies that supply signals to the safety-grade portions of the plant protection and instrumentation system (PPIS) described in Section 7.2. Proposed to be classified as non-safety related are the instrumentation and controls for the safety-related equipment (except actuators for the RSCE) and the invessel flux-monitoring equipment. The startup detector equipment has radionuclide control functions for personnel protection in accordance with 10 CFR Part 20 requirements during startup and refueling for which DOE has made a commitment to meet the appropriate level of quality. This equipment is not needed for reactor protection because of the existence of the PPIS.

4.3.2 Scope of Review

In the review, the nuclear portions of the RCSS and the NCSS have been considered within the framework of the MHTGR nuclear design as a whole, and therefore the review has been based on comparing the relevant information provided in Sections 4.2, "Reactor Core Subsystem," and 4.3, "Neutron Control Subsystem (NCSS)," of the PSID and in the Probabilistic Risk Assessment with the information requested in Section 4.3, "Nuclear Design," of Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," and Sections 4.3, "Nuclear Design," and 4.6, "Functional Design of the Control Rod Drive System," of the Standard Review Plan (SRP). The review addressed (1) the compliance of the nuclear design of the RCSS and NCSS with NRC criteria and the DOE-proposed "10 CFR 100 Design Criteria," as well as appropriate industry criteria, (2) the calculation uncertainties, (3) the status of the validation of the nuclear design methods, (4) the predicted capability to achieve effective reactor shutdown, both inherently and actively by the NCSS, (5) steam and water ingress, (6) the status of the analysis of the reactor-vessel irradiation spectrum and intensity, (7) the status of the decay-heat models and rates, (8) the safety classification of the instrumentation, and (9) the mechanical design of the NCSS. A contractor report from Oak Ridge National Laboratory (Moses, 1988) provided a detailed review and evaluation of the MHTGR nuclear design and formed the major basis for this review.

The staff chose to defer to a later stage of development the review of five technology development needs (TDNs) in the Reactor Technology Development Plan (RTDP) pertaining to the reactor design. These are TDN 9-1, "Reserve Shutdown and Burnable Poison Material Process Development," TDN 9-2, "Corrosion Characteristics of Coated B_4C," TDN 9-3, "Corrosion Characteristics of Core Materials," TDN 10-2, "Guide Tubes Flow-Induced Vibration Testing," and TDN 10-3, "Neutron Control Assembly Flow and Leak Testing."

4.3.3 Review and Design Criteria

The staff has determined that the nuclear design requirements for the MHTGR are to be guided principally by Sections 4.3, "Nuclear Design," and 4.6, "Functional Design of the Control Rod Drive System," of the SRP and that the following general design criteria (GDC) are applicable (with minor exceptions that clearly pertain only to LWRs): GDC 10, "Reactor design," GDC 11, "Reactor inherent protection," GDC 12, "Suppression of reactor power oscillations," GDC 13, "Instrumentation and control," GDC 20, "Protection system functions," GDC 25 "Protection system requirements for reactivity control malfunctions," GDC 26 "Reactivity control system redundancy and capability," GDC 27, "Combined reactivity control systems capability," and GDC 28, "Reactivity limits." In Amendment 1 to the PSID, DOE stated that the MHTGR will meet the intent of these criteria. This statement is acceptable to the staff for the current stage of review.

In addition, DOE has proposed a set of "10 CFR 100 Design Criteria." These criteria have some useful applications to the conceptual nuclear design but are not as comprehensive as the GDC and are not sufficient to ensure adequate defense-in-depth. DOE will need to complete specifications for the MHTGR principal design criteria if such criteria are to be considered in lieu of meeting the current GDC cited above. Further, DOE has not yet proposed lower-level design criteria for meeting the dose limits of 10 CFR Part 20 and the requirements of 40 CFR Part 190.

Although current regulations do not stipulate specific industry standards for the nuclear design process, DOE has made a commitment to validate the MHTGR nuclear design in accordance with certain applicable regulations and industry standards, specifically, ANSI/ASME NQA-1, "Quality Assurance Program Requirements for Nuclear Facilities," and certain of the ANSI/ANS-19-series standards, namely, ANSI/ANS-19.1, "Nuclear Data Sets for Reactor Design Calculations," and ANSI/ANS-19.3, "Determination of Neutron Reaction Rate Distributions and Reactivity of Nuclear Reactors." Although DOE did not specifically commit, at this time, to use ANSI/ANS-19.4, "A Guide for Acquisition and Documentation of Reference Power Reactor Physics Measurements for Nuclear Analysis Verification," or ANSI/ANS-19.5, "Requirements for Reference Reactor Physics Measurements," in response to Comment 4-39, it stated that these standards have been identified for probable applicability as the design develops. These documents relate to the quality assurance requirements and level of documentation for acceptable experimental data to be used in the validation of nuclear design methods and models. DOE's intentions with regard to the quality and documentation of acceptance criteria for experimental data used for verification and validation should be clarified with regard to meeting the intent of Regulatory Guide 1.68, "Initial Test Program for Water Cooled Reactor Power Plants," before the submittal of the preliminary standard safety analysis report (PSSAR) or any topical report on the MHTGR nuclear design.

In addition to the criteria identified above, 10 CFR Part 100, Appendix A, Section III (c), defines safety-related structures, systems, and components as those necessary to ensure the capability to shut down the reactor and maintain it in a safe shutdown condition and to prevent or mitigate the consequences of accidents that challenge the fission-product barriers. The nuclear design of the RCSS and NCSS must meet these design criteria, particularly with regard to ensuring the capability to achieve hot-shutdown margins in response to both normal and abnormal operating conditions in order to preclude the generation of excessive loads that could challenge the integrity of the fission-product barriers.

The nuclear design criteria embodied in the SRP were found appropriate for staff review with the result that the staff recommends that the PSSAR be organized in accordance with Sections 4.3 and 4.6 of the SRP rather than containing this material in separate sections now titled RCSS and NCSS. Before the submittal of a PSSAR or a topical report on the nuclear design, the applicant should develop detailed commitments in accordance with Section 4.3 of the SRP for meeting industry standards and provide appropriate criteria for analytical models and methods for obtaining shutdown margins, core-heatup and decay-heat analyses, and lower-level criteria pertaining to 40 CFR Part 190 and 10 CFR Part 20.

4.3.4 Research and Development

In response to Comment 4-41, DOE made a commitment to provide improvements to the reactor physics data base and the validation of reactor physics codes in a program to be included as a chapter in the RTDP. As part of this program, cooperative programs to utilize data from the Arbeitsgemeinschaft Versuchs Reaktor (AVR) experimental reactor facility in the Federal Republic of Germany and the VHTR-C critical experiments facility in Japan are stated by DOE as being actively pursued.

The validation program proposed in conjunction with the AVR will include demonstrations of inherent safety features (for example, anticipated transient without scram tests), control rod worth measurements, temperature-coefficient measurements, reactor-transient responses, and plutonium buildup in the fuel during irradiation. For the VHTR-C, the program will include criticality measurements, temperature-coefficient measurements, neutron-flux distributions, and fixed burnable poison worth measurements.

The RTDP for reactor physics should be presented to the staff before the PSSAR review stage so that adequate data will be available and appropriate documentation and analyses of the data will be provided or referenced in the PSSAR. Specific technical areas that should be addressed include the following:

(1) Reflector control worth in the unique, tall, annular core configuration, including the effects of control-worth reduction due to water ingress and core cooldown.

(2) The reactivity worth of steam and water ingress into the hot and cold cores, including mechanistic accounting for the amount and rate of moisture diffusion into graphite.

(3) The sensitivity of safety-related and other nuclear instrumentation to neutron attenuation due to water ingress, as well as to core-loading distribution, control-rod position, core cooldown, and where applicable, startup-source location.

(4) The negativity of the prompt and near-prompt temperature coefficients (vs temperature) of reactivity near the end of the cycle when contributions due to plutonium-239 buildup are expected to be most positive.

(5) The capability to detect axial power distribution anomalies in the core during power operation.

(6) Data and analytical techniques to validate estimates of the amount and spectral distribution of vessel fluence. These should incorporate, as appropriate, the methodologies and techniques for vessel-fluence analysis and neutron dosimetry that are being developed from the research on heavy section steel technology sponsored by NRC at Oak Ridge National Laboratory.

(7) Comments provided in Section 4.3.5.F with respect to the review and validation of decay-heat data and methods, plus inclusion of the improved treatment of activities and experimental results that may be available from the AVR test program.

The staff notes that the cooperative programs outlined by DOE do not explicitly identify experiments related to steam and water concerns, vessel fluence, and nuclear instrumentation. The reactor physics chapter of the RTDP is expected to address how these items will be considered and clearly identify for all items whether the technology need will be met by cooperative programs, utilization of existing data, new analyses, new experiments, qualification testing, or pre-operational and startup reactor testing, or a combination of these methods. DOE's commitment to develop a chapter in the RTDP is an acceptable approach toward satisfying the outstanding research and development needs for the nuclear design of the MHTGR.

4.3.5 Safety Issues

The safety issues relate primarily to demonstrating that the nuclear design will perform as predicted or assumed in the PSID accident analyses. This demonstration requires the validation and documentation of nuclear design methods and the adequacy of safety-related and other nuclear process instrumentation. Also, assurance of shutdown margins given the possibility of adverse single failures in the RSCE, as well as the operating bypasses that can inhibit scrams of either the inner control rods or the RSCE, must be addressed. Finally, the accuracy and uncertainty of nuclear design parameters, principally the decay heat, that affect the results of "best-estimate" core-heatup evaluation models must be assessed against experimental data and appropriately documented.

A. Calculational Uncertainties

In Comment G-7.B the staff requested information on uncertainties associated with the calculation of control rod and reserve shutdown system worths, reactivity feedbacks, plutonium buildup, steam- and water-ingress effects, and power distributions, and in Comment G-7.D the staff requested a description of how these uncertainties were factored into the MHTGR safety analysis and the resulting conclusions regarding offsite doses. DOE responded directly to each of these comments and later provided a document entitled "MHTGR Core Nuclear Uncertainty Analysis" (GA, 1987-4). The responses and the document discussed the uncertainties in the physics parameters and significantly indicated that large uncertainties in the reactivity temperature coefficients (a 50-percent reduction in the calculated moderator temperature coefficient coupled with a 10-percent reduction in the Doppler coefficient) can be tolerated with almost no effect on peak fuel temperatures and offsite release during the events postulated in the safety analysis described in Section 15.2. The core reactivity temperature coefficients were also indicated to be relatively insensitive to uncertainties in the important plutonium nuclides.

The staff finds DOE's analyses of the effects and consequences of reactor physics uncertainties encouraging at this conceptual design stage, and a useful illustration of the passive and inherent physics parameters protecting the MHTGR core against various postulated transients. Because of the need to validate both the analytical methods and the experimental data base supporting these methods, as discussed below, the staff finds that the uncertainty analysis will have to be performed again at a later design stage using acceptable, validated models. Furthermore, the analysis will have to consider event initiators and sequences consistent with the bounding events (BEs) described in Section 15.2.3.3.

B. Methods and Data Validation

The staff did not review the document referenced in the PSID (Merrill, 1973) to describe the basic methods used in the nuclear design of the MHTGR because this document was determined not to be applicable to the MHTGR design. Rather the staff concentrated its review on the data base needed to support the development of an acceptable methodology for the MHTGR nuclear design.

An indepth assessment was made (Moses, 1988) of what should be considered in the development of an acceptable methodology, and a comprehensive review was performed of the experimental data bases provided in response to Comments G-7.C and 4-15. With regard to the design methods, the staff found that the information presented or referenced in the PSID did not adequately demonstrate the ability of the nuclear analysis to predict the safety-significant reactor physics parameters without the addition of much larger uncertainties than those assumed in the uncertainty analysis described above. Furthermore, in order to perform an improved uncertainty analysis at a later review stage, it will be necessary for DOE to develop or confirm methods that employ standards for quality assurance as described in Section 4.3.3.

The experimental data reviewed consisted of reactor data from Peach Bottom 1 and Fort St. Vrain and critical-experiment data obtained by Gulf General Atomic in the 1960's in connection with its large HTGR development program, as well as by Battelle Northwest Laboratories as part of a USAEC-funded program (HTLTR). In addition, low-enriched uranium critical-experiment data from France (MARIUS III) and the United Kingdom (HITREX) were reviewed. The staff found that (1) there is a paucity of relevant experimental data and (2) there is a lack of documented analysis of the existing data using the analytical methods employed for the MHTGR nuclear design. For similar reasons, the staff found comparisons with the British computer code WIMSD not a basis for acceptable validation.

As a result of this review and DOE's reevaluation, DOE changed its original position on research needs and now plans to develop a chapter on reactor physics in the RDTP, as described in Section 4.3.4. The end product of this program should be adequate integral data for the construction and validation of an acceptable methodology for the MHTGR nuclear design.

C. Reactor Shutdown

The provisions for hot and cold reactor shutdown are judged to be acceptable for the conceptual design stage but are subject to further discussion and reconsideration at later design stages when additional physics information will be available from the RTDP. This will include consideration of the performance of the inherent shutdown mechanisms for the postulated bounding events described in the Chapter 15 safety analyses. Final acceptance will be subject to successful preoperational testing and plant operational experience.

It is evident that the combination of the two safety-grade mechanical shutdown means, together with the inherent mechanisms available to achieve hot shutdown, should provide the MHTGR a degree of protection against reactivity transients greater than that available for LWRs. However, it should be noted that negative reactivity insertion beyond the outer neutron control assemblies is needed to bring the reactor to a sustained state of hot shutdown about 37 hours later in the accident-event sequences after xenon-135 decay and cooldown of the core, and eventually to achieve cold shutdown. For this reason, the staff believes that the RSCE must meet the single-failure criterion of SRP Section 4.6 and, if total automatic shutdown is to be claimed by DOE, the RSCE must be free of any manual actuation requirements, including the removal of inhibit signals, to achieve cold shutdown with adequate margin for all the bounding-event sequences given in Section 15.2.

Correspondingly, if DOE is to claim that the inner neutron control assemblies provide the additional reactivity necessary to compensate for xenon-135 decay, further discussion is needed at a later review stage regarding the removal of inhibit signals when the automatic insertion of the inner control rods is needed. Currently, in accordance with DOE's response to Comment 4-21, a manual scram would be needed to overcome an inhibit signal provided for the investment protection of the inner rods entering the inner reflector when at high temperature. At present, this appears to be a situation inconsistent with DOE's claim of fully automatic reactor shutdown and is discussed further in Section 13.2. If DOE wishes to claim fully automatic shutdown, the staff believes that, at a later design stage, an acceptable automatic system should be designed to remove any administrative and operator actions now apparent and to achieve the necessary shutdown margins. Such a system would necessarily be classified as safety related and subject to meeting appropriate Institute of Electrical and Electronics Engineers criteria in this regard.

D. Steam or Water Ingress

Achievement of acceptable shutdown margins under cases of postulated reactivity addition by steam or water ingress places greater reliance on the availability of the inner control rods and the RSCE than for dry conditions. The magnitude of this greater reliance cannot, however, be quantified because two different models are now being considered to represent reactivity additions from postulated steam- or water-ingress events. For the method used by DOE contractor General Atomics, the water concentration in the core and reflector was based on the absorption of an equilibrium amount of water in graphite pores. Other workers at Oak Ridge National Laboratory (ORNL) (Cleveland, 1988) believe sufficient time will not be available for significant absorption to occur during the event of interest and recommend that lower reactivity insertions should be calculated for the event. The staff expects that a suitable model for steam or water reactivity insertion should be achieved experimentally as part of the development program described in Section 4.3.4. Results of the independent analysis of the reacti-

F. Decay Heat

Because adequate knowledge of the decay-heat rate is essential in estimating margins for reactor cavity cooling system (RCCS) performance, the staff has rigorously reviewed the decay-heat model and rates proposed. A set of preliminary review criteria was developed for this purpose in an ORNL contractor report (Moses, 1988) to guide the decay-heat review at a later review stage, consistent with available guidance for LWRs. The ORNL report also pointed out that assuming a uniform spacial distribution of decay heat could be nonconservative from the standpoint of shutdown margins. This point will be considered at a later review stage.

Although the decay-heat evaluation models do not currently meet the preliminary review criteria, the staff has concluded that the decay-heat rates presented in the PSID are acceptable for use in the conceptual design and analysis of the RCCS, as discussed in Section 5.5.

G. Safety Classification

In response to Comment 4-43, DOE stated that the MHTGR flux-mapping detectors are included for economic reasons and should not be considered as safety related with respect to monitoring burnup effects and ensuring that undesirable fuel temperatures do not occur in lower core regions. The staff accepts DOE's position that calculational techniques for fuel management are adequate if acceptably validated; however, it is directed in Sections 4.3.4.C and E that the RTDP address the MHTGR's capability to detect axial power distribution anomalies in the tall annular core during power operation.

The staff has deferred to a later review stage the question of the non-safety-related classification proposed for the inner control rods and the equipment that monitors the status of plant protection systems. The staff decision is that this matter will be based largely on the results of research and testing programs planned to demonstrate the adequacy of the shutdown margins provided inherently and mechanically by reactor design.

H. Mechanical Design of Control Rods, Drive Systems, and Reserve-Shutdown Control Equipment

In response to Comment 4-24, DOE identified the similarities and differences between the Fort St. Vrain control rod system and the reserve shutdown control equipment (RSCE) of the MHTGR, with inclusion of a discussion of the "lessons learned" from Fort St. Vrain operations. Because of (1) the overall similarities of the MHTGR design and the successful design features of the Fort St. Vrain equipment and (2) the identified improvements to the MHTGR based on Fort St. Vrain "lessons learned," the staff has confidence that a satisfactory level of mechanical performance can be achieved. At the PSSAR review stage, the applicant should present for the staff's review a plan for qualification, preoperational, and startup testing for this equipment. Both DOE and ORNL calculate that the rapid ejection of a control rod could cause the reactor to go prompt critical. For this reason the potential for rod ejection from the MHTGR must be precluded by design as it is for Fort St. Vrain.

I. Qualification and Startup Testing

With respect to transients and the events postulated in the safety analyses, the staff finds that in order to meet the requirements of GDC 27 and to accept DOE's position that steam and water ingress pose no threat to fuel integrity due to reactivity effects, the nuclear design of the MHTGR needs to be supported by a program of appropriate qualification, preoperational, and/or startup testing, as follows:

(1) Verify the assumptions of system performance capabilities to limit the amount and rate of water ingress to that predicted in the PSID and used in the safety analyses.

(2) Verify the operability of the control rod drive equipment, reserve shutdown control equipment, and supporting instrumentation under conditions predicted to occur during the transients postulated in both the PSID and the Probabilistic Risk Assessment.

(3) Validate the analytical tools used to predict reflector control rod worths in the annular core configuration, the reactivity worth of reserve shutdown material in the annular core configuration, and the reactivity worth of water (hydrogen) in the low-enriched-uranium annular core.

(4) Verify the mechanical design aspects of the NCSS in accordance with Section 4.3.5.H.

4.3.6 Conclusions

The conceptual nuclear designs of the RCSS and the NCSS are conditionally acceptable given that adequate resolutions are provided in the PSSAR regarding the criteria, research needs, and safety issues, as identified herein.

DOE's commitment to develop a chapter on reactor physics in the RTDP is an acceptable approach to resolve many of the nuclear-design safety issues. This plan should be presented for staff review before PSSAR review so that the plan can serve as a supporting document for the review rather than become a document developed during the review.

The staff review found that the present design is not wholly in keeping with DOE's fully automatic control philosophy in that manual actions appear necessary to achieve sustained hot shutdown, cold shutdown, and the insertion of the inner control rods, if needed. The use of manual actions for safety purposes is discussed in Section 13.2 of this report, and DOE's final position on automatic control should be developed consistent with the staff position given in that section.

4.4 Thermal and Fluid-Flow Design

4.4.1 Design Description and Safety Objectives

The essential features of the helium-cooling design are described in Section 4.1. Emergency heat removal from the core by convection, conduction, and radiation from the reactor vessel is described in the context of the reactor cavity cooling system discussion in Section 5.5. The safety objective of the thermal and fluid-flow design is to ensure that for forced helium cooling the 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.

4.4.2 Scope of Review

The review performed by the staff focused on Section 4.2 of the PSID, the relevant portions of the Probabilistic Risk Assessment, and DOE's responses to the staff's comments. Insight into the design features and safety characteristics was also provided by the staff's contractors at ORNL and Brookhaven National Laboratory, who developed independent thermal and fluid-flow models for the reactor and used these models for a spectrum of transient analyses, as summarized in Section 15.4 of this report, with more detail given in Appendixes A and B. The staff requested additional information about the sensitivity of the core response and safety margins to the uncertainties in core flow distributions (including bypass flows), coolant-channel flow blockage (including effects of fuel-element loading misplacements), hot streaks, and laminar flows. The staff also considered relevant Fort St. Vrain experience and other data.

4.4.3 Review and Design Criteria

Because of the fundamental differences between helium and water coolants and HTGR and LWR fuels, only the most general criteria exist for the MHTGR thermal and fluid-flow review and design at present. The NRC Standard Review Plan (SRP), Section 4.4, "Thermal and Hydraulic Design," and related requirements in 10 CFR Part 50 provide no specific guidance for gas-cooled reactors.

The principal design criteria generally applicable to the core thermal and fluid-flow design are GDC 10 ("Reactor design"), GDC 11 ("Reactor inherent protection"), GDC 12 ("Suppression of reactor power oscillations"), and GDC 13 ("Instrumentation and control"). Some guidance for later design stages may be found in certain regulatory guides (RGs) to which DOE has made commitments, such as RG 1.20, "Vibration Assessment During Preoperational and Initial Startup Testing," and RG 1.333, "Loose Parts Detection for the Primary System of Light-Water-Cooled Reactors." DOE is assessing RG 1.68, "Initial Test Program for Water Cooled Reactor Power Plants," as the design develops.

DOE has proposed the following criteria for functional areas for both normal and safety-related operations:

(1) Normal: Maintain geometry for heat-transfer and coolant-flow control.

(2) 10 CFR Part 100-Related: Maintain geometry for conduction and radiation.

The staff has concluded that the design criteria to be used for the core thermal and fluid-flow design are acceptable for this stage of the design. Additional criteria are expected to be developed and approved as a consequence of the resolution of the identified safety issues and to be eventually contained in a format analogous to Section 4.4, "Thermal and Hydraulic Design," of the SRP for LWRs. DOE's commitments to further criteria development, as given in response to Comment G.3-1, are acceptable to the staff.

4.4.4 Research and Development

Except for the flow-induced vibration testing of control rod guide tubes discussed in Section 10 of the RTDP, there are no other issues similarly identified as relating to the core thermal and fluid-flow design. Outside the scope of the RTDP, flow-modeling tests are planned by DOE to assess the effects on flow distributions of machining tolerances, thermal expansion, and irradiation-induced dimensional changes of graphite, and to evaluate the effects of flow distributions on the reactor design. In this regard, the modeling tests will investigate tendencies for fluid mechanically induced flow oscillations and fuel-block displacements, and will be used to help select gap configurations and determine core-pressure-loss coefficients. At a later review stage, DOE will describe and the staff will assess the scope, indicated results, and adequacy of the flow-modeling tests as they pertain to thermal and structural integrity, validation of analytical tools and models, and the safety analysis. As this assessment progresses, those portions of the flow-modeling tests judged to be safety related will be included within the scope of the RTDP. Since there is a similar effort currently under way in Japan at the JAERI HENDL loop, the staff recommends DOE consideration of collaborative efforts in this area. Also at a later review stage, DOE is expected to propose a preoperational and start-up test program to confirm the conclusions of the modeling tests and the vibration tests for the control rods.

4.4.5 Safety Issues

The staff has general concerns about the uncertainties in the thermal and fluid-flow analysis and the sensitivities of the structural integrity margins and the plant safety analyses to these uncertainties. Accordingly, DOE was requested to provide a basis for its uncertainty estimates and an assessment of the margins that are expected to be available in the design before the occurrence of unacceptable structural damage or unacceptable releases of radionuclides. Specific issues are addressed below:

A. Core Flow Distributions

DOE has made a commitment to evaluate, at a later review stage, the effects of core flow distributions on core-exit hot streaking, local fuel temperatures, and the structural integrities of the core support structure and the hot duct. Final resolution of these concerns is dependent on the satisfactory completion of the flow-modeling tests discussed in Section 4.4.4. This and the related concerns given below may require invessel temperature modeling if the flow-modeling tests are not satisfactory.

B. Flow Oscillations and Fuel-Block Displacement

Fluid mechanical forces were found in the Fort St. Vrain reactor to cause variations in flow and coolant temperature as a result of periodic motions of some of the fuel-element stacks. The problem was resolved by mechanical restraints at the tops of the fuel columns. This safety issue is recognized in the MHTGR by the mechanical design of the upper-plenum elements and by the flow-modeling tests described in Section 4.4.4.

C. Flow Blockage

In response to Comment 4-13, DOE clarified its discussion of coolant-channel flow blockage given in the PSID. The likelihood of blockage of a coolant channel is considered to be remote, but it could possibly be caused by foreign materials circulating with the coolant, such as fibrous thermal insulation. DOE stated that if total blockage occurred in a channel, elevated temperatures could result and some local failures of fuel-particle coatings and a release of radionuclides to the coolant could occur. If blockage became extensive, significant fuel failure would be detected by gaseous fission product activity monitors in the primary system. The reactor could then be shut down and the damaged fuel replaced.

In Comment 4-14, the staff requested that DOE consider misloading of reflector elements, particularly as a potential cause of control-rod-channel blockage, but also as a cause of fuel-channel blockage. DOE included in its response a description of the MHTGR's core-loading procedure, which has many features similar to those used at Fort St. Vrain and is designed to prevent such occurrences.

On the basis that fuel-channel blockage would be detected before any significant fission-product release and that Fort St. Vrain experience has resulted in no operational symptoms of either coolant-channel blockage or misloading errors, the staff found the MHTGR's provisions against the occurrence of flow-channel blockage acceptable at this conceptual design stage, except as discussed in Section 4.4.6 with respect to temperature monitoring.

D. Hot Streaks

Helium hot streaks may occur during normal operation in the lower plenum and cause local temperature elevations on the graphite support structure, the hot duct and, potentially, the steam generator. And, under postulated conditions of a pressurized loss of forced convection, naturally convective hot streaks impinging on the upper plenum thermal protection structure (UPTPS) are a cause for concern that was discussed in response to Comments G-8A and G-8B. The staff's conclusions and reservations with respect to the effects of hot streaks on the reactor internals are presented in Section 4.5.5.E. The possible effects of hot streaks on the steam generator and the potential means to control hot streaking, such as flow spoilers in the crossduct, will be considered by the staff at a later review stage.

E. Laminar-Flow Effects

Laminar-flow concerns, noted in Comment 4-29, pertained to the potential conditions at low power and low flow that could lead to flow stagnation and reversal. In response, DOE stated that to ensure that flow reversal does not occur, a minimum core flow rate will be specified as a function of core power level. This is the same as the approach found to be acceptable for Fort St. Vrain, but there may be more of a problem because of the greater buoyancy from the taller core and the lack of orifice control.

It is also of concern that laminar hot streaks typically persist to much greater lengths than turbulent streaks. Consequently, the conditions that would cause laminar streaking and the consequences if such streaking occurred will require later review.

4.4.6 Conclusions

To address the concerns about helium flow distributions, hot streaking, and core power distribution and to provide additional assurance against the possibility of massive flow blockage, such as could be caused by fuel-element misplacement, it is currently the staff's position that invessel temperature measurements should be provided. Without such measurements there would be excessive uncertainty in these areas of concern.

The staff has deferred its review of the flow-modeling experiments to a later design stage when more information can be available and has not yet required that they be part of the RTDP. The scope and results of these experiments will be significant, however, with regard to the staff's confidence in the MHTGR thermal and fluid-flow safety performance. These experiments should be carefully designed and performed and should include all relevant phenomena, including the conditions for and effects of transitions between turbulent and laminar flows. Fully satisfactory tests with complete validation of analytical tools and models may cause the staff to consider revising its position with respect to the need for invessel temperature measurements.

The staff concludes that progress by DOE in thermal and fluid-flow design is satisfactory for the conceptual stage of the design, and that all the safety issues can be resolved by further work in the areas identified.

4.5 Reactor Internals

4.5.1 Design Description and Safety Objectives

The reactor internals 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 safety design objectives are to provide for normal and abnormal thermal loadings; thermal expansions and stresses; mechanical, fluid, and seismic loadings; and resistance to corrosive impurities in the helium coolant. The metallic core internals 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 in progression from bottom to top. The significant aspects of the design are illustrated in Figure 4.1.

Metallic Core Support Structure

The metallic core support structure (MCSS) is a type 2-1/4 Cr-1 Mo steel structure having the form of a spoked wheel that rests on a ring forging. This forging is integral with the reactor vessel and all major loads are transferred to the vessel through this support. The MCSS safety and performance objectives are to support the other core internals and the reactor fuel, provide certain ducting for the reactor coolant, and maintain structural integrity during the transients postulated in the safety analysis.

Core Lateral Restraint

The core lateral restraint (CLR) is a group of metallic structures, all of Alloy 800H, located between the reactor vessel and the graphite permanent side reflectors. The group consists of the core barrel, seismic keys, coolant channels, and the hot-duct boss. The safety objective of the design of 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

The graphite core support structure (GCSS) is an arrangement of graphite posts and blocks that provide a lower plenum and a hot-leg path for the primary coolant and support for 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 safety and performance objectives of the GCSS are to support the core and inner reflector elements, provide for helium exiting the core and entering the hot duct, and maintain structural integrity during postulated transients.

Permanent Side Reflectors

The permanent side reflectors (PSRs), formed by axial columns of keyed or pinned stacked graphite blocks, extend over the full length of the core and, except in the region of the hot duct, extend to and are supported by alumina pads on the MCSS. The safety functions of the PSRs are to 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 are imbedded in the PSRs for the latter purpose.

Hot Duct

The hot duct is an Alloy 800H pipe-like structure that carries hot helium from the lower plenum to the steam generator vessel. It is located within the cross-duct vessel (see Section 5.2), and its exterior is surrounded by coaxial flow of cold-leg helium. For installation, removal, and maintenance, the hot duct is formed of two straight (horizontal) sections and a curved elbow section (with expansion below) for vertical attachment to the steam generator. DOE states that the hot duct is not required to be safety related.

Upper Plenum Thermal Protection Structure

The upper plenum thermal protection structure (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. Its safety objective is to protect the upper reactor vessel from excessively high temperatures during the postulated events. Flow-induced vibrations have been considered in its design with respect to its function.

4.5.2 Scope of Review

In performing this review the staff discussed with DOE the design criteria and safety analysis given in Section 4.4 of the PSID, relevant portions of the Probabilistic Risk Assessment, and the proposed research and development described in Section 7 of the RTDP. The staff requested additional information in areas of hot streaking, mechanical and structural requirements, moisture-ingress potentials, flow distributions, laminar flows, the significance of Fort St. Vrain fuel-element cracking, clarifications in design details, effect of neutron fluence, buildup of Wigner energy, seismic methodologies and fragilities, use of ASME codes, use of NRC regulatory and guidance documents, and applicable industry codes and standards. No independent calculations were performed by the staff or its contractors on the performance of the components described in this section. Staff resources did not permit the utilization of reviewers expert in the topics of structural graphites and mechnical design to help identify safety issues or the resolution of those safety issues thus far identified by the staff.

4.5.3 Review and Design Criteria

Consistent with the functional requirements of SRP Section 4.5.2, "Reactor Internal and Core Support Material," the principal design criteria considered by the staff in its review of reactor internals were GDC 1, "Quality standards and records" and 10 CFR 50.55a, "Codes and Standards." DOE has made a commitment to meet the intent of GDC 1 for items it has classified as safety related. DOE is also committed to the use of suitable versions of the ASME Code and thus will satisfy conformance with 10 CFR 50.55a.

DOE identified all the reactor internals as safety related, with the exception of the hot duct. The metallic components will be designed to ASME Code, Section III, Division 1, requirements. ASME Code Case N-47 will be used for design of the hot duct and the UPTPS. For the MCSS and the CLR, ASME Code, Subsection NG, Code Case N-201-1, will be used. Graphite internals will be designed to ASME Code, Section III, Division 2, Subsection CE, requirements.

DOE stated in response to Comment 4-37 that all safety-related internals will be inspected and surveyed in accordance with the intent of the ASME Code, Section XI, Division 2. Specifically, 25 percent of the accessible areas will be inspected four times during the plant lifetime, and the PSRs, GCSS, and MCSS will be subjected to materials surveillance programs in which coupons removed four times during plant life will be tested to determine tensile strength, fatigue strength, and impact properties, and will be metallurgically examined. All graphite structural components can be removed and replaced, if necessary. DOE has also made a commitment to meet the intent of the following regulatory guides in the design of the reactor:

1.20 Vibration Assessment During Preoperational and Initial Startup Testing (Rev. 2, May 1976)

1.29 Seismic Design Classification (Rev. 3, September 1978)

1.84 Design and Fabrication Code Acceptability, ASME Section III, Division 1 (Rev. 24, June 1986)

1.85 Materials Code Acceptability, ASME Section III, Division 1 (Rev. 24, June 1986)

1.87 Guidance for Construction of Class 1 Components in Elevated Temperature Reactors (Rev. 1, June 1976)

1.92 Combining Model Responses and Spatial Components in Seismic Response Analysis (Rev. 1, February 1976)

1.133 Loose Part Detection Program for Primary System (Rev. 1, May 1981)

DOE stated in the PSID that its "10 CFR 100 Design Criteria" II and IV are applicable to reactor internals. The staff views these criteria as too general and has not and does not plan to utilize them in its review of the reactor internals. In its response to Comment 4-27, however, DOE identified the following functional areas for normal and 10 CFR Part 100-related operations:

(1) Normal: Maintain geometry for heat transfer and coolant-flow control, and maintain geometry for positioning movable poison.

(2) 10 CFR Part 100-Related: Maintain geometry for conduction and radiation, and maintain geometry for insertion of movable poison.

The staff found these to be useful design criteria and considered them in the safety review.

4.5.4 Research and Development

Section 7, "Graphite Technology Development Plan," of the RTDP describes two programs for the development of core-support graphite. The first, TDN 7-1, will provide additional uniaxial strength tests for the Stackpole grade 2020 graphite. These additional tests of unirradiated, large specimens in air will be performed to meet ASME Code statistical requirements for the establishment of a minimum ultimate strength. A full statistical data base is needed and will include special and orientational variabilities from billet to billet and lot to lot. The second program, TDN 7-2, will provide additional data describing the corrosion of 2020 graphite by coolant impurities during normal operation and moisture-ingress events. Accelerated tests will be used with the aim of determining reaction kinetics as a function of temperature, helium impurity concentrations, pressure, and time. Measurements of strength loss of oxidized specimens will be made to confirm that no abnormal or unexpected strength loss with oxidation is evident. DOE stated that corrosion and burnoff problems are of concern mainly in locations of hot-streak impingment.

DOE has stated that it has plans for other programs of development and testing but does not consider them to be safety related. These include the flow-modeling test discussed in Section 4.4.4 and the hot-duct validation program mentioned in response to Comment 4-39. These research and development programs could add information in important areas and should be addressed at the PSSAR stage.

4.5.5 Safety Issues

Since it is not intended that any of the reactor internals be replaced during the 40-year lifetime of the plant, assurances of the long-term integrity and integrity under events postulated by the safety analysis provide the focus for formulation of the safety issues. The issues identified below address concerns of design, design criteria, the development program, and performance under normal, offnormal and aging conditions.

A. Design Codes

For metals, the available codes and code cases are expected to be generally sufficient, although additional review and additional data will be required, as discussed for the reactor vessel in Section 5.2. For graphites, proposed ASME codes have not been formally reviewed and approved by the staff. This should be accomplished as part of a PSSAR review, including review by consultants expert in graphite technology.

B. Hot-Duct Safety Classification

The design of the hot duct has not been sufficiently described and reviewed for the staff to conclude that its failure consequences could not impair the functioning of equipment important to safety. Since the design of the hot duct permits inspection and replacement, however, the staff finds it acceptable for the present stage of review, with the question of its safety classification to remain open until the PSSAR review stage.

C. Seismic Design and Fragility Data

As concluded in Section 3.5.6, general provisions for the seismic design of the MHTGR are acceptable for the conceptual review stage. Additional information on analysis methodology and materials data will be needed, however, at the PSSAR review stage. Because of the uniqueness of certain structural components, it may be necessary to develop acceptable analytical modes based on testing.

D. Graphite Corrosion

Review of the adequacy of the research program will be performed at the PSSAR stage. This is not expected to be a fundamental issue in the design.

E. Thermal and Fluid Mechanical Effects

The staff believes that the effects of hot streaks have been adequately considered in the design of the hot duct and UPTPS at the conceptual design stage. These effects should be reconsidered, however, at the time of the PSSAR review when more design detail and the results of the flow-modeling tests will be available. In addition, the analytical methods and supporting data base should be evaluated by independent analysis. The potentials for and the consequence of thermal stresses involving both metallic and graphite components have not been considered in the conceptual review because such effects are usually the result of local design details and flow distributions not available at this stage of the design (for example, temperature distributions as a function of transients in the region of the hot-duct entrance). Information on the effects of flow-induced vibrations and shear forces (from rapid depressurization accidents) has not been developed at this stage of review. Such effects have been considered, however, in previous HTGR designs with respect to the integrity of the thermal barrier, and acceptable design solutions are believed to be available. This topic should be considered at the PSSAR review stage.

F. <u>Cyclic Stresses and Displacements</u>

The potential for and the consequences of cyclic thermal stress and component displacements that might bear on the structural integrity of the reactor internals have not yet been considered in the staff review. Additional information will be requested from DOE to address this issue at the PSSAR review stage.

G. <u>Inservice Deterioration of Materials</u>

Additional information pertaining to inservice deterioration will be needed with respect to coolant chemistry, materials selection, fabrication practices, cycle and transient loadings, incipient failure mechanisms, and examination and testing procedures at the time of the PSSAR review. The staff believes that much acceptable information pertaining to the potential for inservice deterioration exists but has not yet been formally organized for MHTGR applications. Such formal organization could prevent redundant research programs and direct research to higher priority needs.

H. <u>Inservice Inspection Code Development</u>

NRC has not formally accepted ASME Code, Section XI, Division 2, although portions of this code are being used in upgrading the Fort St. Vrain inservice inspection and testing program. Final development of this code for staff approval should include the development of knowledge of the type indicated in Section 4.5.5.G and review by experts in high-temperature metals and structural graphites.

4.5.6 Conclusions

The staff concludes that the design criteria to be used for the reactor internals are in an acceptable stage of development for a conceptual review and that DOE's commitments to and participation in further criteria development provide an acceptable approach at this time. Completion of criteria development as identified in Sections 4.5.5.A, G, and H will be required during the PSSAR review.

The staff believes that, for structural graphites, the most important needs have been identified, but the programs described may not be sufficiently comprehensive to meet these needs. Additional discussion was provided in Sections 4.5.5.C, D, E, and G. Final agreement on the scope of the research and development remains an open item; however, resolution is not necessary to assess the feasibility of the MHTGR concept.

The safety issues identified indicate that the MHTGR reactor internals design requires the use of high-temperature materials in structural configurations and thermal environments for which limited experience is available and long-term deterioration must be precluded. The staff review has not been able to resolve finally any of these issues, but good progress has been made on most. This progress has been sufficient for the staff to conclude that there can be future resolution of the safety issues by the development of further information and that it is acceptable for the reactor internals design to progress generally in its present form.

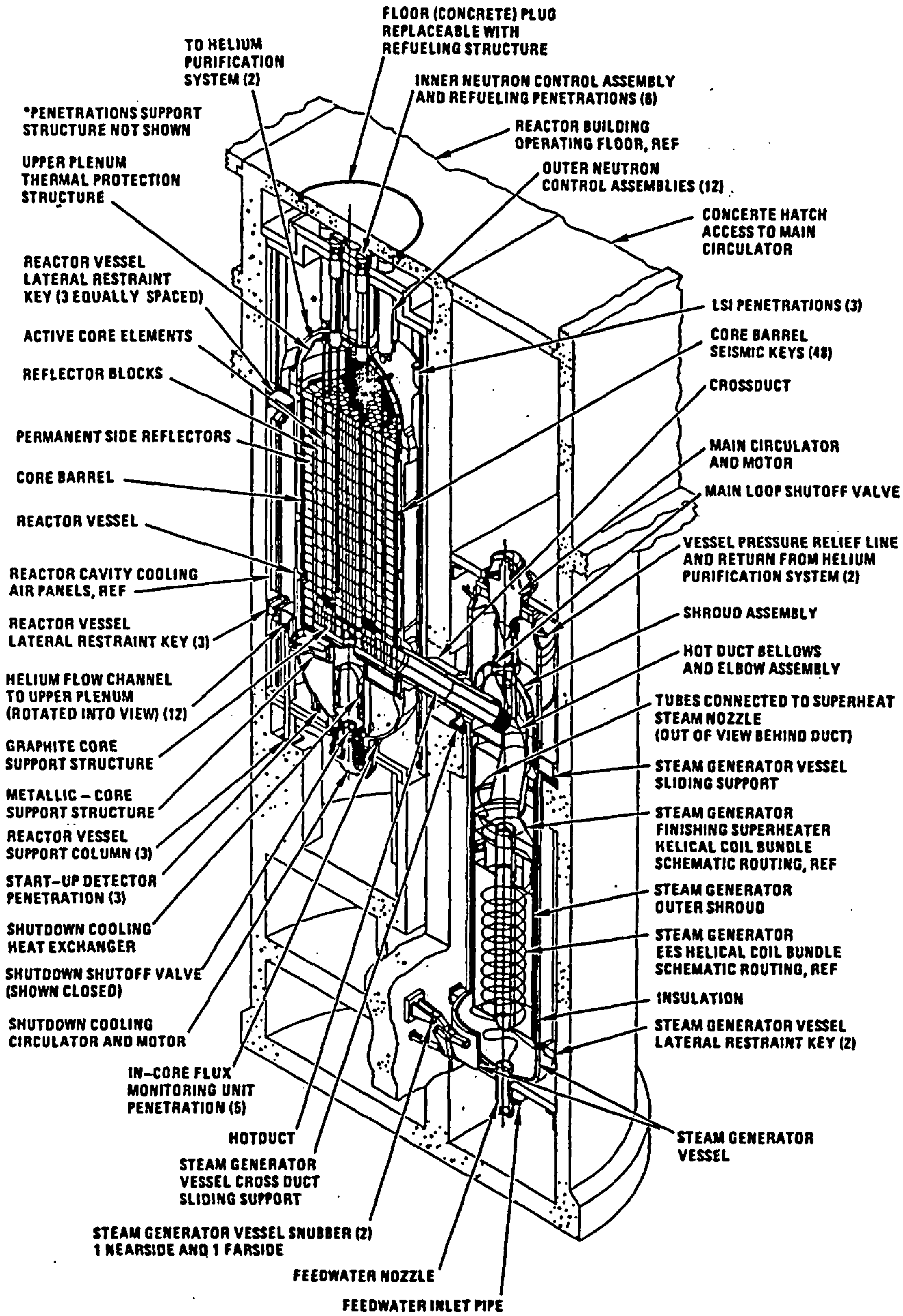

Figure 4.1 Principal features of nuclear steam supply system
Source: DOE, 1987-1

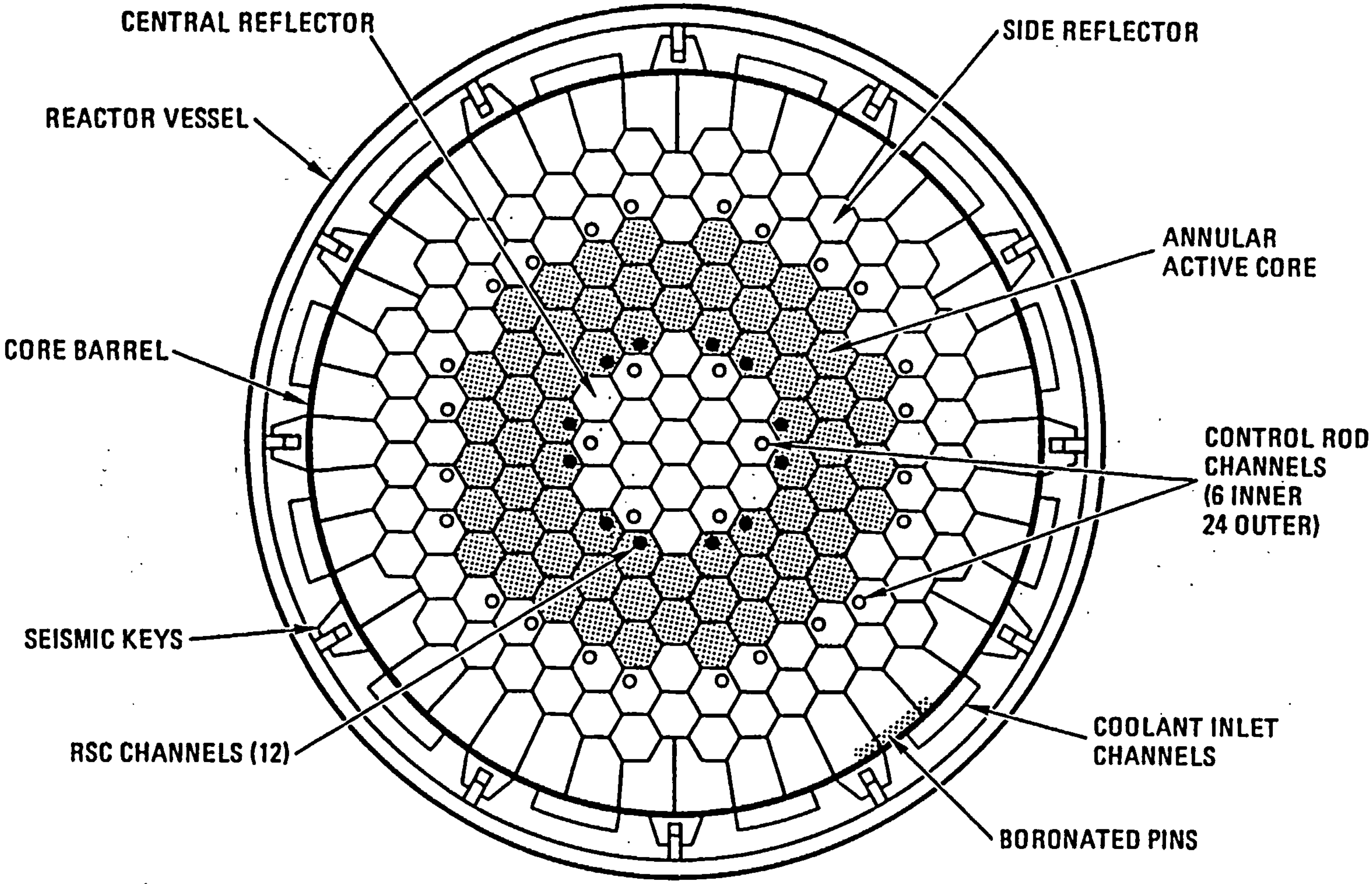

Figure 4.2 Reactor plan view
Source: DOE, 1986-3

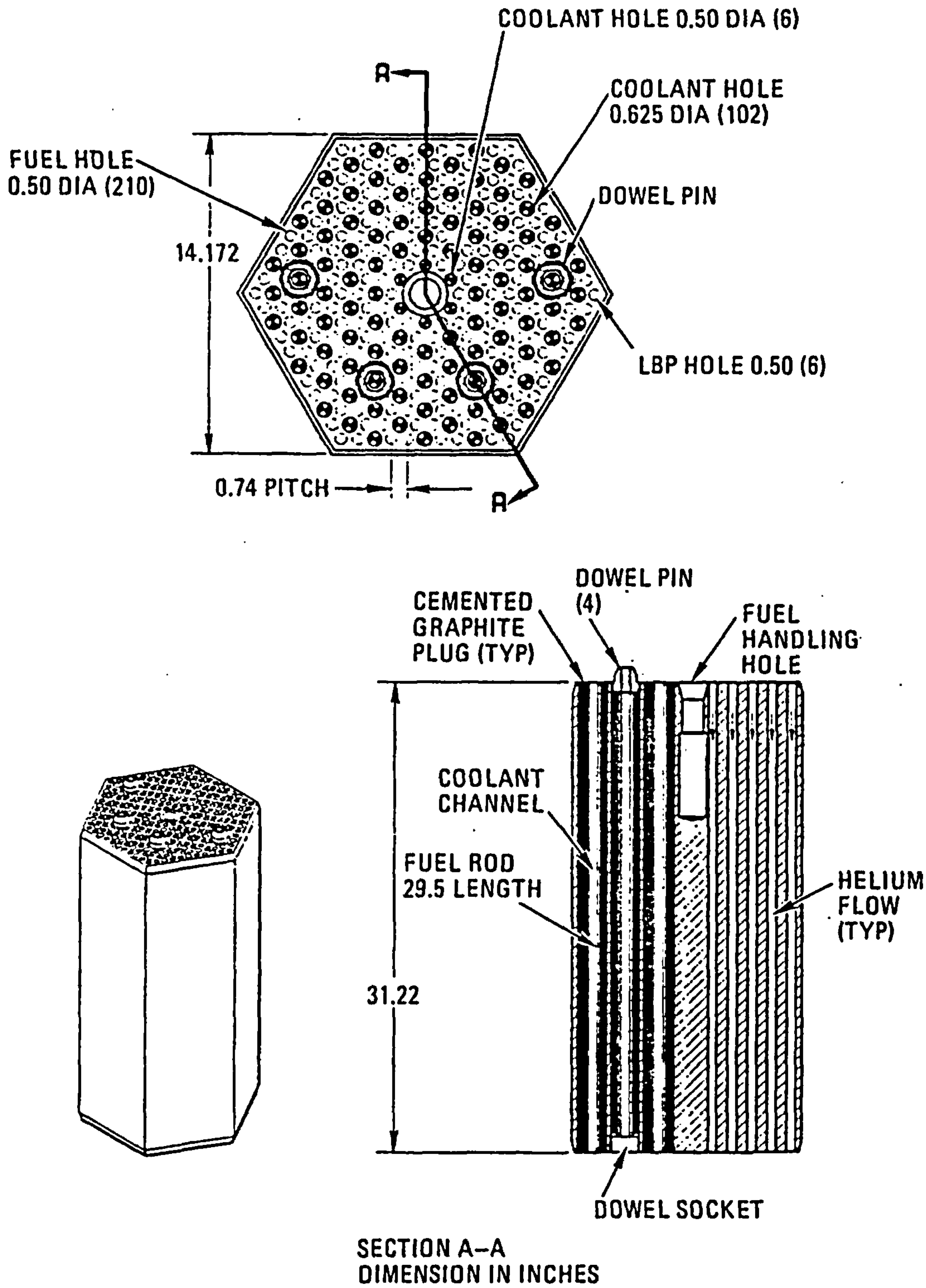

Figure 4.3 Standard fuel element
Source: DOE, 1986-3

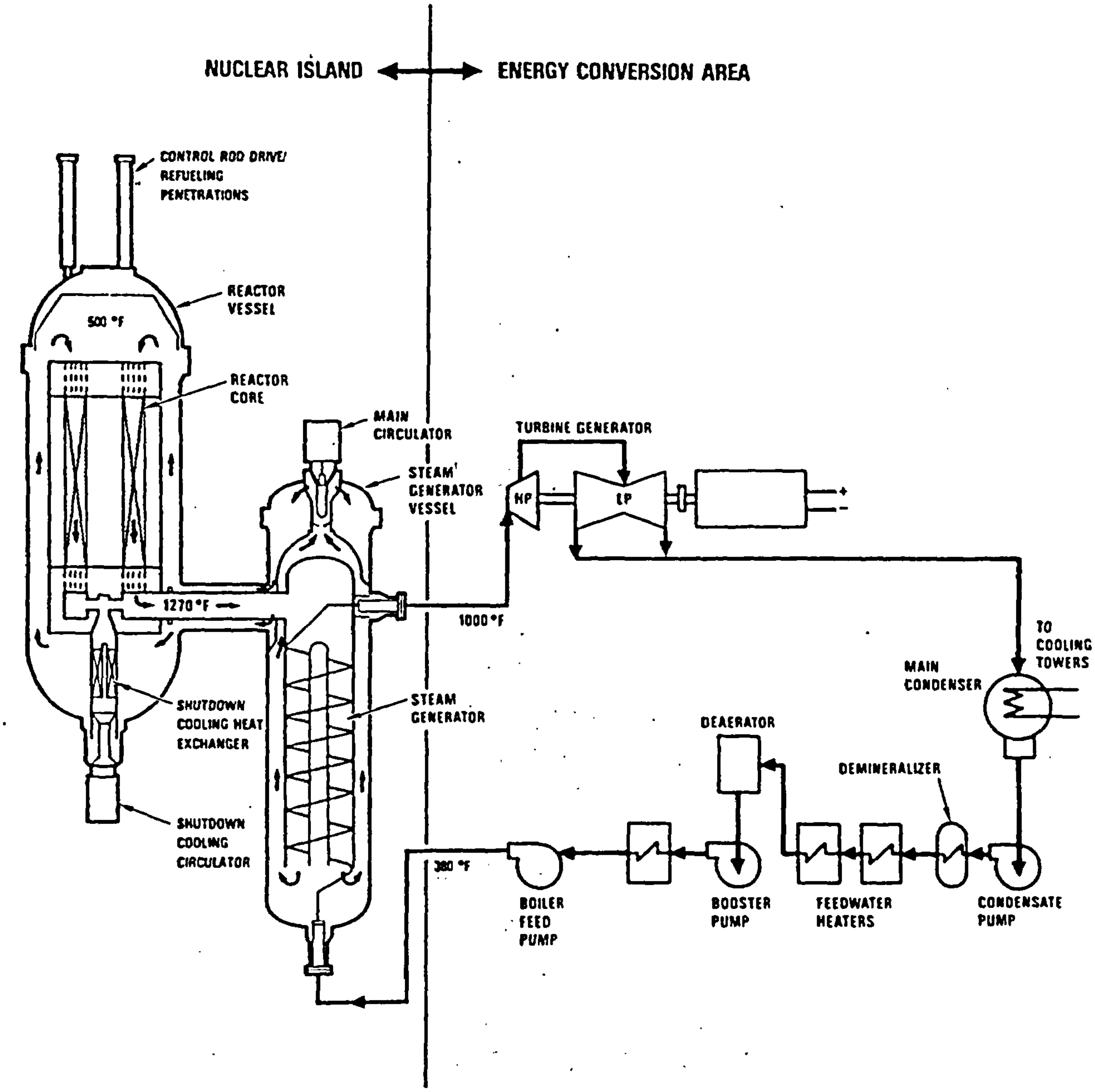

Figure 4.4 Simplified flow diagram
Source: DOE, 1986-3

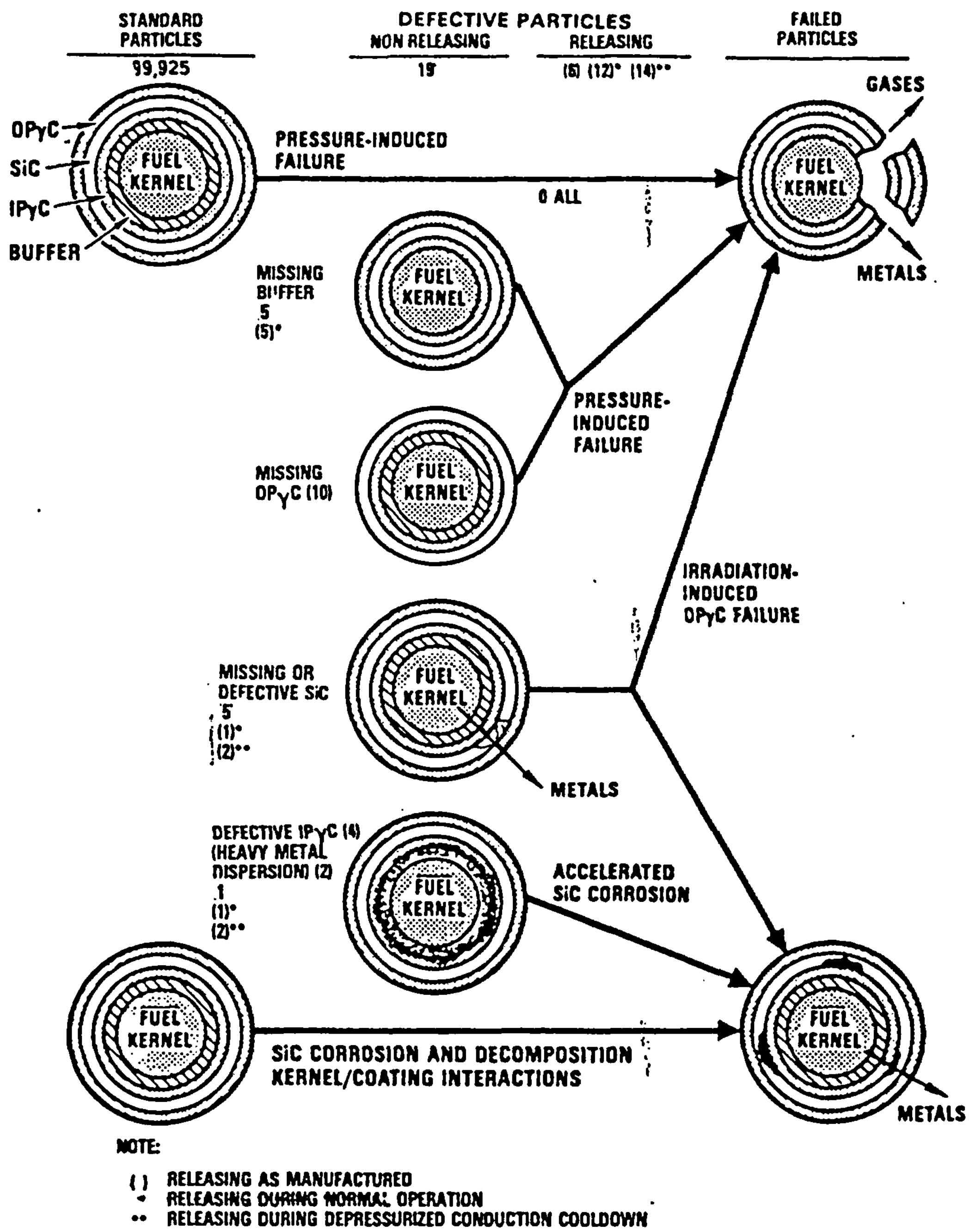

Figure 4.5 Potential TRISO-coated fuel-particle-failure mechanisms
Source: Neylan, 1988-1

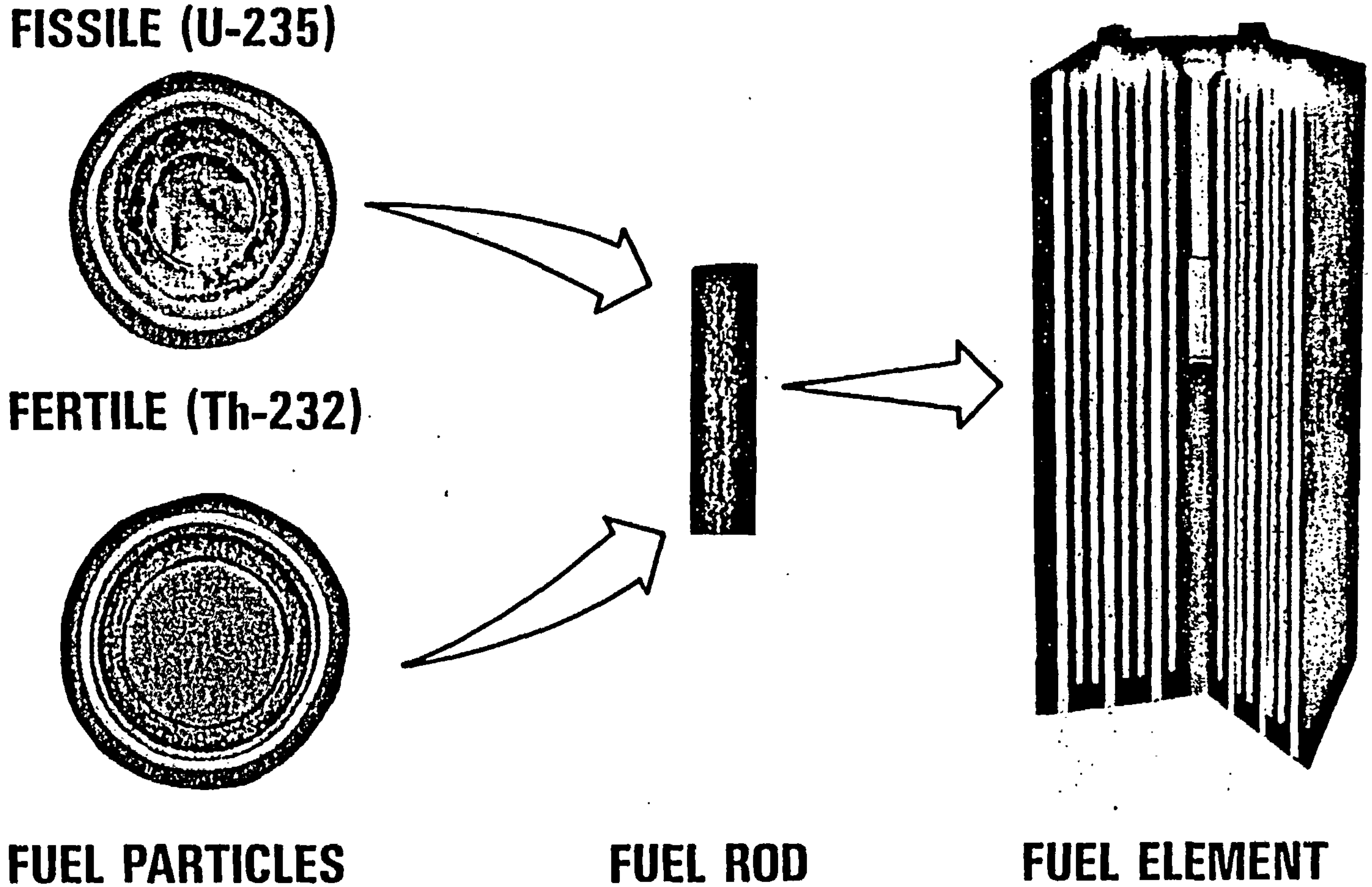

Figure 4.6 MHTGR fuel components
Source: DOE, 1986-3

5 VESSEL AND HEAT REMOVAL SYSTEMS

5.1 Systems Characteristics

Figure 5.1 illustrates the entire MHTGR reactor primary system, including the vessel system (VS) and the two forced-convection heat removal systems. The VS consists of the reactor vessel (RV), the crossduct vessel, and the steam generator vessel (SGV). The two forced-convection systems are the heat transport system (HTS) contained within the SGV and the shutdown cooling system (SCS), a separate system for decay-heat removal contained in the bottom portion of the RV. The HTS consists of the main circulator (MC), the steam generator system (SGS), and the main loop shutoff valve (MLSV). The assessment of the VS is given in Section 5.2, followed by the assessments of the HTS and SCS in Sections 5.3 and 5.4, respectively. The remaining heat removal system, the passive reactor cavity cooling system (RCCS), is assessed in Section 5.5 and is illustrated separately in Figures 5.2 and 5.3. DOE considers the VS and RCCS to be safety-related systems and proposes that the HTS and SCS have non-safety-related functions and that they would not have to fully meet safety-grade quality.

For the safety-related systems, the staff review focused on the ability of these systems to meet their safety objectives, the criteria appropriate for their design and inspection, and the adequacy of their supporting research programs. For the HTS and the SCS, the review focused first on the adequacy of the proposed safety classifications of the individual components of these systems to meet their integrity objectives and to perform their functions in coordination with the safety analyses described in Chapter 15. For those components judged to be either safety related or important to reduce challenges to safety-related equipment, the staff considerations were similar to those for the other safety-related components. In performing the assessments, the staff reviewed Chapter 5 of the Preliminary Safety Information Document (PSID), appropriate portions of the Probabilistic Risk Assessment (PRA), the Reactor Technology Development Plan (RTDP), and DOE's responses to comments and requests for additional information pertaining to this material. The staff was assisted by contractors at Oak Ridge National Laboratory and Brookhaven National Laboratory, who, in addition, performed independent calculations for the performance of the RCCS related to events postulated for the safety analyses.

The principal design criteria applicable to the safety-related components discussed in this section are General Design Criterion (GDC) 1 ("Quality standards and records"), GDC 2 ("Design bases for protection against natural phenomena"), GDC 3 ("Fire protection"), GDC 4 ("Environmental and dynamic effects design bases"), GDC 10 ("Reactor design"), GDC 11 ("Reactor inherent protection"), GDC 12 ("Suppression of reactor power oscillations"), GDC 13 ("Instrumentation and control"), GDC 14 ("Reactor coolant pressure boundary"), GDC 15 ("Reactor coolant system design"), GDC 30 ("Quality of reactor coolant pressure boundary"), GDC 31 ("Fracture prevention of reactor coolant pressure boundary"), and GDC 32 ("Inspection of reactor coolant pressure boundary"). For the two systems that have decay-heat-removal safety objectives, the SCS and the RCCS, the staff believes that together they should meet the intent of GDC 34 ("Residual heat removal"), GDC 35 ("Emergency core cooling"), GDC 36 ("Inspection of emergency core cooling system"), and GDC 37 ("Testing of emergency core cooling system"). In response to Comment G.3-1, DOE made a commitment to meet the intent of these criteria for the MHTGR, although the SCS is classified as non-safety related. DOE also made a commitment that the MHTGR vessel and the safety-related portions of the heat removal systems will meet the appropriate portions of 10 CFR 50.55a, "Codes and Standards," even though this regulation pertains specifically only to light-water reactors (LWRs). DOE also proposed certain general criteria it designated "10 CFR 100 Design Criteria" for the safety-related systems, which the staff found insufficiently detailed and comprehensive.

Lower-level criteria for the design and review of the MHTGR primary system and the RCCS do not exist at present in a form approaching that available for LWR primary systems. No portion of the primary cooling system and the RCCS, including portions of the vessel system, finds direct precedence in LWR technology. Design practices for heat exchangers, gas circulators, ducts, and valves have their earliest precedents in the gas-cooled reactors developed abroad. In the United States, the Peach Bottom 1 and the Fort St. Vrain reactors provided generally favorable experience pertaining to the selection of materials and the functioning of components in a high-temperature helium environment, but that experience did not lead to the development of formalized criteria or industry standards applicable to the MHTGR. The situation is similar to the criteria available for thermal and fluid-flow design of the reactor in that the present approach is judged to be acceptable for the present review stage, but comprehensive and approved criteria equivalent in detail to those contained in the Standard Review Plan (SRP) (NUREG-0800) for LWRs will be needed for a construction-permit review. As will be seen in each subsequent section of this chapter, certain LWR criteria, such as regulatory guides, are helpful and important in guiding the MHTGR conceptual design, but significant gaps remain, particularly those relating to safety issues.

A major concern pertaining to the primary system as a whole is based on the staff's concern that more information is needed on primary-system metals with regard to potentials for detrimental chemical attacks from low concentrations of helium contaminants during long-term operations and from higher concentrations during short-term abnormal operations. Such contaminant or ingress chemicals could include oxidants, hydrogen, hydroxides, nitrogen, chlorides, and carbon dust. This concern includes the aging concern and is similar to the chemical-attack and materials-selection concerns expressed for the fuel and the reactor internals in Sections 4.2.5.I and 4.5.5.G, respectively. The concern includes consideration of potential synergistic effects of trace chemicals and the need to determine quantities of impurities that are either necessary or must be avoided. The staff desires that this concern be addressed by DOE in the forthcoming revision to the RTDP. The staff expects that DOE would first address the status of knowledge in terms of the metals selected for the MHTGR primary system and the contaminants expected or possible and then propose a research program to address the remaining concerns. The knowledge that is available or that could be developed from the experiences at Fort St. Vrain and other operating HTGRs should be described.

5.2 Vessel System and Subsystems

5.2.1 Design Description and Safety Objectives

As illustrated in Figure 5.1, the vessel system (VS) consists of a reactor vessel (RV) and a steam generator vessel (SGV) connected by a crossduct vessel. The subsystems are the pressure relief system (located in the SGV but not illustrated) and the vessel support system. All these systems, with the exception of the thermal insulation surrounding the crossduct and steam generator vessels, are classified by DOE as safety related.

The same type of steel, SA533B, as that used for LWRs is used for the VS. The dimensions of the reactor vessel are 22 meters (72 feet) in overall height, an outside diameter of 6.8 meters (22.4 feet), and a wall thickness of 133 millimeters (5.25 inches), which are approximately the dimensions of a large boiling-water-reactor vessel. For the reactor vessel to function in a sustained conduction-cooldown event (that is, a loss-of-forced-cooling [LOFC] event), such as that described in Section 5.5, it 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 Committee, Section III, Division 1, has been made to extend the allowable temperature to 800°F pressurized and 1000°F depressurized. Application is also being made to the Code Committee to confirm DOE's proposed approach to design of the crossduct as a vessel meeting ASME Code, Section III.

The pressure relief system is stated to meet the ASME Code and is similar to the Fort St. Vrain system except that in each of the two identical relief trains, a burst disk is downstream rather than upstream of the safety-relief valve. A block valve precedes the relief valve, and an interlock system as discussed in Section 7.2.5.D prohibits the closing of both block valves at any given time. The pressure relief system is located in the upper region of the steam generator vessel, downstream from the main loop shutoff valve (MLSV).

The overall safety objective for the reactor vessel as stated by DOE is to meet a level of integrity comparable to that of LWR reactor vessels. Some of the differences between reactor-vessel duty for MHTGRs and LWRs are given in Table 5.1.

5.2.2 Scope of Review

The staff review focused on the reactor vessel with respect to its integrity in relation to LWR vessels and the consequences of and conditions for ASME service level C and D events. The reviews of the vessel support system, the designs of the crossduct vessel and the steam generator vessel, and the conformance of the pressure relief system to appropriate ASME Code requirements have been deferred to a later review stage.

5.2.3 Review, Design, and Inspection Criteria

The general design criteria applicable to the vessel system (VS) are identified in Section 5.1. DOE has proposed that the VS meet ASME Code, Sections III and XI; this effectively satisfies the general design criteria and 10 CFR 50.55a requirements. Because the DOE-requested Code extensions have not yet been approved by ASME and NRC and because of the unresolved status of many of the safety issues discussed in Section 5.2.5, the need for additional review, design, and inspection criteria for the VS will have to be considered at a later review stage.

5.2.4 Research and Development

For technology development need TDN 8-1, "Nil Ductibility Transition Temperature Shift for Reactor Vessel Material Irradiated at Low Temperatures," the RTDP outlines a program to assess damage to the reactor vessel from a neutron fluence of a higher energy spectrum and at a lower vessel temperature than experienced by LWRs. The reactor physics aspects of TDN 8-1 are discussed in Sections 4.3.4 and 4.3.5.E. Detailed planning and execution of TDN 8-1, which includes irradiation in a test reactor, are expected to take into account the response to Comment 5-15, where DOE stated that "experiments to tailor the spectrum appear feasible."

For TDN 8-2, "Properties of SA533B at Elevated Temperatures," the RTDP describes a research program to support the ASME Code extension. Since this program is now being reviewed by ASME, the staff will defer to a later review stage its assessment of TDN 8-2 and take into consideration the findings of ASME.

5.2.5 Safety Issues

A. Probability of a Gross Vessel Failure

In response to Comment 5-45.A, DOE stated that the frequency of a gross vessel failure for the MHTGR has been assessed to be less than 10^{-8} per plant-year for a plant consisting of four modules. This assessment was derived on the basis that the helium coolant would produce less severe thermal transients than water and would not introduce corrosive or erosive attacks as does water. Further, DOE believes that vessel reliability will not be reduced by uncertainties associated with (1) neutron embrittlement, (2) creep at elevated temperatures, and (3) the unique geometry of the vessel system. These uncertainties are to be resolved by tests, research programs, and a Code Case inquiry to ASME (see Section 5.2.5.D). In addition, DOE stated that vessel-failure probability would be independent of the type of failure mode (possibly pneumatic for the MHTGR versus hydrostatic for LWRs) and that unstable crack growth would not cause failure because leakage through the crack would exceed the helium makeup system's ability to maintain pressure, and the reactor would shut down on signal of low pressure.

For comparison purposes, the staff, in conjunction with the Advisory Committee on Reactor Safeguards, established (NUREG-75/014) for LWR nuclear-grade reactor vessels a failure probability estimate of 10^{-7} per reactor-year for failures that could cause core melting. The staff agrees with DOE that the helium coolant provides advantages with respect to thermal transients and corrosion and that there is no potential for pressurized thermal shock or thermal shock by a water-ingress mechanism or for waterhammer effects. At this stage of review, the staff cannot confirm DOE's vessel failure probability estimates. This must be done at a later review stage and must take into account the deliberations and findings from the Code Case inquiry and the planned test and research programs. The staff will also consider that in comparison with LWRs, the MHTGR would require about ten times the number of reactor vessels to achieve the same amount of electric power generation.

B. Neutron Irradiation

The MHTGR reactor vessel will be irradiated at a lower temperature (about 400°F) with a higher energy neutron spectrum than are LWRs. In response to Comment 5-15, DOE stated that the predicted shift in the nil-ductility transition temperatures (NDTTs) of the MHTGR reactor vessel to be caused by neutron irradiation is less than for current-generation pressurized LWR steel vessels because of the expected lower fluence, even though the irradiation temperature is lower and the neutron spectrum contains a greater fraction of fast neutrons. Although the objective of the planned research program is to confirm this prediction, the issue of neutron damage with respect to the vessel's long-term integrity and its effects on the probability of failure will remain open until completion of both the experimental program described in the RDTP and verification by the surveillance to be performed in accordance with ASME Code, Section XI. This concern has been exacerbated by the reported neutron damage to the steel reactor vessel of a test reactor at Oak Ridge National Laboratory, which operates at low fluence but at low temperature. Neutron-streaming effects would need to be considered in analysis of the reactor vessel (RV). The physics aspects of RV fluence are discussed in Section 4.3.5.E.

C. Service Levels for Conduction-Cooldown Events

As described in Section 5.5.1, reactor decay heat will be transmitted, mostly by radiation, from the outer surface of the reactor vessel to the air-cooled panels of the reactor cavity cooling system (RCCS) for various cases when all forced-convection cooling is lost, a set of postulated events that have been termed "conduction-cooldown events" by DOE. Usually, the reactor would remain pressurized in these events, but events with the reactor becoming depressurized must also be considered. In the case with the RCCS unavailable, as discussed in Section 5.5.5.K, the vessel system must be manually depressurized, as discussed in Sections 5.5.5.K and 9.2.5.D.

In response to Comment 5-45.D, DOE clarified the reactor-vessel duty for these conduction-cooldown events in terms of service level requirements, as defined in the ASME Code, and the expected lifetime frequency of their occurrence per module. The ASME service level definitions range from level A for normal operation to level D for conditions that might require removal and replacement of the vessel, with level B traditionally corresponding to anticipated operational occurrences and level C to design-basis accidents. For total, immediate, and sustained loss of forced cooling, the expected service levels and frequencies are listed in Table 5.1 with Item 10 corresponding to the pressurized-conduction-cooldown case and Item 11 to the depressurized case. The frequencies are 2.5×10^{-2} and 3×10^{-3}, respectively. DOE states that for both cases the reactor vessel would experience level C design conditions. Since the vessel would experience temperatures to 900°F and possibly greater for the depressurized case, however, it is possible that later study may determine that level D conditions would actually be experienced.

The LWR frequency of less than 10^{-3} given in Table 5.1 is based on customary regulatory staff practice. Accordingly, DOE is proposing that for MHTGR events with frequencies corresponding to anticipated operational occurrences (AOOs) and design-basis accidents (DBAs) an ASME service level condition normally associated only with DBAs and events of lower frequency for LWRs be used. This proposal is unacceptable to the staff. It is the staff position that in order to ensure that the margins of integrity of the MHTGR reactor vessel are at a level comparable to that for LWRs, some combination of plant systems design and additional safety analyses must be pursued during the next stage of the MHTGR design to lower the expected frequency of level C occurrences to a value consistent with that for LWRs and to ensure that if level D conditions could actually occur, such an occurrence would be extremely rare and consistent with the frequency for LWRs.

The expected frequency of level C and, potentially, level D occurrences is unacceptable for two other reasons. First, because level C and D occurrences require extensive inspection and repair (or possible replacement for level D), considerable occupational radiation exposure would result. The staff judges that comparison of such occupational doses with total doses that could be estimated from the many fewer level C and D events expected for an equivalent amount of electric power generation by an LWR, the MHTGR would not be consistent with the ALARA (as low as is reasonably achievable) principle as applied to occupational radiation exposure. Second, and equally important, level C and D occurrences could, in the words of the Safety Goal Policy Statement (51 FR 28044) with respect to accidents apart from their health and safety consequences, "...erode public confidence in the safety of nuclear power...."

Also in response to Comment 5-45.D, DOE gave two examples of pressurized-conduction-cooldown events in which level B conditions would be expected. For a level B event, the component could withstand loadings without damage that would require repair, and such occurrences are more generally acceptable to the staff. In the examples given, DOE estimated that level B would not be exceeded if (1) forced-convection decay-heat removal existed for at least 24 hours before reliance on the RCCS or (2) forced convection could be restored within 24 hours after initial reliance on the RCCS. At a later review stage, the staff will review these examples and other means DOE may propose to decrease the frequency of events that could lead to level C and, potentially, level D conditions.

D. ASME Approval for Elevated-Temperature Service

DOE is seeking ASME Code Committee approval of a special code case that addresses elevated temperatures during level C and D events for SA-533B Class 1 steel plate, SA-508 Class 3 steel forgings, and related weldments at temperatures ranging between 700 and 1000°F for exposure times not to exceed 1000 hours. An important determination of the Code Committee will be its assessment of whether depressurized conduction cooldown will result in a level C occurrence, as expected by DOE, or actually fall in the level D domain. DOE states that this inquiry completely covers the design for the MHTGR vessel system, including the unique aspects of the crossduct vessel, and has as its objective the achievement of the same level of integrity as currently exists for LWR reactor vessels. When deliberations and the findings of ASME are made available, the staff will evaluate them in the context of a staff safety analysis for the vessel system, which will be performed at a later stage of review.

The staff is considering assigning an observer or a voting member to the involved ASME subgroups or working groups to follow or participate in the code case development. In this way the staff would have the benefit of the Committee's deliberations, as well as its findings, in its subsequent decision-making.

E. Pneumatic Failure Mode

The pneumatically pressurized MHTGR vessel system could potentially fail by catastrophic rupture rather than by a stable tearing mode characteristic of hydrostatically pressurized, gross vessel failures. DOE contends that the toughness of the material and the size of a through-wall crack are the governing conditions rather than the pressurizing medium. The staff recognizes that the temperature-time toughness and the crack size, along with the state of stress, should determine whether or not an unstable fracture will initiate. Ample evidence can be cited, however, to show that in steel vessels the extent of fracture following initiation varies with the pressurizing medium (for example, the lengthy tearing failures of gas pipelines.) Accordingly, the staff will keep open the concern of catastrophic vessel failure and will review this issue more thoroughly at a later review stage. The staff agrees with DOE that catastrophic failure of the crossduct vessel would not result in a graphite fire, as discussed in response to Comment 5-45.B and in Section 15.2.6.2, but at this stage has not evaluated the effects of such rapid depressurization on the reactor structural internals or the core, or the consequences of a catastrophic reactor-vessel failure.

F. Leakage Detection

The vessel system (VS) forms the major portion of the reactor coolant pressure boundary (RCPB), with the principal exceptions being the steam generator and the heat exchanger for the shutdown cooling system. DOE's position with respect to leakage detection is that the MHTGR need not conform to the provisions of the Standard Review Plan (NUREG-0800), Section 5.2.5, "Reactor Coolant Pressure Boundary Leakage Detection," because the MHTGR RCPB essentially meets the integrity standards of reactor vessels and is not similar to the connected piping or components of an LWR RCPB. Further, the leak-before-break philosophy is stated not to be needed or applicable to the MHTGR. DOE also states that reactor shutdown would occur for any leak greater than 0.05 square inch because sufficient makeup helium could not be supplied to maintain vessel pressure. A leak of this size would correspond to a circumferential crack about 7 inches long, a size DOE states has been demonstrated by fracture mechanics to be stable against further growth for all postulated loads, including the safe-shutdown earthquake. Cracks this size would not be expected to exist in the VS because of inservice-inspection requirements.

The staff has not developed its position on whether leakage measurements and source identification would make a significant contribution to MHTGR vessel-system integrity. Such a position will be developed at a later review stage.

G. Thermal Stress, Strains, and Creep-Fatigue Interaction

Experience has shown that thermal stresses and strains have turned out, in many cases, to be much higher than predicted by generally accepted design methods. In response to this concern, DOE pointed out that reactor-vessel construction will be in accordance with the same quality assurance program found acceptable to the NRC for LWR reactor vessels and noted that should thermal stresses occur at levels greater than predicted, experience has shown that they cause minimal distortion and would not contribute to vessel failure. At a later review stage, DOE is expected to present a quantitative identification of anticipated thermal stresses and demonstrate how they have been accommodated in the reactor-vessel design, including justification for the rules to be followed for service level C and D conditions.

In response to the staff's concern about creep-fatigue interaction, DOE stated that time durations at temperatures greater than 700°F would not be sufficient to accumulate a significant amount of creep strain. Since the Code Case inquiry will address the stress allowable values for the elevated-temperature service conditions needed to ensure that no significant creep strains will be produced, the staff will defer its evaluation of DOE's positions that it is not necessary to address creep-fatigue interaction and that low-temperature design rules will be appropriate.

5.2.6 Conclusions

Although the use of steel pressure vessels for gas-cooled reactors has been established, the MHTGR's use of the reactor vessel at elevated temperatures for passive decay-heat removal is novel for the amount proposed and presents a unique regulatory issue. The staff has raised important safety concerns with respect to vessel-system integrity, neutron fluence, duty requirements for the reactor vessel, the nature of failure modes, and other issues. It has been unable to resolve the safety issues developed without additional analytical, design, and experimental information. The deliberations and findings of the ASME committees and working groups pertaining to the Code Case inquiry for limited elevated-temperature service will be an important input to staff decisions at the preliminary standard safety analysis report (PSSAR) stage of review. At present, the staff has found unacceptable the expected frequency of level C occurrences and takes the position that DOE must pursue some combination of systems design and safety analysis to ensure frequencies of level C and, potentially, level D occurrences comparable to those for LWRs. The staff appreciates that the passive decay-heat-removal function of the reactor vessel is central to the MHTGR concept and will work closely with DOE to resolve the identified safety issues.

5.3 Heat Transport System and Subsystems

5.3.1 Design Description and Safety Objectives

In the MHTGR design the heat transport system (HTS) consists of the steam generator (SG), main circulator subsystem (MCS), and the main loop shutoff value (MLSV). These components are located as shown in Figure 5.1 in the separate steam generator vessel (SGV). In normal operation the HTS serves to transfer energy from the reactor primary coolant (helium) to the secondary coolant (water) to convert the incoming feedwater to superheated steam to be sent to the steam turbine in the energy-conversion area (ECA), as illustrated in Figure 4.4. 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 can also operate without steaming, as required in some transients.

The normally operating flow path of the primary system consists of hot helium from the core entering the SGV through the hot duct, the inner passage of the crossduct vessel. Flow is 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 is redirected upward through the annulus between the SGV and the SG shell toward the main circulator (MC) inlet ducting. After passing through the MLSV, a flow-activated check valve, it enters the MC and, after flowing downward again, is directed horizontally to the outer annulus of the crossduct vessel and returns to the reactor. The SGV outside walls are thermally insulated to minimize heat losses from the primary coolant. The main function of the MLSV is to prevent damage to the steam generator system by providing limited bypass flow through the HTS when the steam generator is not in operation. A helium-jet mechanism is provided to cause MLSV closure by operator action if necessary.

As shown, the steam generator is a vertically oriented, cross-counterflow, shell-and-tube, once-through heat exchanger. The economizer-evaporator superheater (EES) section is 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 are 22.2 millimeters in outside diameter with 3.3-millimeter wall thickness, and thus are substantially heavier than LWR tubes but slightly lighter than those at Fort St. Vrain. The EES tube section is of type 2-1/4 Cr-1 Mo steel, while the FS section is Alloy 800H. The bimetallic welds between EES and FS sections are located in a quiescent region. Feedwater enters the SGV at the bottom and is directed to a tube sheet from where it flows upward through the helical tubing and exits as superheated steam through an upper, side-mounted tube sheet.

The main circulator subsystem (MCS) is located at the top of the SGV as shown in Figure 5.1. It includes 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 fully floating on a set of active magnetic bearings, with a backup system of conventional antifriction catcher bearings. Safety-grade trip logic and actuators are provided to prevent operation of the MC when the rest of the heat transport system (HTS) is shut down.

The electric-motor cavity is kept at a pressure slightly above the HTS pressure by a continuous supply of purified helium. The heat exchanger that provides cooling of the motor winding and the magnetic bearings is also located in this cavity and is water cooled. The water pressure is kept below the primary-system pressure to minimize the potential for water ingress from this source.

The operational safety objectives of the HTS are to prevent or minimize (1) long-term degradation of or damage to fuel and other components by controlling temperatures within acceptable limits; (2) low-level moisture concentrations that could result in fuel hydrolysis, as discussed in Section 4.3, and the oxidation of graphite structural components, as described in Section 4.5; (3) challenges to the pressure relief system; and (4) the potential for a large steam-ingress event, as identified in Table 3.7. The HTS, together with the shutdown cooling system (SCS) described in Section 5.4, must also function to minimize the frequency of extended reactor-vessel temperature elevations in accordance with the staff position stated in Section 5.2.5.C.

The safety objectives with regard to integrity are to prevent challenges to the reactor coolant pressure boundary such as could be caused by circulator vibrations, compressor rotor and deblading failures, steam generator tubing vibrations, differential thermal expansions, thermal stresses, and deformations.

5.3.2 Scope of Review

The review focused mainly on licensing criteria and the quality classifications appropriate for the individual components, with emphasis on the steam generator. Also, the staff requested and reviewed additional information concerning circulator failure modes and the circulator development program.

5.3.3 Review, Design, and Inspection Criteria

DOE does not consider any portion of the HTS to be safety related and proposed no "10 CFR 100 Design Criteria." DOE stated however, that the MHTGR will meet the intent of GDC 14 ("Reactor coolant pressure boundary") and GDC 15 ("Reactor coolant system design"). Consequently, DOE plans to design the steam generator system to meet ASME Code, Section III, Division 1, and also will have provisions for inspections that could meet ASME Code, Section XI, Division 1. It is DOE's position that integrity of the steam generator system is not required to ensure adherence to 10 CFR Part 100 dose limits. In response to Comment 5-30, DOE stated that the design of the steam generator provides access for leak testing and plugging of individual tubes in excess of that required by the maintenance standards for existing steam generators, but also stated that the imposition of the ASME Code tube-inspection requirement is not necessary for safety reasons. At a later review stage, the staff will consider the general applicability to the steam generator of GDC 32, "Inspection of reactor coolant pressure boundary"; SRP Section 5.4.2.1, "Steam Generator Materials"; and SRP Section 5.4.2.2, "Steam Generator Tube Inservice Inspection."

With respect to missiles that could be generated by circulator failure, GDC 4, "Environmental and dynamic effects design bases," is applicable. In response to Comment G.3-4, DOE made a commitment in Amendment 7 to meet the intent of the following regulatory guides for both the HTS and the shutdown cooling system, reviewed in Section 5.4:

1.20 Vibration Assessment During Preoperational and Initial Startup Testing (Rev. 2, May 1976)

1.29 Seismic Design Classification (Rev. 3, September 1978)

1.45 Reactor Coolant Pressure Boundary Leakage Detection Systems (Rev. 0, May 1973)

1.49 Power Levels of Nuclear Power Plants (Rev. 1, December 1973)

1.60 Design Response Spectra for Seismic Design of Nuclear Power Plants (Rev. 1, December 1973)

1.61 Damping Values for Seismic Design of Nuclear Power Plants (Rev. 0, October 1973)

1.84 Design and Fabrication Code Acceptability, ASME Section III, Div. 1 (Rev. 24, June 1986)

1.85 Materials Code Acceptability, ASME Section III, Div. 1 (Rev. 24, June 1986)

1.87 Guidance for Construction of Class 1 Components in Seismic Response Analysis (Rev. 1, October 1978)

1.92 Combining Model Responses and Spatial Components in Seismic Response Analysis (Rev. 1, February 1976)

1.130 Design Limits and Loading Conditions for Class 1 Plate-and-Shell-Type Component Supports (Rev. 1, October 1978)

1.133 Loose Part Detection Program for Primary System (Rev. 1, May 1981)

Also, in Amendment 7, DOE made a commitment to assess the applicability of the following regulatory guides as the MHTGR design develops:

1.31 Control of Ferrite Content in Stainless Steel Weld Metal (Rev. 3, April 1978)

1.34 Control of Electroslag Weld Properties (Rev. 0, December 1972)

1.38 QA Requirements for Packaging, Shipping, Receiving, Storage and Handling (Rev. 2, May 1977)

1.43 Control of Stainless Steel Weld Cladding of Low Alloy Steel (Rev. 0, May 1973)

1.44 Control of the Use of Sensitized Stainless Steel (Rev. 0, May 1973)

1.50 Control of Reheat Temperature for Welding Low-Alloy Steel (Rev. 0, May 1973)

1.68 Initial Test Programs for LWR Power Plants (Rev. 2, August 1978)

1.73 Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants (Rev. 0, January 1974)

1.39 Guidance for Residual Heat Removal (Rev. 0, May 1978)

1.147 In-service Inspection Code Acceptability, ASME Section XI, Div. 1 (Rev. 5, August 1986)

1.148 Functional Specifications for Active Valve Assemblies in Systems Important to Safety in Nuclear Power Plants (Rev. 0, March 1981)

The guidance provided by the applicable GDC is satisfactory, and DOE has identified many other applicable and potentially applicable regulatory guides that will be helpful in the design and review, although this list will be reviewed in detail at a later review stage. Additional lower-level criteria may be needed, particularly for gas-cooled-reactor compressors and valves as further information becomes available. Also, SRP Section 3.5.1.2, "Internally Generated Missiles (Inside Containment)," may offer guidance in the development of lower-level criteria.

5.3.4 Research and Development

It is DOE's position, regarding main-circulator failures, that disk-catcher tests are not required, since data developed with the Fort St. Vrain disk containment tests are bounding. Similarly, circulator temperatures are bounded by Fort St. Vrain and Peach Bottom 1 data, and no efforts in this regard are considered in the RTDP. DOE does, however, plan non-safety-related tests in the design verification of the magnetic bearings and catcher bearings, as well as verification of the cooling flow in the motor cavity and prototype testing of circulators at full load in a helium facility. The staff plans to monitor these tests and will assess at a later design stage whether the scope and results are adequate with respect to the integrity of the main circulator subsystem in terms of consequences for safety-related items. DOE has proposed no safety-related research program for the steam generator, although the staff expects that appropriate monitoring of steam-generator performance would be included in the startup test program in accordance with the intent of Regulatory Guide 1.68, "Initial Test Programs for Water-Cooled Reactor Power Plants."

5.3.5 Safety Issues

A. Classification of Components

DOE does not classify any component of the heat transport system (HTS) as safety related, although the steam generator and the cooling water system of the main circulator form portions of the reactor coolant pressure boundary (RCPB). DOE contends that because of the continuous availability of the safety-related reactor cavity cooling system (RCCS) and the likely availability of the non-safety-related shutdown cooling system (SCS), the HTS is not essential for decay-heat removal and for preventing overheating of the fuel and other safety-related components of the reactor system. The staff agrees that to meet the four operational safety objectives (helium temperature control, low moisture levels, few challenges to the pressure relief systems, and reduction of the potential for large steam ingress), it is not necessary to classify as safety related those portions of the HTS that would accomplish these objectives. The objectives can be accomplished by design or by technical specification requirements that would detect failures to meet these objectives before the development of any consequences that would exceed a small fraction of 10 CFR Part 100 guidelines. The staff also believes that the main loop shutoff valve need not be safety related because, while its failure to close would reduce the efficiency of the SCS, the proper functioning of the SCS also is not needed to ensure that 10 CFR Part 100 guidelines are met, as described in Section 5.4. Safety issues pertaining to RCPB integrity and defense-in-depth considerations do, however, require the safety-grade classification for the steam generator and the cooling water system of the main circulator, as discussed in Safety Issues B and C below.

B. Steam Generator

A failed steam generator tube could form a potential pathway for fission-product release to the environs via the secondary coolant system. The need to meet a level of RCPB integrity equivalent to that of an LWR and to maintain a comparable and achievable degree of defense-in-depth is the basis for the staff's judgment that the steam generator system should be classified as safety related. Regardless of its stated classification for the steam generator, DOE has made a commitment to meet the ASME Code for its construction, and its design provides for inservice-inspection capability in full conformance with the Code. The staff agrees with DOE that it should not be necessary to perform individual tube inspections on the basis that the actuation system for steam-generator isolation (see Section 7.2.5.E) will be made safety related to provide a continuous and assured means for tube-leak monitoring. The staff believes that for the present and until sufficient operating experience has been achieved, DOE should plan on full conformance with all other portions of the appropriate ASME codes. It is noted that DOE already provides for safety-grade isolation valves for the steam generator.

C. Main Circulator Cooling Water System

The cooling water system for the main circulator forms a portion of the RCPB. While the staff agrees with DOE that its failure would result in core-damage consequences bounded by the steam-generator failure, the cooling system also forms a potential pathway to the environs. For this reason the staff judges it to be necessary to design, construct, operate, and inspect those portions of the cooling water system that form the RCPB to a safety-grade level of integrity, and to make provisions for the safety-grade isolation of this system from its non-safety-grade portions outside the RCPB.

D. Mechanical Failures of the Main Circulator

DOE contends that mechanical failures of the main circulator, such as blade shedding, will be accommodated by a disk catcher and will not degrade the performance of any safety-related system. The staff accepts this position at the present review stage, but final acceptance will depend on the detailed design review to be performed for the construction permit and on the results of the tests, as stated in Section 5.3.4. At that time, the staff will determine whether circulator testing should be made a portion of the RTDP. Furthermore, if the disk catcher cannot be shown to be capable of containing all missiles, it will be necessary for DOE to demonstrate that internally generated missiles will not damage safety-related equipment.

E. Startup, Shutdown, and Part-Load Operation

At a later review stage, the staff will consider transients of the steam and power conversion systems, including the steam bypass system, with respect to the ability of the HTS to function smoothly during these transients and anticipated operational occurrences. The general features of the startup and shutdown subsystems are reviewed in Section 10.2.

5.3.6 Conclusions

The staff finds that the HTS should be able to meet its operational safety objectives of helium temperature control, low moisture levels, few challenges to the pressure relief systems, and the reduction of the potential for large steam-ingress events without safety-grade equipment. The integrity objective with respect to the RCPB will, however, require safety-classification upgrades for the steam generator and the main circulator cooling water system to ensure prevention of pathways to the environs with the same degree of defense-in-depth as required for LWRs.

5.4 Shutdown Cooling System and Subsystems

5.4.1 Design Description and Safety Objectives

The shutdown cooling system (SCS) is the cooling system utilized during maintenance on the heat transport system (HTS) and is proposed as a non-safety-grade backup to the HTS. 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 main loop shutoff valve, but a minor reverse flow through the HTS components (about 10 percent of SCS flow) will permit gradual cooldown of the components of the steam generator vessel as the reactor components are being cooled. The SCS removes decay heat under both pressurized and depressurized conditions, including refueling. It also, in the staff's view, provides the major means for minimizing the frequency of extended reactor-vessel temperature elevations, in accordance with the staff position stated in Section 5.2.5.C. The SCS consists 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 centerline in the bottom of the reactor vessel, as shown in Figure 5.1. The SCHES is served by the shutdown cooling water subsystem (SCWS), which consists of a single water-cooling loop that serves all modules in the plant. Heat from the SCWS is rejected to the service water system (see Section 10.4).

During normal operation and during reactor cooldown by the HTS, the SCS is 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 caused by HTS operation. For initiation of the SCS cooldown mode, which is automatic on signal of HTS shutdown, the SCWS coolant flow rate is raised from its standby level of 15 percent of design flow rate to 100-percent flow; this causes the SLSV downstream to open. The heat-removal rate is then 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 is powered either by the normal or standby (non-Class 1E) electrical power.

In the SCS cooldown mode, hot helium is drawn downward from the lower plenum through a central passage in the core-support floor into the SCHES, where it continues downward over the helical coolant coils to enter the SCCS. From the SCCS the helium flow is discharged through the SLSV 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 is reversed through the crossduct vessel and the steam generator vessel.

The staff views the primary safety objective of the SCS to be that of minimizing the potential for a long-term cooldown by the RCCS, during which reactor-vessel temperatures would become elevated. The other safety objectives are similar to those of the HTS and are designed to prevent or minimize (1) degradation of or damage to the fuel and other components by controlling temperatures within acceptable limits, (2) moisture that could result in fuel hydrolysis or structural-graphite oxidation, and (3) challenges to the pressure relief system. Safety integrity objectives are generally the same as for the HTS; namely, the maintenance of those portions of the SCS that are part of the reactor coolant pressure boundary (RCPB) and the consequences for the RCPB from circulator vibrations, compressor rotor and deblading failures, and thermal stresses, expansions, and deformations that could degrade safety.

5.4.2 Scope of Review

In a manner similar to that for the review of the HTS, the review focused on the licensing criteria and the quality classifications appropriate for the individual components. While many of the safety issues are much the same as those for the HTS, the SCS differences in the staff's perceived safety importance were emphasized. Differences derived from the location within the reactor vessel and the character of the reverse flow through the steam generator vessel were not evaluated at this review stage.

5.4.3 Review, Design, and Inspection Criteria

As for the HTS, DOE does not consider any portion of the SCS safety related and proposes no "10 CFR 100 Design Criteria." DOE has made a commitment, however, that the SCS will meet the same general design criteria and ASME Code requirements as those for the HTS, as well as the lower-level requirements specified in Section 5.3.3 for the HTS. Because of a significant difference in potential safety importance, the staff is considering that special criteria should be developed pertaining to its integrity and availability, as discussed in Section 5.4.5.B.

5.4.4 Research and Development

Again, as for the HTS, DOE plans no safety-related tests for the SCS, but the testing program identified in Section 5.3.4 will be evaluated by the staff. At a later review stage, when more detailed design information is available, the staff will consider whether the resolution of the safety issues identified in Section 5.4.5 should be included in the RTDP.

5.4.5 Safety Issues

A. <u>Issues Similar to HTS Safety Issues</u>

The shutdown cooling heat exchanger should be classified as safety related and a safety-grade means provided to isolate it from non-safety-grade systems. The staff positions with respect to the circulator cooling water system and circulator mechanical failures for the HTS are also applicable to the shutdown cooling system.

B. <u>Safety Classification and High Reliability</u>

Although the staff believes that it is not necessary to classify the shutdown cooling system (SCS) as safety related, its integrity, availability, and performance capability should be established at a high level to reduce the potential for ASME level C or possibly D occurrences for the reactor vessel, in accordance with the discussion in Section 5.2.5.C, and to reduce the amount of challenges to the reactor cavity cooling system (RCCS), in accordance with the findings of the Probabilistic Risk Assessment (PRA) (see Section 15.3). At a later review stage, the staff suggests that a special safety classification or a safety program for the SCS and its power supply be explored and proposed by DOE that would ensure a higher level of availability and performance than would be expected by the staff if the SCS were designed, constructed, and operated without quality standards for its performance functions. As a suggestion, such a program should include preoperational and startup testing and technical specifications or equivalent administrative controls pertaining to its integrity, inspections, maintenance, and out-of-service time limits.

C. Single Heat Sink for Multiple Modules

The same ultimate heat sink for the SCS and the service water system would be used for all four reactor modules in the reference plant. Unavailability of this heat sink could effectively multiply by four the probability of unavailability of the SCS on a per-reactor basis. This concern should be explored in a future PRA and addressed by design if found to be a significant contributor to risk.

D. Diversion of Flow From the Core

Failure of the main loop shutoff valve (MLSV) to close would divert a substantial portion of the SCS from cooling the core. Based on pressure-drop information given in the PSID and the fact that the SCS heat-removal capacity of 23.7 MW is roughly 10 times that of the RCCS, it would not appear that fuel and vessel temperatures would approach those for RCCS cooldowns. This issue could be more complicated than this estimate suggests, however, and should be reconsidered at a later review stage and possibly taken into account in developing the safety-classification proposal for the SCS and the safety classification of the MLSV itself.

5.4.6 Conclusions

The proper role of the SCS with respect to safety is not clear at this review stage. It largely depends on additional information pertaining to the staff concerns about the frequency of service level C and D occurrences to the reactor vessel, as expressed in Section 5.2.6. The staff has suggested in Section 5.4.5.B that DOE propose means to ensure that the RCCS can perform its functions with high reliability without conforming to full safety-related requirements.

5.5 Reactor Cavity Cooling System

5.5.1 Design Description and Safety Objectives

The reactor cavity cooling system (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 events classified as loss of forced cooling (LOFC); that is, when both the heat transport system and the shutdown cooling circulator subsystem are inoperable. The safety objective of the RCCS is to serve as an ultimate heat sink that must meet design requirements with respect to ensuring the thermal integrity of the fuel, outer control rods, the reactor vessel, vessel internals, the reactor-vessel supports, and the reactor cavity. Its thermal performance and structural integrity must be ensured for all environments accompanying the various event category II and III sequences postulated in the safety analysis. The LOFC events in which the reactor is cooled by the RCCS are termed by DOE as "conduction-cooldown" events to emphasize that the dominant mode of cooling the fuel is heat conduction through the core and reflector graphite to the reactor-vessel surface.

The RCCS is a closed system within the reactor cavity with the cooling panels serving as barriers between the outside air and the reactor-cavity air. The natural-convection cooling scheme for the RCCS is shown in Figure 5.2. No active components or moving parts are used in the design, and the system functions at all times, including during 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 staff did not review instrumentation to be provided for this purpose. Radiation detectors provided in two exhaust ducts that monitor for increases in air activation would be considered indicators of panel leakage.

The major thermal performance requirements stated by DOE for the RCCS are (1) capability for maintaining the maximum fuel temperature below 1600°C (2900°F) during LOFC events; (2) for events other than sustained LOFCs, capability for maintaining the maximum vessel temperature below 370°C (700°F); and (3) for LOFC events, capability for maintaining the vessel temperature below 425°C (800°F) pressurized and 530°C (1000°F) depressurized. The RCCS will also be used to maintain the reactor-vessel supports and the reactor-cavity concrete at acceptable temperatures during both normal operation and conduction-cooldown events. The principal mechanical design requirement, other than seismic or other external-event requirements, is that the panels be capable of withstanding differential pressures up to 10 psi for postulated cavity overpressures that could occur from depressurization events or feedwater or steamline breaks. The performance of the RCCS was calculated at an air inlet temperature of 110°F. DOE calculated that a maximum outlet temperature of 330°F would occur at about 100 hours for pressurized conduction cooldown (the controlling event).

In the design of the panels, thermal insulation is 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. Sensitivity studies have shown, however, that the performance of the RCCS will be very insensitive to the amount of insulation used. 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 flow-path blockages. A special "secondary-chimney" design has been provided to address concerns pertaining to the effects of high winds and regenerative heat transfer from the lower-level inlet and outlet ducts, as depicted in Figure 5.3.

5.5.2 Scope of Review

The staff and its contractors at Brookhaven National Laboratory (BNL) and Oak Ridge National Laboratory (ORNL) reviewed Section 5.5 of the PSID and relevant portions of the Probabilistic Risk Assessment, and the NRC contractors also performed independent analyses relating to the predicted performance of the RCCS under normal and abnormal conditions. The reviews concentrated on the safety issues described in Section 5.5.5, which include the sensitivity studies of uncertainties in the thermal-analysis models and model input data, instrumentation and inservice-inspection concerns, and various potential failure modes. The independent sensitivity studies did not fully confirm the DOE studies; these concerns are summarized in Section 5.5.5. The contractors' results are outlined in Section 15.4 and more fully described in Appendixes A and B. Reliability concerns were also addressed by the staff and an ORNL subcontractor (Minarick, 1988) and are described in Section 15.3. The uniqueness and importance of the RCCS resulted in the staff's heavy concentration of its resources in this area.

5.5.3 Review, Design, and Inspection Criteria

DOE has stated that it will meet the intent of GDC 34 ("Residual heat removal") and GDC 35 ("Emergency core cooling") with the qualification that interconnections, leak detection, and isolation capabilities are not required for the RCCS. DOE has also stated it will meet the intent of GDC 36 ("Inspection of emergency core cooling system"). Furthermore, DOE stated that the design, fabrication, and materials acceptability will be in accordance with ASME Code, Section III, Division 1, and inservice inspection and testing will be in accordance with ASME Code, Section XI, Division 1. Based on a review of the regulatory guides affecting LWR emergency cooling systems, DOE concluded that it was appropriate for the RCCS to meet the intent of Regulatory Guides 1.29, "Seismic Design Classification," and 1.139, "Guidance for Residual Heat Removal." DOE plans to assess the following additional guides for possible future use as the RCCS design evolves:

1.31 Control of Ferrite Content in Stainless Steel Weld Metal (Rev. 3, April 1978)

1.34 Control of Electroslag Weld Properties (Rev. 0, May 1973)

1.44 Control of the Use of Sensitized Stainless Steel (Rev. 0, December 1972)

1.50 Control of Preheat Temperature for Welding of Low-Alloy Steel (Rev. 0, May 1973)

1.68 Initial Test Programs for Water-Cooled Reactor Power Plants (Rev. 2, August 1978)

1.84 Design and Fabrication Code Case Acceptability ASME Section III, Div. 1 (Rev. 24, June 1986)

1.85 Materials Code Case Acceptability ASME Section III, Div. 1 (Rev. 24, June 1986)

1.147 In-Service Inspection Code Case Acceptability - ASME Section XI, Div. 1 (Rev. 5, August 1986)

For normal operation, site release limits from the top-level regulatory criteria require that doses shall be less than those of 10 CFR Part 50, Appendix I, and 40 CFR Part 190, and that effluent concentrations shall be less than the maximum permissible concentrations of 10 CFR Part 20, Appendix B, Table 2.

Additional criteria are expected to be developed during the course of preliminary design and as a consequence of further supporting studies. Those foreseen by the staff as likely at this time are (1) explicit commitment to meet GDC 2, "Design bases for protection against natural phenomena," for the RCCS; (2) the applicable portions of SRP Section 6.3, "Emergency Core Cooling System"; and (3) modifications of the ASME Code or the development of Code cases to accommodate unique features of design or inspection requirements.

5.5.4 Research and Development

The Regulatory Technology Development Plan (RTDP) does not identify any research and development activities for the RCCS design or function; however, DOE has stated that it is in the process of reviewing the need for such research. DOE has made a commitment to further analysis and validation of the RCCS evolving design, including studies with detailed computer models, additional sensitivity studies, and development of testing strategies for performance verification and code validation. If system, component testing, or other data needs are deemed to be required, the appropriate plans will be incorporated into the RTDP.

The staff has identified as potential research needs (1) an integral test, possibly in conjunction with the liquid-metal reactor (LMR) programs, to demonstrate the effectiveness and reliability of the naturally convective flow design, including the chimney design; (2) additional information to support materials data currently used for graphite thermal conductivity (irradiated and annealed); (3) the establishment of the resistance of the RCCS structure to large seismic events and other potential external-event failure modes; and (4) improvement in the understanding of the long-term failure modes of the RCCS to aid the development of a suitable inservice inspection program and to address aging concerns.

Overall confirmation of the analytical models and assumptions pertaining to the RCCS's performance and the chimney and duct design will be required by the prototype-plant safety performance testing discussed in Chapter 14. At a later review stage, DOE is expected to describe these tests and how the RCCS's capabilities will be demonstrated.

DOE's commitments to meet the criteria identified in this section are acceptable for the present design stage. Because of the uniqueness of both the design and function of the RCCS, however, additional criteria, some of which have been noted, are expected to be developed during the course of preliminary design and as a consequence of further supporting studies. Development of these criteria should be given a high priority.

5.5.5 Safety Issues

Although the RCCS appears to be a simple, reliable system, the staff reviewed it in detail because it is the only safety-related heat removal system. The staff concerns are primarily with regard to the RCCS's first-of-a-kind design for which the uncertainties regarding its performance and reliability are either incompletely understood or cannot currently be resolved because of a lack of research, operational data, and experience.

A. <u>Heat-Transport Design</u>

Heat-transport-design concerns are considered in two domains: within the vessel and external to the vessel. For the within-vessel domain, both pressurized and depressurized cases are considered, with fuel and vessel temperature limits driving the design. External to the vessel the effectiveness of the natural-convection air flow inside the cooling panels, the vessel and panel emissivities, and the environmental conditions within the vessel-panel interspace are determining.

<u>Vessel Hot Spots</u>. In the modeling of the pressurized-conduction-cooldown events, DOE assumed that 100 percent of the coolant flow was in the active core region and all the downflow, as well as the upflow, passed though the active core. In the ORNL reference-case model, it was assumed that 10 percent (initially) of the core flow was through the center and side reflectors, with most of the

natural-draft downflow through the cooler side reflectors. This tended to redistribute the heat flow to cause temperature variations within the vessel not recognized by the DOE analysis. DOE plans to include this analytical feature in future analyses. An additional concern is that the design for the vessel-head insulation, as provided by the upper plenum thermal protection structure (UPTPS), described in Section 4.5.1, is sensitive to modeling uncertainties and assumptions and could also lead to vessel hot spots. In response to Comment 5-43, DOE performed a sensitivity analysis and stated that it did not believe experimental verification of the UPTPS performance was needed. The staff does not consider, however, that the question of vessel hot spots can be fully resolved until some acceptable form of data to confirm the analytical models and assumptions becomes available from preoperational and startup testing, as noted in Section 5.5.4.

Invessel Conduction. The thermal conductivity characteristics of the fuel and reflector regions are needed in the calculations of maximum fuel and vessel temperatures for the analysis of depressurized-conduction-cooldown events. The fuel and graphite conductivities provided in the PSID and supplemented in detail by General Atomics (Neylan, 1987) were used as the reference values in independent analyses by both BNL and ORNL. The independent studies confirmed DOE calculations that, using conservative model and parameter assumptions, the maximum fuel temperature would remain below the 1600°C fuel design temperature, but gave concern that the maximum vessel temperature could be higher than the DOE current conservative estimate. Sensitivity studies were done by BNL to show the sensitivity to conductivity variations due to irradiation damage and temperature; the results are reported in Appendix B. Since wide variations in both maximum fuel and maximum vessel temperatures were found, DOE needs to demonstrate at a later design stage, by sensitivity studies or by additional data, that conductivity uncertainties in the safety analyses are sufficiently accommodated by fuel and vessel design margins.

Reactor Vessel and Panel Emissivities. Thermal radiation accounts for a large fraction of the heat transferred from the reactor vessel to the RCCS hot-riser panel, and the overall heat transfer is very sensitive to the effective thermal emissivity. Hence, the characteristic thermal emissivity of surfaces for both the reactor vessel and the RCCS hot-riser panels, which are functions of surface temperature and surface finish, must stay sufficiently high over the expected lifetime of the reactor plant, and in no case drop significantly below values used in the safety analysis. The staff expects that DOE will propose appropriate technical specifications or other administrative controls to address this concern.

Effect of Water Vapor. In response to Comment 15-9, DOE estimated the effects of a reduction in radiant heat transmission from the reactor-vessel surface to the RCCS cooling panels that could be caused by the presence of water vapor in the reactor cavity. For a bounding case of an assumed concentration of one atmosphere partial pressure, BNL found an increase of about 90F° for the reactor-vessel temperature and a negligible increase for the maximum fuel temperature. As stated in Section 5.2.5, elevated vessel temperatures are a major concern for the MHTGR concept that will be addressed at a later review stage. The staff believes that the bounding concentration of water vapor could be approached as a consequence of either a main feedwater- or steam-line rupture, which would be included in event category II, but would be in event category III if combined with a conduction-cooldown event. DOE and the staff's consultants also considered the heat-transmission effects of carbon dioxide and carbon monoxide, both within and exterior to the reactor vessel, and concluded that the concern was bounded by the water-vapor case.

B. Repair and Recovery

The DOE assessment of the time available for RCCS repair following an event in which both the heat transport system and the shutdown cooling system as well as the RCCS are all inoperative concluded that RCCS restart within about 24 hours would preclude more than modest elevation in fuel temperatures and would be sufficient to prevent the vessel temperature from exceeding service level B design limits for the case with the vessel remaining pressurized. Times on the order of days would be available for cases with the vessel depressurized.

Independent calculations of vessel and fuel temperatures for partial and/or temporary total failures of the RCCS were performed at ORNL using the MORECA code. The predictions showed that maximum fuel temperatures were relatively insensitive to temporary heat-sink failures in the initial phase of either pressurized or depressurized long-term conduction-cooldown events. With respect to the vessel temperature, however, in a case in which one of the four quadrant panels was assumed to be severely blocked, the MORECA code predicted serious overheating of the vessel in that quadrant. In another case in which the RCCS was fully inoperative for 24 hours near the start of depressurized conduction cooldown, the maximum vessel temperature significantly exceeded the extended ASME Code design limit. Since these results are contrary to DOE's predictions, resolution of the difference in the calculational models is an important issue for the next review stage. An essential reason for ensuring availability of time for RCCS repair is the need to depressurize the vessel in the remote case when the vessel temperature might exceed 800°F. As discussed in Section 9.2.5.D, this can be accomplished by manual depressurization through the helium purification system.

In the event category III safety analysis, recovery of 25 percent of the flow capacity of the RCCs is assumed. The special aspects of recovery are significant, however, because of local heating concerns and must be considered in the safety analyses.

C. Modeling Conservatisms and Sensitivities to Uncertainties

In response to Comments 5-2 and 5-40, DOE quantitatively, where practical, identified conservatisms and sensitivities to the following parameters: (1) decay-heat rate, (2) decay-heat distribution, (3) air inlet temperature, (4) RCCS panel heat-transfer coefficient, (5) graphite thermal conductivity, (6) surface contact and gap resistances between graphite blocks and the inner surface of the core barrel, (7) effects of convection flows in the core and reflector, (8) emissivities on the core barrel inner and outer surfaces, (9) effects of helium convection and helium ducts on the heat transfer across the core barrel, (10) the emissivity on the inner and outer surfaces of the reactor vessel, (11) effects of convection flow exterior to the reactor vessel, (12) emissivity of the RCCS panel surface, and (13) influence of the upper plenum thermal protection structure.

A similar study was performed by the staff's consultants at ORNL and BNL. Both the DOE and the independent studies concluded that the key sensitivities are the decay-heat rate, emissivities, and graphite thermal conductivity. DOE deferred to the preliminary design stage studies of geometrical and asymmetrical effects and other possible sensitive design parameters, which could include changes in structural configuration as a result of high reactor-cavity temperatures, as discussed in Safety Issue D below.

D. Reactor-Cavity Temperatures

In the staff's discussions with DOE, the staff noted concerns about the effects of certain temperature maximums reached in the reactor cavity during the postulated event category II sequences and the possible ensuing damage to essential instrumentation and structures. DOE replied that the maximum cavity temperature during normal operation (near the top) would be 175°C, and that it would reach 290°C during pressurized cooldowns and 315°C during depressurized cooldowns. Essential instrumentation, such as neutron monitors, will be located in the cold inlet air spaces of the RCCS panels and thus will not be affected by cavity or vessel temperatures. No postaccident monitoring instrumentation is currently identified as being located in the reactor cavity. With respect to concrete structures, DOE plans to use regular building-grade concrete and has not satisfactorily addressed concerns about the consequences of concrete failures at higher temperatures. Final resolution of concerns about elevated temperatures and temperature distributions in the reactor cavity and their effects on concrete performance, vessel support, and critical instrumentation and equipment will be completed at a later design stage.

E. Inservice Inspection Program

The staff requested additional information about the objectives and specific details of the proposed inservice inspection (ISI) program. DOE plans to adhere to ASME Code (visual) inspections to detect cracks, partial blockages, or weld failures. Details of the ISI requirements are to be based on sensitivity analyses of RCCS performance under adverse conditions. The staff noted that some disassembly of the RCCS may be necessary to meet ISI objectives and, in Section 5.5.4, recommended that a development program be undertaken to improve understanding of potential long-term failure modes. As discussed in Safety Issue I below, the RCCS may be subject to hidden failure modes because of its innovative design, and a fully adequate ISI program is needed to compensate for lack of operating experience.

F. Panel Drains

The RCCS design provides drains for accumulated-water removal at the bottom of the panels. In response to the staff's concern that blockage of drains could cause air-flow blockage by accumulated water, DOE noted that the major function of the drains was to reduce the potential for corrosion from collected water rather than to prevent air-flow blockage. DOE stated that it did not consider significant flow blockage to be credible, since any accumulation of water would tend to boil off. In addition, the drains are redundant and passive, and the ISI program will ensure their ability to function.

G. RCCS Instrumentation

The staff requested justification for the non-safety classification of the RCCS instrumentation and the safety classification of readouts in the remote-shutdown area (RSA), together with more information about expected operator actions in response to indicated failures. DOE stated, in response, that because the operator does not need to take any immediate action in response to RCCS trouble, and instrumentation malfunctions would not prevent the RCCS from operating properly, the instrumentation is not classified as safety related. DOE plans further study to determine the seismic and power supply qualification needs of RCCS instrumentation. The staff finds that the issue of RCCS instrumentation safety qualification should be reviewed further, noting that there must be adequate assurance of the availability of the RCCS, and that its performance needs to be closely monitored. A staff concern arising from independent analyses was that partial blockage of air-coolant flow that led to failures or degraded performance of part of a cooling panel might not be detected by overall RCCS performance measures and, in a long-term heatup accident, might lead to local reactor vessel temperatures exceeding design limits. In this case, local vessel and panel temperature monitoring would be required. Vessel-temperature monitoring is also required to determine the service level experienced as discussed in Section 5.2.5.C

H. Conformance With 10 CFR Part 100, Appendix I, and 40 CFR Part 190

To ensure that effluent concentrations will be less than maximum permissible concentrations, uncertainties with respect to the RCCS effluent radioactivity level should be addressed at a later review stage.

I. Failure Modes

The staff questioned the quality of the analysis of the RCCS failure modes and effects study performed by DOE. The study considered inlet/outlet blockages, duct breaks, a main-steamline break, icing, wind, tornadoes, windblown debris, and insect swarms. DOE stated that it considered this assessment preliminary and planned to do further and more detailed analyses during the preliminary and final design phases. The staff concurs that more study of RCCS reliability and failure modes is required as the design matures. As an example of the potential for hidden failure modes, the staff postulated a gross circumferential failure near the top of the baffle between the inlet and outlet air-panel flows, a case that could lead to a loss of panel cooling capability by short circuiting the air flow. This can be addressed by inservice inspection requirements or corrected by design changes, but it illustrates that innovative concepts may have innovative (hidden) failure modes.

J. Duct and Chimney Design

The staff believes that the duct and chimney design should provide protection against possible common-mode failure and sabotage. For this reason the staff believes that the dual RCCS chimney stacks should be physically separated and not arranged in a single row as now designed.

In addition the staff believes that design changes may be needed to protect against thermal stratification. Thermal stratification that might cause flow stagnation has not yet been considered in the RCCS review. As long as there is no horizontal segment of the air side that is heated, the air, driven by buoyancy forces in the vertical direction, should circulate properly throughout the RCCS. From the conceptual design drawings available, heating of the air appears to exist only in the vertical hot-riser section of the RCCS panels. During the completion of the design, however, the RCCS should be analyzed for horizontal heating paths, either inside the reactor cavity or inside the reactor building, where thermal stratification might cause flow stagnation.

K. Total Failure

In Appendix G to the PRA, DOE reported an analysis that showed that the safety objectives of the RCCS could be achieved by radiation and convection to the cavity surfaces, provided the vessel was depressurized as the vessel temperature approached the extended ASME Code value of 1000°F. Independent analyses by the staff's consultants showed this heat-removal mode to be feasible but raised concerns related to vessel and cavity integrity and the adequacy of the means to depressurize the vessel. This issue is discussed further in Section 5.2.5, in Section 6.2.5.C in regard to the reactor cavity, and in Sections 8.1.5.A and 9.2.5.D in regard to emergency depressurization. A feature of concern for the RCCS panel design for this event sequence is its use as an insulator and radiation shield. As stated in Section 5.5.1, the RCCS cooling function is not sensitive to the amount of insulation used and thus it is not evident why insulation is inserted between the downcomer and riser.

5.5.6 Conclusions

Many safety issues are described above that, while believed to be resolvable, may have a major impact on the RCCS design and function. DOE should be prepared at an early stage of the PSSAR review to complete discussion of these topics. The staff believes that well-defined programs in all these areas are necessary, unless suitable alternatives can be identified and approved by the staff. Furthermore, it appears that the establishment of a suitable program within the Regulatory Technical Development Plan will be necessary for the satisfactory resolution of some of these issues.

Of major importance for early resolution are differences in analytical modeling between DOE and the staff consultants at ORNL with regard to the heat-transport design. These differences, with ORNL predicting higher reactor-vessel temperatures at earlier times in the conduction-cooldown events, represent a fundamental open issue that will affect later decisions on both modeling and development requirements. Resolution of the concern will require detailed review of both the ORNL and DOE models and assumptions.

In summary, both the uniqueness and the importance of the RCCS have resulted in the staff's heavy concentration of its resources in this review area. Although there are no questions regarding the feasibility of the RCCS function and the ability of the core to conduct decay heat to the reactor-vessel surface without significant fuel failure, it is necessary to emphasize that proper analytical verification, design, construction, and operation of the RCCS must be achieved in order to meet high-reliability goals.

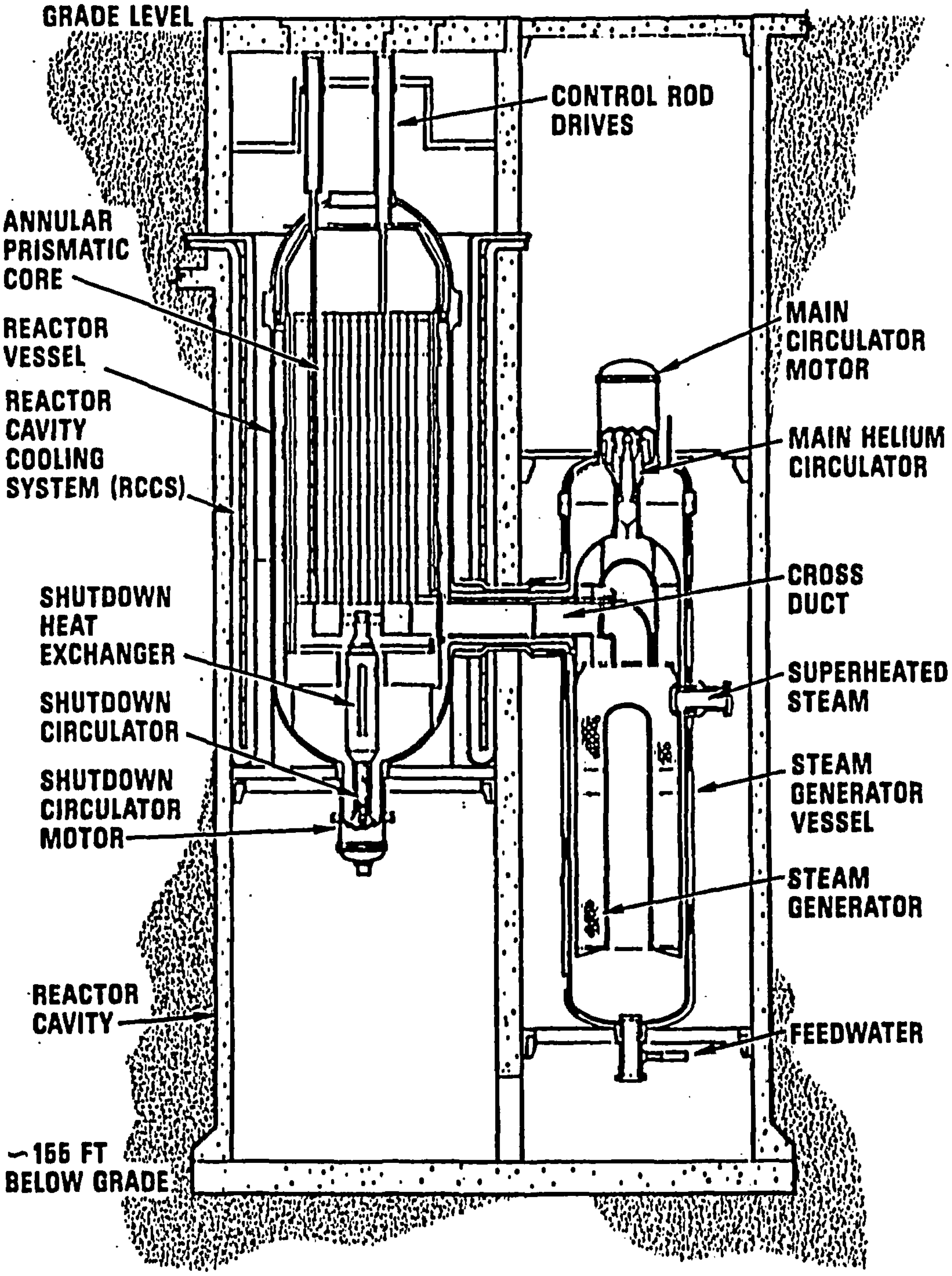

Figure 5.1 Principal features of vessel and heat removal systems
Source: DOE, 1986-3

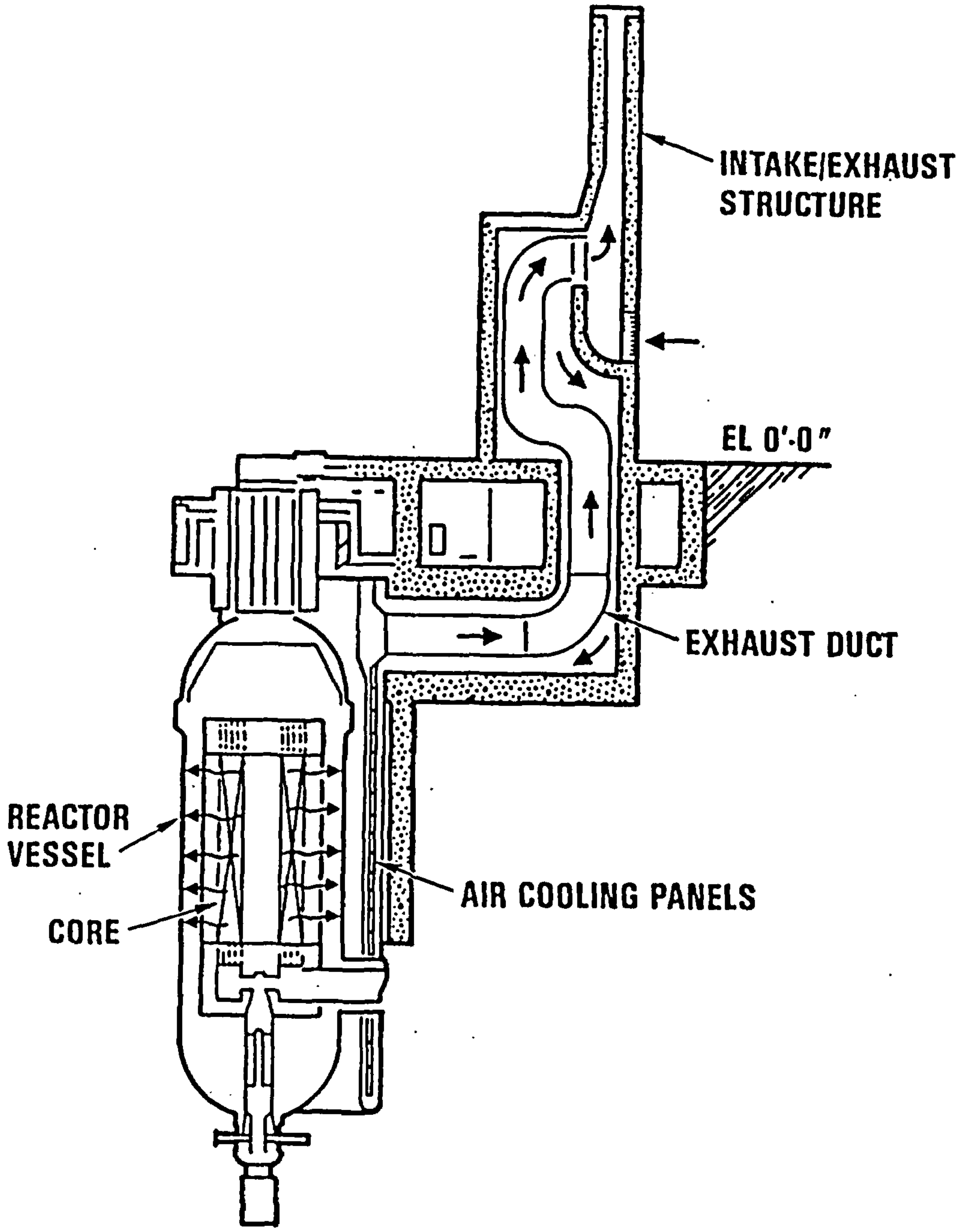

Figure 5.2 Passive reactor cavity cooling
Source: DOE, 1986-3

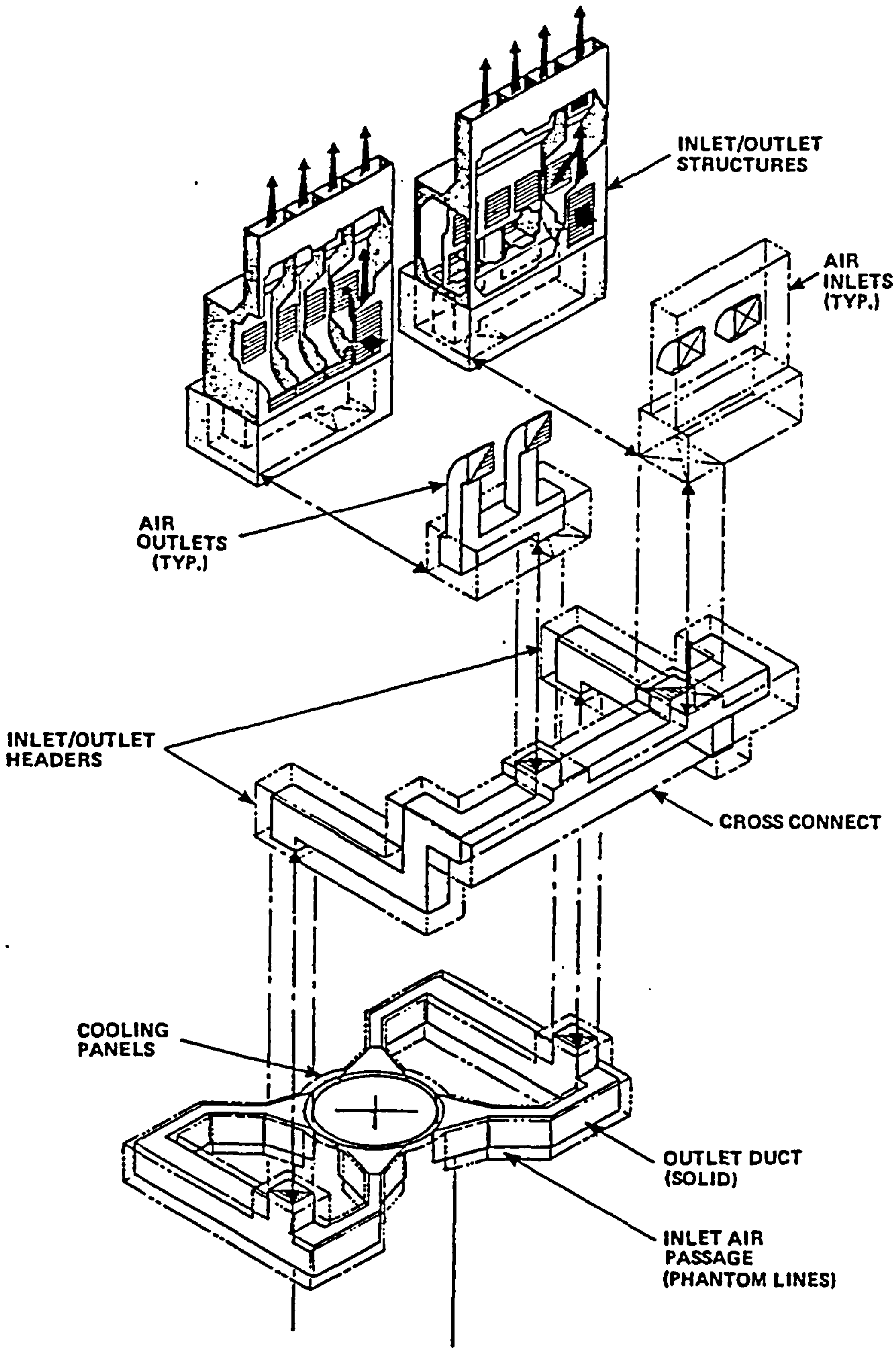

Figure 5.3 RCCS ductwork for air-cooling flow
Source: DOE, 1986-3

Table 5.1 Differences between reactor-vessel duty for MHTGRs and light-water reactors (LWRs)

Item	MHTGR	LWR
(1) Maximum Code-allowable temperature at operating pressure	800°F*	700°F
(2) Maximum Code-allowable temperature at ambient pressure	1000°F*	-
(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}	$< 10^{-3}$

*ASME Code Committee approval is being sought for limited times of exposure not to exceed 1000 hours.

6 PLANT ARRANGEMENT, REACTOR BUILDING, AND CONTAINMENT

6.1 Plant Arrangement

6.1.1 Description and Safety Objectives

The plant is divided into two distinct areas - the nuclear island (NI) and the energy-conversion area (ECA). A plant layout drawing that identifies both the NI and ECA buildings by number is shown in Figure 6.1. Two buildings, the operations center (5) and the NI warehouse (21), as well as portions of the main steam and feedwater piping (2) are considered part of the ECA, but they form interfaces with the NI. No portion of the ECA is proposed as safety related by DOE, including sources for cooling water. The staff has focused its review mainly on the buildings that DOE proposed as safety related, although at a later review stage other buildings will be reviewed at a level consistent with the current review level for light-water-reactor (LWR) buildings.

The NI safety-related buildings consist of four identical reactor buildings (1), two identical reactor auxiliary buildings (3), and the reactor service building (4), all of which are mostly below grade. A steel-framed maintenance enclosure with metal roofing and siding, a portion of which is illustrated in Figure 6.2, shelters the entire operating floor formed by the at-grade slab covers of the below-grade buildings. Located on the north side of the reactor service building are the NI cooling water building (8), personnel services building (6), and the radioactive waste management building (7). Also, part of the NI are the freestanding helium storage building (14) and the two liquid-nitrogen enclosures (11) separately adjacent to the east sides of each reactor service building. The reactor building is discussed separately in Section 6.2 because of its many safety functions, including its roles in decay-heat removal and the release of radionuclides in accordance with the mechanistic siting source term.*

The reactor service building (RSB) houses facilities, systems, and components shared by all four reactor modules. These include the new-fuel-storage area, fuel-handling machinery, a fuel sealing and inspection facility, a hot service facility, and provision for the storage of activated or contaminated nuclear steam supply system components (for example, helium-purification filters, control rods, shutdown cooling system circulators). The RSB also houses the remote-shutdown area (RSA), portions of the safety-related essential dc and essential uninterruptible electrical power supply systems, and the plant protection and instrumentation system. An at-grade washdown bay on the west wall of the maintenance enclosure provides for the cleaning of incoming fuel casks and equipment

*Indicates statement is particularly sensitive to change by evaluation of forthcoming DOE information.

and for the decontamination of outgoing spent-fuel casks. Railroad and truck access to the RSB is through the washdown bay. A 125-ton bridge crane serves this and other areas of the RSB, as well as the two reactor auxiliary buildings (RABs) and the four reactor buildings (RBs). The two RABs are located between each of two of the four RBs. These identical buildings contain a spent-fuel-storage pool, house portions of the essential power supply systems, and provide for occupational and routine offsite radiation control.

The non-safety-related grade-level personnel service building houses the fuel-handling control station and provides facilities for dealing with radioactive materials and personnel and equipment decontamination (for example, a hot-chemistry laboratory, decontamination facilities, a laundry, and clothing storage). It also houses 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, to 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. The subject of occupational exposure is discussed in Chapter 12.

6.1.2 Safety Issue - Location of Control Room and Protection of Reactor Operators

The control room is a portion of the operations center that is located at the interfaces between the nuclear island, the energy-conversion area, and the non-protected portion of the plant site. It is the staff's position that the control room and equipment associated with its function be considered as a vital area for security purposes (see Section 13.3, "Safeguards and Security") and be located within the nuclear island. Furthermore, the control room building must ensure protection of the reactor operators with respect to both internal and external events in a manner consistent with the control room habitability requirements for LWRs. Although the staff does not require that the control room necessarily house the safety-related equipment needed for the operators to perform their safety-related functions (see Section 13.2, "Role of the Operators"), the control room and other structures on the nuclear island must provide for both operator protection and access to the locations where this equipment is available.

6.1.3 Conclusions

With the exception of the control room location, the staff has not reviewed the plant overall layout and building designs to determine whether the arrangement and designs are satisfactory. The staff believes, nevertheless, that acceptable designs can be achieved at a later review stage based largely on contemporary LWR criteria. However, because the reactor plant cooling water system (see Section 9.4) interfaces with the service water system (see Section 10.4), which has its major components in the ECA, the proposed objective of excluding the entire balance of plant from consideration of interactions with safety-related equipment has not been met in this case.

6.2 Reactor Building

6.2.1 Design Description and Safety Objectives*

The reactor building (RB) is illustrated in Figure 6.2, which shows that it is predominantly a multicell, reinforced-concrete structure, set below ground. The lower cylindrical portion, or silo, contains the reactor and steam generator vessels and all related components. The portion containing the reactor is known as the reactor cavity and houses the reactor cavity cooling system (RCCS) panels and some RCCS inlet and outlet ducting. The upper rectangular prism portion houses most of the helium purification system (HPS) equipment, plant protection and instrumentation system (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 are 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 extends from elevation -10.67 meters 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 are of various thicknesses, depending on shielding and load requirements. The two major cells of the silo 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 crossduct. 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, has several hatchways for equipment access that are normally closed with concrete plugs. This upper slab provides biological shielding, as well as protection from external hazards.

The upper part of the RB is 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 is accessible only at some later time following shutdown. DOE states 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 leaktight, pressurized containment function, such as that in conventional LWRs, but instead provides for controlled venting. Small releases, as identified in Table 15.1, are 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 would provide overpressure protection for the RB and its contents. A large primary-coolant release from the reactor vessel 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 this 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.

*Indicates statement is particularly sensitive to change by evaluation of forthcoming DOE information.

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 proposed for the RB are to ensure adequate structural support for the safety-related systems, structures, and components it houses, to assure protection of the RCCS, and to provide some retention of radionuclides in accordance with the use of the mechanistic siting source term as described in Section 15.4. The RB must maintain the 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 plant protection and instrumentation system (PPIS) and the neutron control subsystem (NCSS). In serving to control, limit, and mitigate the spread of radioactive contamination or the release of radionuclides from the primary system, the RB design is based on radionuclide sources proposed by DOE and described in Sections 4.2.1, 11.1, and 15.5. That is, the siting source term arises mainly from the "liftoff" of the radionuclide inventory plated out on primary-system surfaces.

6.2.2 Scope of Review

The staff reviewed Section 6.1.1, "Reactor Building," in the Preliminary Safety Information Document (PSID) and associated this review with relevant information developed in other portions of this SER. Review emphasis and requests for additional information were directed at the following topics: (1) interpretation of relevant general design criteria (GDC), (2) protection of the RCCS, (3) residual heat transmission to the earth, (4) recovery-action assessments, (5) reactor-cavity failure, (6) overpressure-protection features, (7) combustible-gas control, and (8) the retention of radionuclides for event category II and III depressurization sequences.

6.2.3 Review and Design Criteria

DOE states that the MHTGR will meet the intent of GDC 16 ("Containment design") primarily by the fuel-particle coatings but does include the RB among the additional sequential barriers for controlling the release of radionuclides. DOE states, however, that GDC 38 ("Containment heat removal") is not applicable because the design meets the intent of GDC 16 and 34 ("Residual heat removal"). The staff's position on GDC 16 will be established later on the basis of a Commission decision; the staff finds DOE's position on GDC 38 to be acceptable.

In response to Comment 6-5, DOE stated that the MHTGR will meet the intent of 10 CFR 50.49, "Environmental Qualification of Electrical Equipment Important to Safety for Nuclear Power Plants." This response specifically considered degradation of equipment such as could be caused by high-energy-line breaks in the RB. The RB and the other safety-related buildings and structures will meet the intent of the following regulatory guides:

1.29 Seismic Design Classification (Rev. 3, September 1978)

1.60 Design Response Spectra for Seismic Design of Nuclear Power Plants (Rev. 1, December 1973)

1.61 Damping Values for Seismic Design of Nuclear Power Plants (Rev. 0, October 1973)

1.117 Tornado Design Classification (Rev. 1, April 1978)

1.142 Safety-Related Concrete Structures for Nuclear Power Plants (Other Than Reactor Vessels and Containments) (Rev. 1, October 1981)

1.143 Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants (only Section 5.2 is relevant to structures) (Rev. 1, October 1979)

Regulatory Guide 1.69, "Concrete Radiation Shields for Nuclear Power Plants," will be assessed for usefulness as the design develops.

During normal power operation, radioactive release from the RB must conform to the requirements of 10 CFR Part 50, Appendix I, for routine emissions and to the occupational exposure standards of 10 CFR Part 20. For event categories II and III, radionuclide releases must fulfill the 10 CFR Part 100-related requirements, as described in Section 15.1.

6.2.4 Research and Development

As proposed by DOE, the RB is essentially a state-of-the-art reinforced-concrete structure that includes steel supports, linings, and louvers. Although the design of these structures and components may be critical, no areas have been identified by DOE that require further development or inclusion in the Regulatory Technology Development Plan. Related areas of radionuclide transport through the vent paths and RCCS performance are discussed in Sections 4.2.5, 5.5.5, 6.3, and 11.1.

6.2.5 Safety Issues

A. Containment Function*

NRC will decide whether to require the reactor building (RB) or an effective alternative to provide a containment building comparable in purpose to that for LWRs and full conformance with GDC 16. Following evaluation of the new information discussed in the "Preface," the staff will complete its review of the DOE position that the RB need not provide pressure-retaining capability if the design objectives for the fuel are met.

B. Protection of Reactor Cavity Cooling System

Since the reactor cavity cooling system (RCCS) is the only decay heat removal system designated safety related by DOE, particular attention must be given to its protection by the RB. Many potential RCCS failure modes are identified in Section 5.5.5.I and failures by seismic events are discussed in Section 15.3. Although the staff believes that the RB can be designed to provide the necessary level of protection for the RCCS, including adequate seismic resistance

*Indicates statement is particularly sensitive to change by evaluation of forthcoming DOE information.

for the above-grade inlet and outlet RCCS structures, substantial further study and analysis will be required at later review stages.

C. Heat Transmission to the Earth

In the PRA, Appendix G.2, complete and sustained failure of all residual heat removal systems, including the RCCS, is postulated. For this case, DOE and the staff's contractors have provided analyses showing that direct heat transmission from the vessel to the reactor-cavity wall and subsequently the earth could offer a feasible alternative to RCCS operational availability. In addition, the safety analyses developed by the staff will permit DOE to assume that recovery of 25 percent of the RCCS flow capacity within 36 hours may be possible. In this case, it will be necessary that DOE describe the means for recovery as discussed in Safety Issue E below. Should DOE also wish to take credit for and publicly claim residual-heat removal without RCCS recovery, it would be necessary to design the reactor cavity to ensure that such heat transmission could be accomplished as predicted and without a potential for a large fission-product release. A major safety issue that would have to be addressed at that time would be the integrity of the reactor cavity with respect to the potential for and the consequences of concrete and vessel-support failures at the elevated cavity temperatures to be encountered during such a heat-transmission mode.

D. Overpressure Protection of Building and Contents

The design of the vent path from the RB cavity to the atmosphere proposed by DOE is based on a main-steamline break, a main-feedwater-line break, or an 82-square-centimeter helium leak caused by failure of the primary system pressure relief line. All these breaks are considered by both DOE and the staff to be within event category II (EC-II). At a later review stage, DOE will need to investigate in detail the design and capacity of the vent paths to determine the consequence for the building and the safety-related systems, components, and structures it contains from sequences with EC-III classification. This would include bounding event 5 (see Table 3.7), which would be initiated by a catastrophic failure of the crossduct vessel.

E. Recovery of Reactor Cavity Cooling System

In order to assume that 25 percent of the RCCS capacity can be recovered within 36 hours, it will be necessary for DOE to illustrate the RB's role in this process and demonstrate that nothing in the RB's design would preclude this recovery action within the allowed time limit.

F. Retention of Radionuclides*

In the safety analysis, DOE takes credit in both EC-II and EC-III sequences for the reduction of radionuclide releases within the RB by both radioactive decay and deposition in the "tortuous" vent paths. Further analysis and information from a research program and prototype-plant testing to support this credit, as indentified in Sections 4.2.4 and 11.1.4 and Chapter 14, are needed. In the event this work is not fully supportive or fission-product retention in the

*Indicates statement is particularly sensitive to change by evaluation of forthcoming DOE information.

fuel is less than projected and used in DOE's current analyses, or more margin is desired in the containment design, DOE should be prepared at a later review stage to consider design features to increase the necessary levels of retention by the reactor building. One option could be the installation of appropriate filters in the flow path to the environs.

G. Combustible-Gas Control

In Section 15.2.6.1 the staff discusses the potential for combustible-gas generation and states that substantial further information and review are needed to resolve this concern. DOE proposes that combustible gas be released at the safety-relief-valve discharge and vented directly to the steam generator cavity. At a later review stage, it will be necessary to assess the potential for and the effects of a combustible-gas explosion on the reactor building and its safety-related contents. At the present time the staff believes that ducting from the relief-valve discharge to outside the reactor building, as in Fort St. Vrain, will be needed to prevent potential accumulation of explosive mixtures.

H. Hinged-Louver and Blowout-Panel Designs

Movable hinged louvers in the RB provide for steam-generator-cavity isolation, and the blowout panels between the reactor and steam generator cavities provide for isolation between these two cavities. In the case of substantial primary-coolant or steam releases, these panels and louvers are designed to open to prevent excessive pressures in either cavity. In the reactor cavity, excessive pressures could impede RCCS performance by collapsing the RCCS structure. In response to Comment 6-1, DOE stated that the louvers would be of sufficiently light construction to open well before dangerous pressures would be reached in the cavities. The staff agrees that blowout panels, as well as louvers, that cannot fail to open under given differential pressures can readily be designed. Nevertheless, since opening of the blowout panel and louvers would protect the safety-grade RCCS, detailed design information for these blowout panels and louvers should be available at a later review stage to provide assurance of their opening at sufficiently low pressure differentials.

I. Cavity Flooding

The staff has not yet investigated potential hazards that may be associated with natural, accidental, or intentional flooding of the reactor cavity and subsequent contact of the reactor vessel and other equipment and structures located below grade with large quantities of water. The present design provides for a continuous waterproof membrane, water stops between joints, instrumented sumps, and siting criteria or provisions that would preclude surface flooding. If inflow to the sumps exceeds the removal capacity, DOE states that the potentially affected systems and components will be placed in a safe condition and appropriate remedial measures will be taken. At a later review stage, DOE is expected to investigate potential safety concerns related to the flooding of the below-grade equipment, including the potential for significant thermal shock to the vessel system and the possible desirability of preserving the ability to intentionally flood the reactor cavity with respect to possible recovery actions for certain EC-III and possibly EC-IV sequences.

6.2.6 Conclusions

The safety-related functions of the reactor building can be performed by a properly designed structure. Although the currently proposed design appears to satisfy the proposed requirements, the safety issues cited indicate that changes may be needed. As the design progresses, some of the DOE evaluations remain to be confirmed quantitatively and the identified safety issues need to be addressed. The public policy issue pertaining to the containment function will be determined by the Commission.

6.3 Containment Criteria and Design

This section is reserved for the evaluation of the expected new information on containment adequacy as discussed in the "Preface" to this SER.

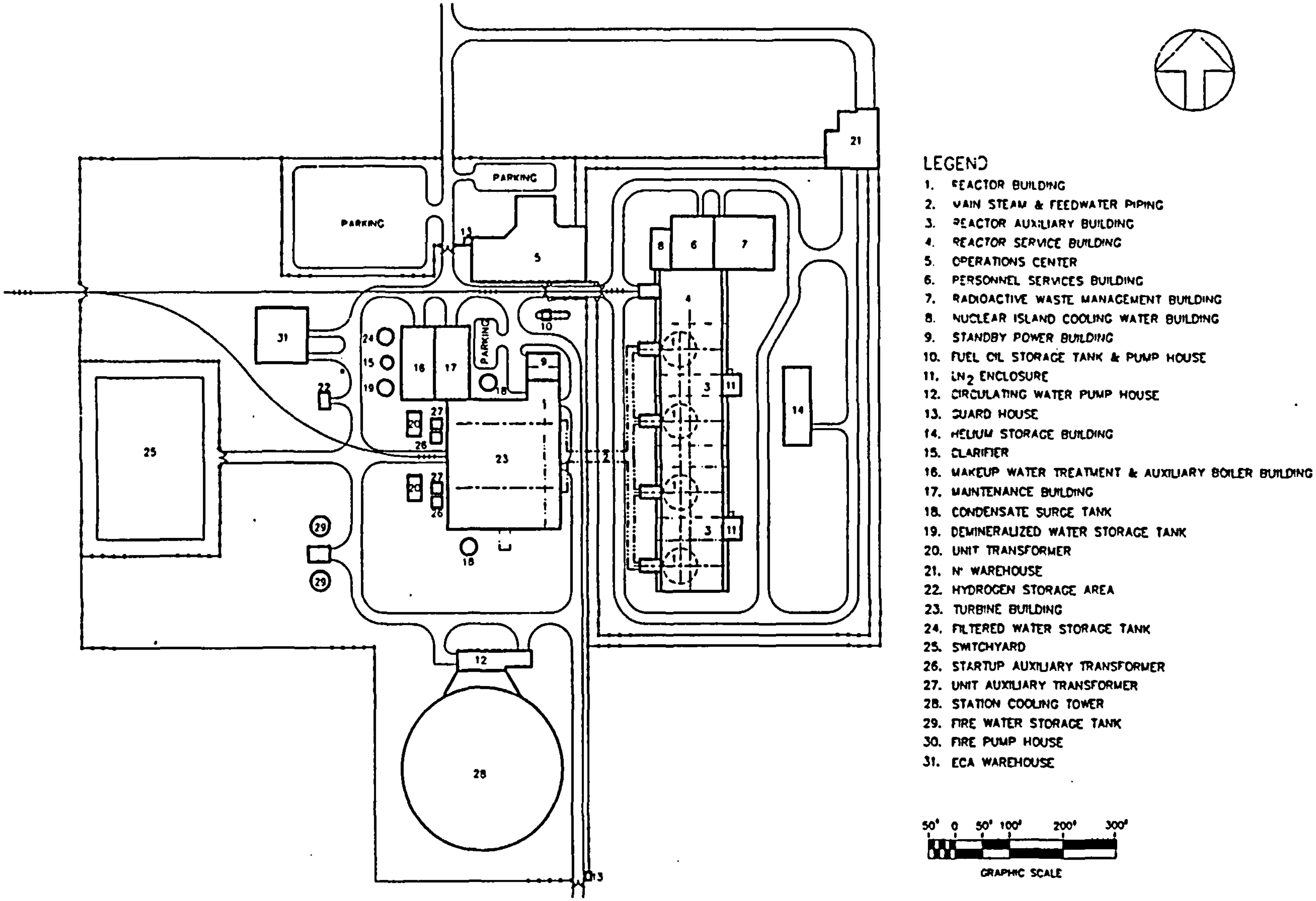

Figure 6.1 Plant building arrangement
Source: DOE, 1986-3

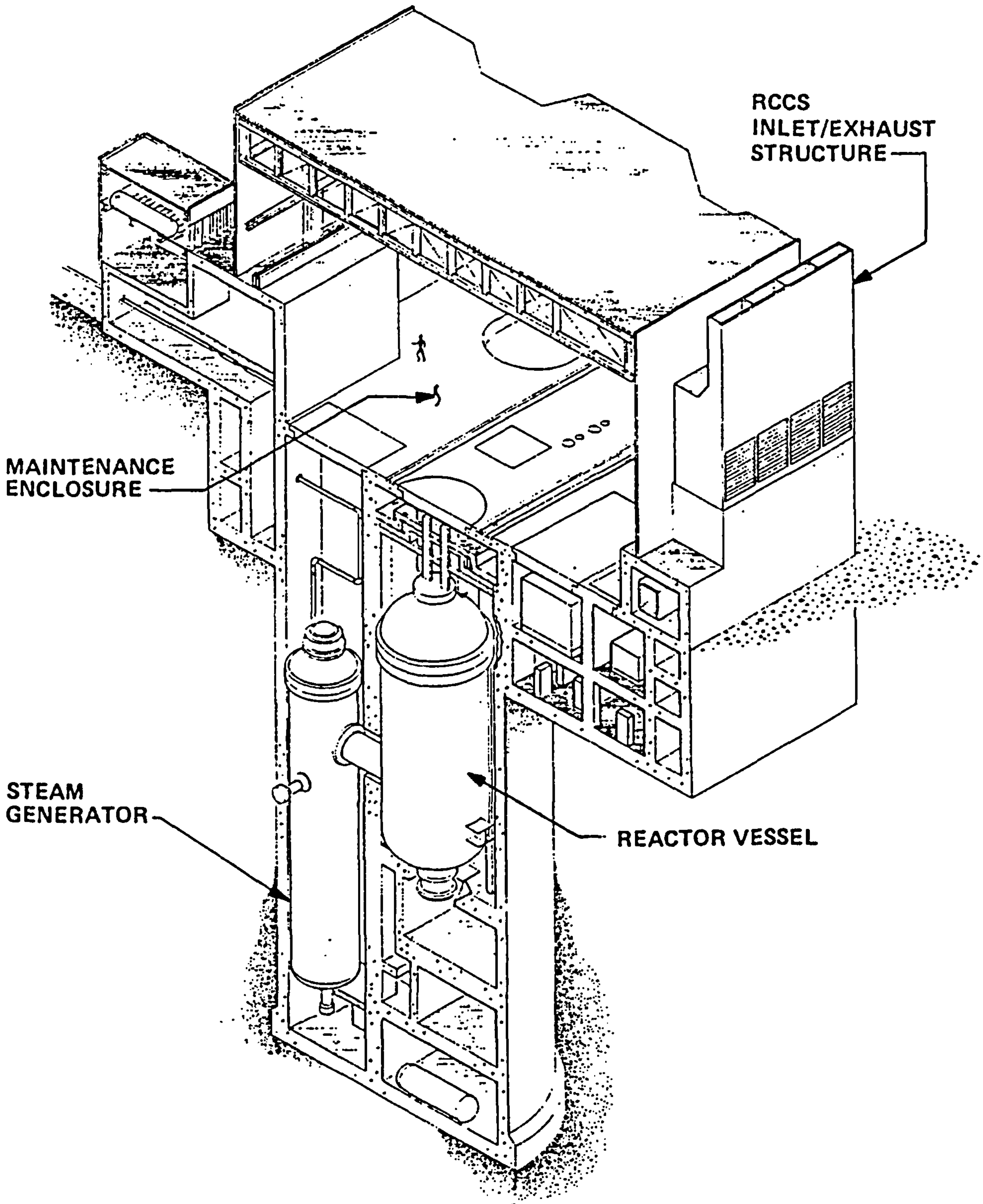

Figure 6.2 Isometric view through reactor building
Source: DOE, 1986-3

7 PLANT PROTECTION, INSTRUMENTATION, AND CONTROL SYSTEMS

7.1 General Description and Design Process

Chapter 7, "Plant Protection, Instrumentation and Control System," of the Preliminary Safety Information Document (PSID) describes a design that provides for fully automatic control for the four reactor modules and the two turbine-generator systems constituting the MHTGR power plant. This arrangement is shown in Figure 7.1, where complete interconnection between the reactor modules and the turbine-generator sets is indicated for both control systems and steam flows. Automatic control is used for both normal operations and abnormal events, with the goals of maintaining power generation, protecting plant investment, averting challenges to the safety system, and coping with the event categories postulated in the safety analyses without manual operator actions. The plant-safety-protection function is performed by separate safety-related instrumentation and control equipment. The multimodule plant is controlled principally from a single, main control room (CR) with limited monitoring and control functions available at a remote-shutdown area (RSA) located in the reactor service building (RSB) and in the plant protection and instrumentation system (PPIS) equipment room located in each reactor building (RB).

Three separate systems provide plant protection and automatic control for the MHTGR. These are the PPIS, some parts of which are safety related, and the two non-safety-related systems - the plant control, data, and instrumentation system (PCDIS) and the miscellaneous control and instrumentation group (MCIG). Each is evaluated separately in Sections 7.2, 7.3, and 7.4, respectively, with emphasis given to the review of safety-related portions of the PPIS. The staff review is based on information provided in corresponding sections of the PSID and DOE responses to staff comments and questions on this material.

DOE used a "top-down" design process for the plant protection, instrumentation, and control system. This process is termed by DOE the "integrated approach to design" and is outlined below. Because this design process was used for the design of all other MHTGR systems, both safety and non-safety related, the staff chose to illustrate this process by describing how it was used for the design of the plant protection, instrumentation, and control system, a system that lends itself well to the illustration.

The process began with four design goals: (1) maintain plant operations, (2) maintain plant protection, (3) maintain control of radionuclide release, and (4) maintain emergency preparedness. Institutional requirements of top-level regulatory criteria and utility/user requirements were allocated to the various functions associated with the four goals. DOE then conducted analyses and trade studies to identify selections that perform the functions according to the requirements. Figure 7.2 depicts the process in general, and Figure 7.3 illustrates how the design goals are proposed to be met.

The proposed features of the plant protection, instrumentation, and control system, which resulted from decisions based on the integrated approach, are:

(1) A modular, distributed control system is provided with performance monitoring and discretionary load allocation from a central control room to meet normal plant-operation requirements for all four reactor modules and the two steam turbine-generators of the plant.

(2) An independent and fully automatic protection system is provided that includes a remote-shutdown area (RSA) for monitoring and discretionary backup investment-protection actions. The design proposes that only the automatic reactor trip and loop shutdown portions of the PPIS are needed to meet 10 CFR Part 100 limits. Therefore, these are the only safety-related portions of the system.

(3) The bulk of the PPIS circuitry is local and contained in the PPIS equipment room for each reactor module. Neither the control room nor the RSA are proposed as safety related, although the RSA and PPIS equipment room are housed in safety-related structures.

(4) The main control room operators perform plant mission management tasks. The role is primarily a supervisory one of monitoring and confirming plant behavior. The operators can also execute discretionary manual actions, including cold shutdown of the reactors. Because of the automated systems, DOE has proposed that the operator have no safety-related role.

(5) The RSA is an alternate area from where an operator can achieve and maintain plant-shutdown conditions. Manual initiation of protective actions for each reactor is also available in each PPIS equipment room.

(6) The control room and RSA designs ensure that operators do not receive radiation exposure during accident conditions in excess of the limits specified in 10 CFR Part 20. As proposed, these areas do not require any special design features because of the low projected radionuclide releases from the MHTGR in offnormal conditions.

The system descriptions and evaluations given in this chapter provide background for the human-factors evaluations in Section 13.2 for the major man-machine interfaces within the plant and the role of the operators.

7.2 Plant Protection and Instrumentation System

7.2.1 Design Description and Safety Objectives

The plant protection and instrumentation system (PPIS) is an independent system of hardware and software provided to indicate plant status and to actuate automatically both the safety-related control system and the control systems that protect the plant investment. The PPIS, which is independent of the balance-of-the-plant instrumentation and control system, monitors selected process variables, compares the sensed values to preselected levels and, as required, commands and initiates predetermined corrective actions. The PPIS subsystems included to perform and support the above functions are (1) the safety protection subsystem, (2) the special nuclear area instrumentation subsystem, and (3) the investment protection system.

The safety protection subsystem contains the safety-grade equipment that provides the sensing and command features necessary to initiate a reactor trip using the outer control rods and the reserve shutdown control equipment (RSCE), and to initiate main-loop shutdown. The special nuclear area instrumentation subsystem provides certain plant-protection interlock and monitoring features. This includes the closure interlock for the vessel system (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 abnormal conditions. The special nuclear area instrumentation subsystem is proposed to contain only equipment that is not safety related. The investment protection subsystem provides the sense and command features necessary to initiate protective actions to limit plant investment risk. These actions include reactor trips, steam generator isolation and dump, and initiation of the shutdown cooling system (SCS). This system is proposed as non-safety related.

The safety-protection functions of the PPIS are implemented on a per-reactor basis with a fully automatic, remote-multiplexed, microprocessor-based protection system. The protection-system architecture consists 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 safety protection subsystem operator interfaces for each reactor module are located in the PPIS equipment room and the RSA. The operator interfaces include color video displays, function input devices, and keyboards. Since DOE proposed that no operator action be 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 safety protection subsystem are transmitted through a unidirectional isolator to the data management subsystem (DMS) for a display by the plant supervisory control subsystem (PSCS) in the main control room. The PPIS operator interfaces in the remote-shutdown area provide an operator the capability of initiating reactor trip or main-loop shutdown from a position remote from the main control room. No manual inputs to the safety protection subsystem are provided from the main control room. The reactor can be shut down, however, with normal plant-control equipment from the main control room, as discussed in Section 7.3.

7.2.2 Scope of Review

The staff reviewed Section 7.2, "Plant Protection Instrumentation and Control," of the PSID and DOE's responses to the staff's comments developed in the course of this review. The PPIS conceptual information, including the availability and relevance of proposed design criteria, was reviewed to determine whether the system could meet its safety-related functions and to establish that it would not be degraded by interfaces with other plant instrumentation and control systems not designed to be safety related. The staff reviewed the manual reactor trip system and selected other trip systems (Sections 7.2.5.A and E) for their adequacy and safety classifications, but this review did not include all of the trip systems proposed. It will be necessary at a later review stage to perform a comprehensive study of the reactor and equipment trip systems when more information, including a detailed probabilistic risk assessment, is available.

7.2.3 Review and Design Criteria

The applicable general design criteria are GDC 20 ("Protection system functions"), GDC 21 ("Protection system reliability and testability"), GDC 22 ("Protection system independence"), GDC 23 ("Protection system failure modes"), GDC 24 ("Separation of protection and control systems"), GDC 25 ("Protection system requirements for reactivity control malfunctions"), GDC 26 ("Reactivity control system redundancy and capability"), GDC 27 ("Combined reactivity control capability"), GDC 28 ("Reactivity limits"), and GDC 29 ("Protection against anticipated operational occurrences"). In Amendment 1 to the PSID, DOE made a commitment to meet the intent of these criteria. In addition, DOE identified the following applicable regulatory guides and agreed to meet their intent:

1.22 Periodic Testing of Protection System Actuation Functions (Rev. 0, February 1972)

1.29 Seismic Design Classification (Rev. 3, September 1978)

1.30 Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment (Rev. 0, August 1972)

1.47 Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems (Rev. 0, May 1973)

1.53 Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems (Rev. 0, June 1973)

1.62 Manual Initiation of Protective Actions (Rev. 0, October 1973)

1.63 Electric Penetration Assemblies in Containment Structures for Light Water-Cooled Nuclear Power Plants (Rev. 2, July 1978)

1.75 Physical Independence of Electric Systems (Rev. 2, September 1978)

1.89 Qualification of Class 1E Equipment for Nuclear Power Plants (Rev. 1, June 1984)

1.114 Guidance on Being an Operator at the Controls of a Nuclear Power Plant (Rev. 1, November 1976)

1.118 Periodic Testing of Electric Power and Protection Systems (Rev. 2, June 1978)

1.152 Criteria for Programmable Digital Computer Software in Safety-Related Systems of Nuclear Power Plants (Rev. 0, November 1985)

1.153 Criteria for Power, Instrumentation, and Control Portions of Safety Systems (Rev. 0, December 1985)

It should be noted that Regulatory Guide 1.153 endorses Institute of Electrical and Electronics Engineers (IEEE) Standard 603, "Standard Criteria for Safety Systems for Nuclear Power Generating Stations." As stated in the regulatory guide, this standard is intended to replace IEEE Standard 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations," which is cited in the Code of Federal Regulations. IEEE Standard 603 is applicable to both the instrumentation and control system described here and the electrical power system described in Chapter 8.

The staff concluded that these criteria commitments form a sufficient base for the design and review of the safety-related portions of the MHTGR at the conceptual design stage and that additional design and review criteria are unlikely to be needed at a later stage of review, with the exception of criteria that may be developed pertaining to the role of the operator, as discussed in Section 13.2.

DOE also proposed "10 CFR 100 Design Criteria," which were not found sufficiently detailed for use in either the proposed design or in the design review.

7.2.4 Research and Development

The PPIS will use digital, microprocessor-based sensor, data processing, communication, and logic, all of which require programmed software. DOE has stated that software programs are to be simple and factory preprogrammed and tested. For example, the computer logic may be "burned in" on a programmable read-only memory. The microprocessors are not intended to be "reprogrammed" in the field, and only commercially proved microprocessor logic will be used. The PPIS design will be kept as simple as possible to provide reliable protective action. While DOE believes that no new development will be needed, the staff reserves its opinion on this matter until a later review stage when more background information should be available. Furthermore, some development needs are anticipated to resolve the human-factors concerns discussed in Section 13.2.4.

7.2.5 Safety Issues

A. Manual Trip and Role of the Operators

Contrary to the proposed safety aspects of the design, it is the staff's position that an IEEE Class 1E-qualified means to manually trip the reactor must be available and accessible and that the operators therefore must perform safety-related functions. These staff positions are discussed in Section 13.2.4 based on the background provided in this chapter.

B. Sharing of Protection and Control Instrumentation Senors and Other Items

As a result of the staff's review, the PSID now clearly states that even though some common parameters are required for both protection and control functions, separate sensors are to be used to measure these parameters. To ensure independence of the protection system from the control system, additional review of shared items other than sensors will be conducted at a later design stage.

C. Non-Safety Classification of Portions of the Plant Protection and Instrumentation System

The PPIS is proposed to include some non-safety-related trip functions provided for investment protection only. Such a conceptual design will invariably lead to some portions of the PPIS being common to both safety- and non-safety-related portions. As a result, DOE has made a commitment that for any non-safety-related trips, IEEE Standard 603 criteria will be met by those portions of the system common to the safety-related portions. This topic will be investigated at a later review stage when a detailed failure modes and effects analysis is submitted.

D. Block Valve Closure Interlock System

It is the staff's position that the block valve closure interlock system, which is now part of the special nuclear area instrumentation subsystem, should be made safety related. This interlock is a component of the safety-related pressure relief system for the vessel system and, as such, should be considered to be safety related.

E. Actuation of Steam Generator Dump and Isolation Valves

It is the staff's position that at a later review stage consideration should be given to the safety classification of the actuation system for the safety-grade steam generator dump and isolation valves, the moisture-monitor portion of the investment protection subsystem, and the shutdown portion of the main loop on a signal from the steam generator dump and isolation valves. This is consistent with the staff's finding in Section 5.3.5 that the steam generator and its isolation system should be classified as safety related, and that there is a need to ensure main-loop shutdown in the event of a failure of the steam generator, as discussed in Appendix A.

F. Plant Protection System Status Monitoring

The staff will determine at a later review stage whether equipment that monitors the status of the plant protection system and plant-safety parameters, now a portion of the non-safety-related special nuclear area instrumentation subsystem, should be made safety grade.

7.2.6 Conclusions

Based on review of the conceptual design presented in the PSID, the staff concludes that the plant protection and instrumentation system can be implemented in an acceptable manner, subject to resolution of the identified safety issues and the concerns regarding the operators' role and human-factors issues discussed in Section 13.2 and comprehensive review of the trip systems. At a later review stage, the design will be reviewed in detail to ensure that the GDC and other applicable criteria have been fully satisfied.

7.3 Plant Control, Data, and Instrumentation System

7.3.1 Design Description

The plant control, data, and instrumentation system (PCDIS) consists of instrumentation and control hardware and software that automatically control the MHTGR plant from startup to full power and orderly return to a shutdown condition.

The subsystems of the PCDIS are (1) the plant supervisory control subsystem (PSCS), (2) the nuclear steam supply system (NSSS) control subsystem, (3) the energy-conversion area (ECA) control subsystem, and (4) the data management subsystem (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 are 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 provides monitoring and control for those systems that directly impact the continuity of power generation. The computer-based DMS provides plant-wide communication and centralized data processing. The DMS supports the PCDIS by transmitting control and monitoring communications between subsystems. The DMS is further described in Section 13.2.1.

7.3.2 Scope of Review

The staff reviewed Section 7.3, "Plant Control, Data, and Instrumentation System," of the PSID and DOE's responses to staff comments during the course of this review. DOE stated that the subsystems of the PCDIS have no safety design bases because none of them are required to perform any 10 CFR Part 100-related radionuclide-control functions. The staff generally agrees with this position and, therefore, has focused its review on interfaces with safety-related systems. As stated in Section 7.3.5.A, however, the staff has some reservations with respect to power-generation stability in the event of transients.

7.3.3 Review and Design Criteria

DOE stated that MHTGR safety is, by design, insensitive to non-safety-grade control-system failures or malfunctions. Therefore, no safety design criteria are specified. In addition, though not required for safety, the control systems are designed to reduce the severity of plant transients and thus avoid protection-system action. The staff agrees that no safety-related criteria are needed other than those that may be required to resolve the safety issues identified in Section 7.3.5 and those relative to the human-factors principles discussed in Section 13.2.3.

7.3.4 Research and Development

The plant control systems are to be totally automated by utilizing digital computers and fault-tolerant practices. The equipment described is within the state of the art and, therefore, no research and development program is required other than that which may be necessary to address the human-factors concerns discussed in Section 13.2.4. The staff will expect, however, some form of demonstration that the PCDIS will meet its stability objectives before full-term operation of a multimodule MHTGR.

7.3.5 Safety Issues

A. <u>Power-Generation Stability</u>

The interconnected, four-reactor, two-turbine-generator system is a novel arrangement that has no precedent in the nuclear power industry. Control of this system appears to be ambitious, particularly since DOE's goal is to maintain reduced but stable electric-power output in the event of loss of offsite power or a trip of either a single module or a single turbine-generator set. The staff plans to review further the PCDIS with respect to these stability objectives and potentials for challenges to safety-related components. More is said on this matter in Section 10.1.5.A with respect to the steam and energy conversion systems.

B. Isolation of the Normal Plant Control Systems From the Plant Protection and Instrumentation System (PPIS)

The only PCDIS that has a functional interface with the PPIS is the data management subsystem (DMS) because the PPIS transmits information to the DMS. The DMS equipment is physically separate, has no safety-related functions, and is electrically isolated from the PPIS by a safety-grade, optical, signal-cable/coupler design that carries unidirectional signals. Although the staff finds this isolation proposal conceptually acceptable, it will be necessary to review this interface provision in detail at a later review stage.

C. Control-System Failures

Failures in the non-safety-grade control systems will cause the MHTGR to react in some manner to failed or erroneous control actions. At a later review stage, the staff plans to investigate control-system failures in sufficient detail to establish that none of these failures can put the MHTGR outside the scope of event category II sequences.

7.3.6 Conclusions

DOE has described the plant control, data, and instrumentation systems and stated that these systems will have no safety design bases. The staff agrees with DOE that these control systems are not required to perform any 10 CFR Part 100-related radionuclide-control functions because the MHTGR is committed to be designed as insensitive to non-safety-grade control-system failures or malfunctions. Based on review of the PCDIS and the expectation of the successful resolution of the safety issues identified above, the staff concludes that the design can be implemented in an acceptable manner.

7.4 Miscellaneous Control and Instrumentation Group

7.4.1 Design Description

The miscellaneous control and instrumentation group (MCIG) will sense, acquire, and process various data from the plant for display to the plant operator and/or retention for historical purposes. The subsystems that support these functions 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.

7.4.2 Conclusions

DOE proposes that the subsystems described above will not perform any safety-related functions and will have no interfaces with the safety-related instrumentation and control systems. Based on this conceptual information, the staff concludes that further review at this design phase is not necessary. It is the current staff position, however, that these subsystems should, in general, meet the criteria (for example, regulatory guides and industry standards) current for LWRs unless it can be justified at a later design stage that different criteria are appropriate.

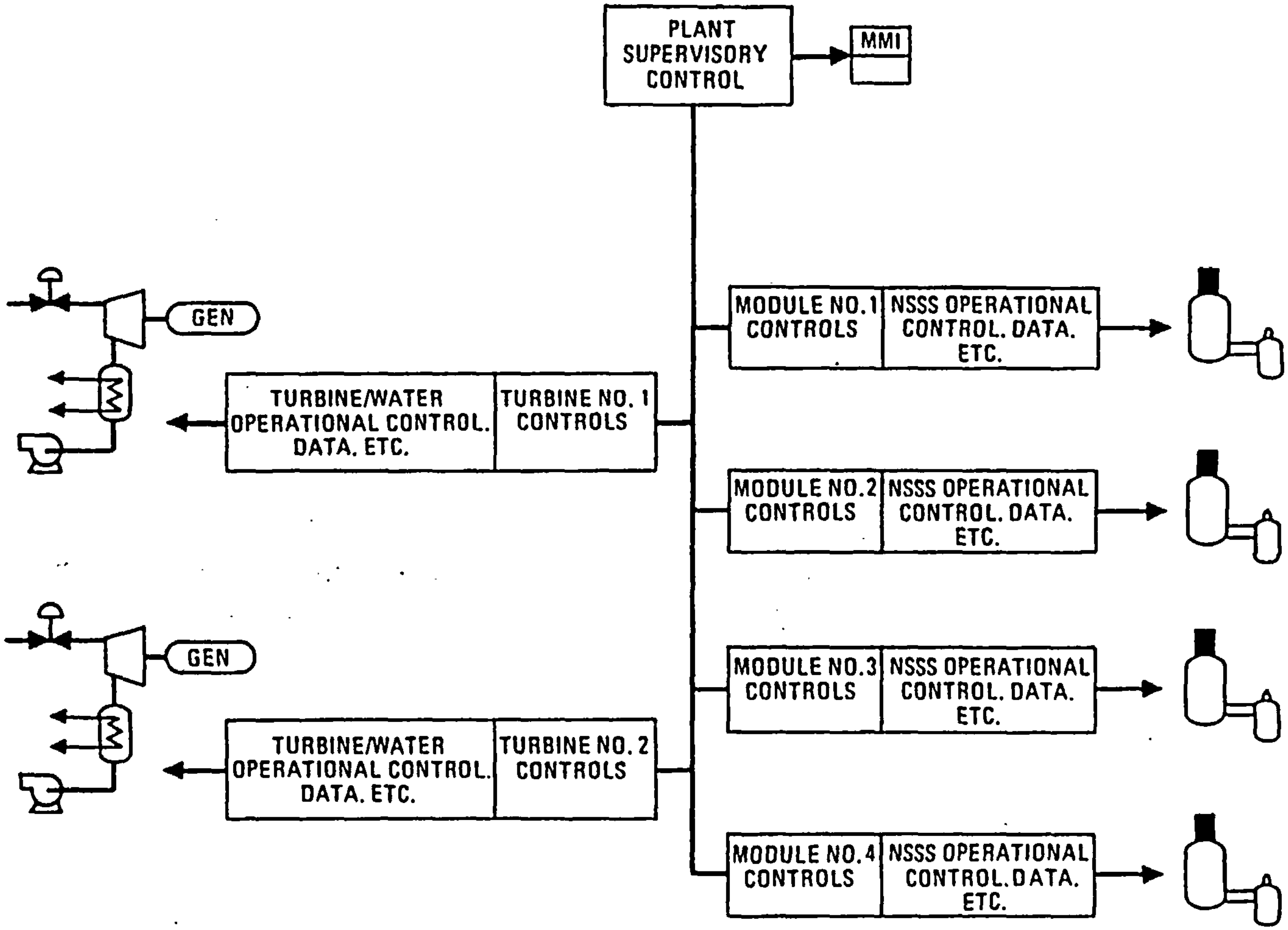

Figure 7.1 Plant control system
Source: DOE, 1986-3

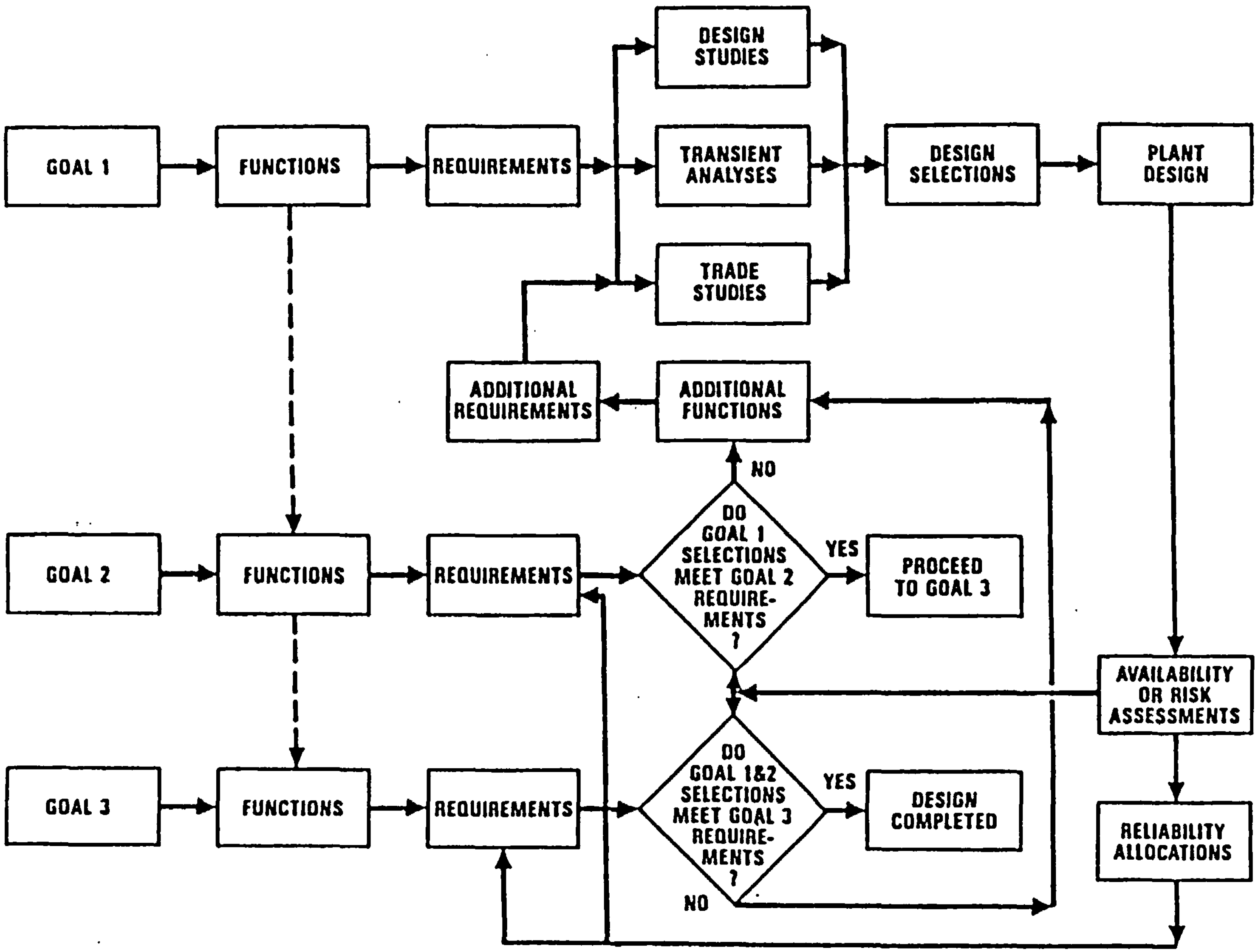

Figure 7.2 Integrated approach to design
Source: DOE, 1986-3

MHTGR TOP LEVEL GOALS AND DESIGN SELECTIONS

GOAL 1: PLANT OPERATION

- PCDIS
- CONTROL ROOM OPERATIONS
- LOCAL OPERATIONS
- SURVEILLANCE AND MAINTENANCE

GOAL 2: PLANT PROTECTION

- INDEPENDENT PPIS
- UNIDIRECTIONAL INTERFACE FROM PPIS TO PCDIS DISPLAYS IN CONTROL ROOM
- REMOTE SHUTDOWN AREA PPIS OPERATOR INTERFACES
- SURVEILLANCE AND MAINTENANCE

GOAL 3: RADIONUCLIDE RELEASE CONTROL

- AUTOMATIC PPIS ACTUATION OF OF RESERVE SHUTDOWN
- SURVEILLANCE AND MAINTENANCE

GOAL 4: EMERGENCY PREPAREDNESS

- COMMUNICATION WITH OFFSITE AUTHORITIES AND PUBLIC

Figure 7.3 Relationship of design goals to protection, instrumentation, and control systems
Source: DOE, 1986-3

8 ELECTRICAL SYSTEMS

8.1 Overall Design

8.1.1 Design Description and Safety Objectives

The electrical systems consist of the essential uninterruptible power supply (UPS) system, the essential dc power system, the offsite power and main generator transmission system, the nonessential ac distribution system, the nonessential UPS system, the nonessential dc power system, grounding, lightning protection, heat tracing, cathodic protection, the communication systems, and the lighting and service power systems.

The MHTGR design proposes 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 essential UPS system and the essential dc power system 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 light-water reactors (LWRs) and should be satisfied by onsite batteries and associated power conversion and distribution equipment.

8.1.2 Scope of Review

The staff review was based on information provided in Chapter 8 of the Preliminary Safety Information Document (PSID) and on DOE responses to staff comments and questions on this material. The review focused on the essential UPS system and the essential dc power system discussed in Sections 8.2 and 8.3, respectively. The conceptual designs and criteria for the other electrical systems were reviewed only to verify that they would not degrade the essential systems. This is consistent with findings from the safety analysis (Chapter 15), which confirm that only modest requirements exist for safety-related power and that such can be satisfied by onsite batteries.

8.1.3 Review and Design Criteria

In Amendment 1 to the PSID, DOE made a commitment to meet the intent of General Design Criterion (GDC) 17 ("Electric power systems") and GDC 18 ("Inspection and testing of electric power systems") for the essential electric power systems only. Also, DOE made a commitment in Amendment 7 to meet the intent of the following regulatory guides with respect to the essential electric power systems:

1.6 Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems (Rev. 0, March 1971)

1.22 Periodic Testing of Protection System Actuation Functions (Rev. 0, February 1972)

1.29 Seismic Design Classification (Rev. 3, September 1978)

1.32 Criteria for Safety-Related Electric Power Systems for Nuclear Power Plants (Rev. 2, February 1977)

1.41 Preoperational Testing of Redundant Onsite Electric Power Systems To Verify Proper Load Group Assignments (Rev. 0, March 1973)

1.47 Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems (Rev. 0, June 1973)

1.53 Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems (Rev. 0, June 1973)

1.75 Physical Independence of Electric Systems (Rev. 2, September 1978)

1.89 Qualification of Class 1E Equipment for Nuclear Power Plants (Rev. 1, June 1984)

1.93 Availability of Electric Power Sources (Rev. 0, December 1974)

1.106 Thermal Overload Protection for Electric Motors on Motor Operated Valves (Rev. 1, March 1977)

1.118 Periodic Testing of Electric Power and Protection Systems (Rev. 2, June 1978)

1.128 Installation Design and Installation of Large Lead Storage Batteries for Nuclear Power Plants (Rev. 1, October 1978)

1.129 Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Nuclear Power Plants (Rev. 1, February 1978)

1.131 Qualification Tests of Electric Cables, Field Splices and Connections for Light-Water-Cooled Nuclear Power Plants (Rev. 0, August 1977)

1.153 Criteria for Power, Instrumentation, and Control Portions of Safety Systems (Rev. 0, December 1985)

Also in Amendment 7, DOE stated that it would assess the use of the following regulatory guides as the MHTGR design progresses:

1.81 Shared Emergency and Shutdown Electric Systems for Multi-Unit Nuclear Power Plants (Rev. 1, January 1975)

1.120 Fire Protection Guidelines for Nuclear Power Plants (Rev. 1, November 1977)

The staff concluded that these criteria are a sufficient base for the design and review of the safety-related portions of the MHTGR electrical systems at this conceptual design stage.

8.1.4 Research and Development

Both DOE and the staff find that the electrical systems for the MHTGR are within the state of the art and therefore a development program is not needed.

8.1.5 Safety Issues

Based on the conceptual information presented in the PSID, the staff identified capacity and duration concerns with respect to the essential UPS system and the essential dc power system.

A. Power Capacity for Operator Information Needs and Actions

The general concern regarding the role of the operators (see Section 13.2) has had an impact on the design of the essential electric power systems with respect to capacities. Specifically, the present design does not classify the operators' information needs for postaccident monitoring and communication as safety related, and therefore the essential power systems do not make provisions for this additional load under all potential loss-of-power scenarios. In addition, power is needed for the depressurization of the primary system through the helium purification system, as described in Section 9.2.5.D, when the reactor cavity cooling system is postulated to be unavailable when needed for 36 hours (bounding event 3) and with respect to reactor shutdown, as indicated in Section 4.3.5.C. The staff's position is that these information needs and actions are safety related and that adequate essential electrical power for related equipment must be available.

B. Power-Duration Needs and Station Blackout

Because of the identified additional operator information and action needs, the staff requires that adequate essential power must be available for periods substantially longer than 1 hour and that the design objectives for such power systems should take guidance from staff actions pertaining to Unresolved Safety Issue (USI) A-44, "Station Blackout." The acceptability of the DOE design and its objectives, including duration as well as capacity, should be reviewed in detail at a later review stage based on the power needs of event category II and III sequences and the fact that the staff considers that adequate station power can be restored within 36 hours following a station blackout of the MHTGR.

8.1.6 Conclusions

On the basis of the safety analysis of Chapter 15, the staff concludes that it may be possible for essential electric power needs for the MHTGR to be provided by station batteries and associated power conversion and distribution equipment. However, it should be noted that the resolution of some of the safety issues listed in Table 1.5 could result in the need for essential onsite ac power. With respect to the essential electric power supplies described in the PSID, the staff concludes that DOE has conceptually described a plant electrical system that can be transformed into a design that will satisfy NRC requirements. At a later review stage, the staff will make a final determination of the acceptable levels for capacity and duration for the essential power systems and the conformance of this design with the guidance provided by staff actions pertaining to USI A-44.

8.2 Essential Uninterruptible Power Supply System

8.2.1 Design Description and Safety Objectives

The essential uninterruptible power supply (UPS) system is designed to be a reliable electric system consisting of four redundant and independent channels, each with adequate capacity, capability, and reliability to supply power for the essential plant loads. The essential UPS system includes 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 of the essential UPS system channels 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 essential dc power system 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 plant protection and instrumentation system (PPIS) and to some equipment not related to safety. The four plant UPS channels will serve all four reactor modules, each of which requires four UPS channels (for example, plant UPS channel A will serve all four reactor module PPIS loads that require UPS channel A power). Each channel will consist 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 power to ac power.

8.2.2 Scope of Review

The staff's review focused on the safety-related portion of the essential UPS and the interfaces with equipment not considered to be safety related.

8.2.3 Review and Design Criteria

The review and design criteria include those listed in Section 8.1.3. No development of new criteria is expected, except as related to the safety issues identified herein.

8.2.4 Research and Development

Both DOE and the staff find that the electrical systems for the MHTGR are within the state of the art, and therefore the need for a development program has not been identified.

8.2.5 Safety Issues

A. Sharing of Essential Uninterruptible Power Supplies Among Reactor Modules

The sharing issue is discussed in conjunction with the essential dc power system in Section 8.3.5.A.

B. Fault Clearing on Essential Uninterruptible Power Supply Channels

In response to staff comments on the MHTGR design approach to clearing faults on a branch circuit of the essential UPS system, DOE described a method that relied on the normal ac power source. The staff is concerned about this design because this feature would not be available under station blackout conditions. Consequently, the staff will require further discussion of fault clearing at a later review stage. It should be noted that Regulatory Guide 1.75, "Physical Independence of Electric Systems," does not permit non-safety-related loads on safety-related buses as proposed by DOE.

8.2.6 Conclusions

Based on the review of the essential UPS system, the staff concludes that DOE has described a system that can be transformed into a design that will satisfy the staff's requirements. Fault clearing and the power capacity and duration requirements remain open, as discussed above and in Sections 8.1.5.A and B.

8.3 Essential DC Power System

8.3.1 Design Description and Safety Objectives

The essential dc power system 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 panelboards 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, each of which requires four dc channels (for example, plant dc channel A will serve all four reactor module dc loads that require dc channel A power). Each channel will have 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 nonessential motor control center, is proposed so that any unit can be removed from service without degrading the systems to which dc electrical power is provided by each channel.

The battery chargers will normally supply dc power for 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.

Maintenance and surveillance of these power supplies are to be performed under the cognizance of the central control room operators to ensure that the 2-out-of-4 protection voting logic in each reactor is always operational and has the appropriate redundancy. DOE states that these maintenance and surveillance activities are simpler and easier to monitor and administer in the selected configuration than in the case of using 4 independent power supplies for each reactor module, which would involve a total of 16 supplies to maintain and test.

The safety objectives proposed by DOE for the essential dc system are to supply power to

(1) the rectifier-inverter assembly and the essential UPS system, which in turn supplies power needed by the safety protection subsystem of the PPIS to sense any upset conditions and command (initiate) appropriate remedial actions, such as a reactor trip or main-loop shutdown

(2) shut the steam generator isolation valves to limit the total amount of water or steam available for ingress following a steam generator tube leak

(3) actuate the reserve shutdown control equipment, which can dump boronated pellets into the core for failures of control-rod insertion or large moisture-ingress events

(4) battery room exhaust fans

In addition, the staff requires that essential power be available to satisfy station blackout needs and criteria, to satisfy operator needs as discussed in Section 13.2, and to depressurize the primary system. Other possible accident-mitigation power-supply needs may be identified at later review stages.

8.3.2 Scope of Review

The review focused on the ability of the essential dc power system to meet its safety objectives.

8.3.3 Review and Design Criteria

In addition to the general design criteria and the regulatory guides identified in Section 8.1.3, DOE has identified several industry standards as applicable to the essential dc power system. The basis for selection of batteries, with regard to installation, capacity, and capability, was to meet the requirements of IEEE Standards 308, 484, and 485. The reliability of the dc power supplies is to be ensured by periodic discharge of the batteries per IEEE Standard 450. Physical independence of redundant essential dc power systems is to be maintained by locating the batteries, charger, and switchboards of each channel in separate rooms and maintaining minimum separation distances between redundant essential circuits and between essential and other circuits, in accordance with the requirements of IEEE Standard 384. All cables are to be tested in accordance with IEEE Standard 383 to ensure ability to perform the intended function under expected ambient and accident temperatures during plant life.

8.3.4 Research and Development

Both DOE and the staff find that the electrical systems for the MHTGR are within the state of the art, and therefore the need for a development program is not evident.

8.3.5 Safety Issues

A. Sharing of Essential DC Power Among Reactor Modules

The MHTGR conceptual design utilizes four independent essential UPS and dc power supply systems that are, in effect, shared among the four reactor-module instrumentation systems. GDC 5 ("Sharing of structures, systems and components") generally discourages sharing but permits it in cases where it can be shown that it does not impair the ability of safety equipment to perform safety functions. Regulatory Guide 1.81, "Shared Emergency and Shutdown Electric Systems for Multi-Unit Nuclear Power Plants," was, however, developed on the basis of GDC 5 and experience that showed a number of undesirable interaction effects when electrical systems were shared. The recommendations of Regulatory Guide 1.81 with respect to dc systems clearly state they should not be shared, and the staff cited this regulatory guidance in discussions with DOE. In response, DOE stated that the MHTGR overall design and safety features are radically different from those of the current LWR designs and that the four reactors used in the

MHTGR design form a single power plant station with respect to power generation and control. DOE stated, in addition, that the essential dc and essential UPS power systems are totally separate and independent of the nonessential dc and UPS power systems used for normal plant control.

The staff agrees with the power-supply sharing proposed by DOE for the conceptual design. This issue will be reconsidered, however, at later review stages from the standpoint of common-cause and common-mode failure potentials and the interdependence of all four models on the status of the essential power supply for a single module. The staff believes it likely that the conceptual DOE position can be supported because of the long recovery time available to mitigate event category III sequences, because the MHTGR places so little reliance on power supplies for ensuring plant safety, and because any advantages of having a greater number of power supplies can be outweighed by the advantages involving easier maintenance and surveillance with an overall plant system limited to four essential dc and UPS systems.

B. Physical Independence

At a later stage of review, the staff will determine conformance with Regulatory Guide 1.75, "Physical Independence of Electric Systems." At that time, such questions as the connections of essential buses to nonessential ac power supplies will be considered.

8.3.6 Conclusions

On the basis of the review of the essential dc system, the staff concludes that DOE has described a system that can be transformed into a design that will satisfy NRC requirements. The issues of power-supply sharing, physical independence and power requirements for station blackout, and other needs remain open. As indicated previously, a power-supply duration of 1 hour will not be acceptable.

8.4 All Other Nonessential Electrical Systems

8.4.1 Design Description and Safety Objectives

These nonessential electrical systems have objectives that relate to safety, but these objectives have not been established at this stage of the review.

8.4.1.1 Offsite Power and Main Generator Transmission Systems

The offsite power transmission system will consist of two physically separate and independent circuits from the transmission network that, 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.

8.4.1.2 Nonessential AC Distribution System

The nonessential ac distribution system will provide electric power at 4160 V, three phase, and 480 V or less, three phase and single phase, 60 Hz to electrical switchgear associated with each generator to feed the plant's auxiliary equipment and services. The nonessential ac distribution system is to be normally fed from each generator unit 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.

8.4.1.3 Nonessential Uninterruptible Power Supply System

The nonessential UPS system 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 unit.

8.4.1.4 Nonessential DC Power System

The nonessential dc power system 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.

8.4.1.5 Grounding, Lightning Protection, Heat Tracing, and Cathodic Protection

Equipment grounding is designed to ensure personnel safety by connecting the plant's non-current-carrying metallic parts to the grounding grid. System grounding provides fast, selective clearing of the plant's ground faults to limit equipment damage.

Lightning protection provides a metallic, low-impedance path to earth to direct lightning strikes and is designed to prevent lightning current from passing through the nonconductive parts of a building or structure in the plant.

Heat tracing provides electric heat, as required, to the plant's piping or vessels in order to maintain a desirable temperature in the liquids inside the piping or vessels.

Cathodic protection is designed to arrest corrosion on the plant's underground and underwater structures by passing direct current from anodes in the electrolyte to the structures to be protected.

8.4.1.6 Communication Systems

The communication systems will provide separate, independent, and diverse types of intraplant communications between essential plant areas and the control room and plant-to-offsite communications to locations remote from the plant during normal operation or under emergency conditions.

8.4.1.7 Lighting and Service Power Systems

The lighting system will provide normal ac and emergency ac and dc lighting to support plant activities. The service power system will provide ac power to service outlets located throughout the plant for use with portable equipment, tools, and lighting.

8.4.1.8 Security System Power Requirements

DOE has specified a dedicated security backup generator and dedicated security UPS for the sole function of providing power for the security system, which includes exterior lighting needed for security. The requirements for such power supplies are contained in 10 CFR 73.55. Section 13.3 provides additional information.

8.4.2 Scope of Review

The staff did not review these nonessential systems because it did not believe that they affected significantly any fundamental safety decisions pertaining to the MHTGR conceptual design. They will be reviewed at a later design stage in a manner consistent with the then-current practices for LWRs.

8.4.3 Review and Design Criteria

These nonessential systems should be reviewed at a later review stage to ensure designs based on criteria (for example, regulatory guides and industry standards) that may pertain to the design and review of nonessential electrical systems consistent with the then-current practices for LWRs.

8.4.4 Research and Development

Since these nonessential systems are not unique to the MHTGR and are believed to be well within the state of the art, no special research and development programs pertaining to the MHTGR are needed.

8.4.5 Safety Issues

Safety issues will be determined and proposed solutions will be evaluated at a later review stage.

8.4.6 Conclusions

The staff has no conclusions because it did not review these systems for the reasons stated in Section 8.4.2.

9 SERVICE SYSTEMS

The service systems for the MHTGR are described in Chapter 9 of the Preliminary Safety Information Document (PSID). Only selected services systems are discussed below and reviews of many other systems judged less important to the staff's conceptual design review conclusions have been deferred to a later stage, as discussed in Section 9.7.

9.1 Fuel Handling and Storage

9.1.1 Design Description and Safety Objectives

The conceptual design of the MHTGR fuel-handling machine and the fuel transfer system is essentially an extension and further development of the Fort St. Vrain design tailored for application to a steel vessel, a different radial core-and-control-assembly (access) arrangement, and a taller core. Spent fuel is stored in dry, helium-filled wells surrounded by water in one of two spent-fuel storage pools; each pool is contained in one of the reactor auxiliary buildings (RABs). Decay heat is 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, with water replacement from a makeup-water supply that is not described. The general arrangement of the fuel handling and storage system is shown in Figure 9.1.

The safety objectives are to avoid exceeding the dose limits of 10 CFR Part 20 by containing fission-product contamination on the fuel elements and by avoiding fuel damage due to either structural challenges or overheating by decay heat.

9.1.2 Scope of Review

The staff has compared the description of the conceptual design given in PSID Section 9.1.1 with the acceptance criteria given in Section 4.3, "Nuclear Design," and Sections 9.1.1 through 9.1.5 of Chapter 9, "Auxiliary Systems," of the Standard Review Plan (SRP) and Regulatory Guide 3.15, "Standard Format and Content of License Application for Storage Only of Unirradiated Power Reactor Fuel and Associated Radioactive Material," Revision 1. The staff review focused on the acceptability of the conceptual design with regard to maintaining fuel-element integrity, fuel-element cooling, and subcriticality. These functions relate to meeting the dose limits of 10 CFR Part 20. No independent calculations were performed.

9.1.3 Review and Design Criteria

The applicable review and design criteria are the acceptance criteria of Sections 9.1.1 through 9.1.5 of the SRP. These sections of the SRP specify acceptance criteria for new-fuel storage, spent-fuel storage, spent-fuel cooling, the handling of light loads, and the handling of heavy loads. Staff resources did not make it possible to perform a detailed assessment of the conceptual design relative to the set of acceptance criteria provided in the SRP. For later review, the design descriptions in the preliminary standard safety analysis report (PSSAR) should be written in a form consistent with the SRP.

The prevention of criticality in stored fuel is required to be obtained by physical systems or processes utilizing geometrically safe configurations. The effectiveness of such physical systems, processes, and configurations in preventing criticality has to be based on the use of demonstrated techniques and validated analysis in accordance with the guidance of Regulatory Guide 3.15, "Standard Format and Content of License Application for Storage Only of Unirradiated Power Reactor Fuel and Associated Radioactive Material," Revision 1, and ANSI/ANS-8.1, "Nuclear Criticality Safety in Operations With Fissionable Materials Outside Reactors."

DOE has made a commitment to meet the intent of General Design Criterion (GDC) 62, "Prevention of criticality in fuel handling and storage," and Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors," and will assess as the MHTGR design progresses the appropriateness of Regulatory Guide 5.1, "Serial Numbering of Fuel Assemblies for Light-Water-Cooled Nuclear Power Reactors." At a later review stage, the staff will discuss with DOE how the fuel handling and storage system will meet GDC 20, "Protection system functions," and Regulatory Guide 3.15.

9.1.4 Research and Development

The fuel handling and storage system is not included in the Regulatory Technology Development Plan (RTDP); however, since the analytical methods used for criticality-safety studies and design need to be validated against experimental data, as prescribed in Regulatory Guide 3.15 and ANSI/ANS-8.1, a commitment to obtain these data should be included in the planned RTDP revision. Additionally, experimental confirmation of the accuracy and uncertainty of projected long-term decay-heat rates of MHTGR fuel should be addressed along with the technology needs for core-heatup evaluation models, as discussed in Section 4.3.

9.1.5 Safety Issue - Detailed Design

The PSSAR should provide a detailed design description and performance analysis of the fuel handling and storage system. A detailed engineering design description is expected that will demonstrate the integrity and leaktightness of the system. Experimental confirmation is required for the analysis of subcriticality and decay-heat loads.

9.1.6 Conclusions

The PSID indicates that the proposed design is to be based on proven technology. The staff concludes that the conceptual design of the MHTGR fuel handling and storage system can be transformed into an acceptable design at a later review stage, but a detailed design description and safety analysis must be documented in the PSSAR, along with experimental confirmation where appropriate.

9.2 Helium Purification System

9.2.1 Design Description and Safety Objectives

The conceptual design of the MHTGR helium purification system (HPS) is essentially an extension and further development of the Fort St. Vrain design tailored for applicability to the MHTGR. The experience with the Fort St. Vrain HPS has been good from the standpoint of both performance and reliability. The HPS is designed to purify a helium side stream from the reactor primary coolant system and to remove both oxidants and radioactive contaminants in drying and purifying the helium. The HPS provides purified helium on a continuous basis to the buffer seals of both the main and shutdown circulators and purifies helium routed to storage during controlled depressurizations.

The HPS equipment for each module is housed mainly in the reactor building and consists of a high-temperature adsorber/filter section for iodine and particulate removal, an oxidizer/cooler and dryers, a low-temperature adsorber (LTA) for noble-gas removal, and a purified-helium recirculator compressor. There is a separate regeneration train for the dryers and the LTA that services 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 is provided to the HPS by the liquid nitrogen system (LNS), with one LNS serving two modules.

The HPS has three safety objectives: (1) to remove oxidants from the primary coolant system and to maintain chemical impurities to less than 10-ppm total oxidants, (2) to 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) to 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 were out of service, manual crossconnect valves would 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. The components of the HPS will be assembled into modular units that can be valved off and maintained and/or replaced during normal plant operation. Some of the components identified to be most critical also have installed spares.

9.2.2 Scope of Review

The staff review focused on the acceptability of the conceptual design with regard to meeting the radionuclide-control design criteria of 10 CFR Part 100 and the dose limits of 10 CFR Part 20 and 10 CFR Part 50, Appendix I. No independent calculations were performed.

Also, the staff has attempted to compare the brief description of the HPS given in Section 9.1.2.7 of the PSID with acceptance criteria for systems that might be considered to be functionally equivalent to LWR systems, as described in appropriate sections of the SRP. The staff believes that SRP Section 9.3.4, "Chemical and Volume Control System (PWR)," is applicable to the MHTGR and that

SRP Section 9.2.2, "Reactor Auxiliary Cooling Water Systems," and SRP Section 6.5.3, "Fission Product Control System and Structures" should be considered for applicability. Furthermore, at a later review stage, clarification is needed as to whether the MHTGR HPS purge and pressurization functions at vessel penetrations also imply component-cooling functions similar to those of the cooling of control rod drive mechanisms at Fort St. Vrain. Additionally, since the MHTGR fuel-particle coatings are viewed by DOE as the primary containment vessels, the MHTGR HPS might also be reviewed against the acceptance criteria of SRP Section 6.5.3, "Fission Product Control Systems and Structures," wherein the MHTGR reactor vessel is viewed as providing secondary containment for which the HPS provides an equivalent containment atmosphere cleanup system.

In terms of safety objectives, DOE proposes that there should be no "10 CFR 100 Design Criteria" for radionuclide control that apply to the HPS; however, adequate HPS effectiveness may be required in the event of less than expected fuel quality to maintain the level of radionuclides circulating with reactor helium below the value used in the safety analysis.

9.2.3 Review and Design Criteria

The brevity and incompleteness of the description of the HPS conceptual design do not make it possible to perform a detailed assessment of the proposed HPS against the set of acceptance criteria provided in the SRP for potentially equivalent functions in LWRs. The design descriptions in the PSSAR should be developed in a form consistent with the functional intent of the SRP, since it applies to the primary coolant cleanup systems and the integrity of the primary coolant boundary.

At a later review stage, it must be determined whether criteria in SRP Section 9.3.4 are to be functionally applicable to the MHTGR HPS, with the major exception of GDC 35 ("Emergency core cooling"). If the purge and pressurization functions of the MHTGR HPS are subsequently identified as having component-cooling functions, acceptance criteria in accordance with SRP Section 9.2.2 may also apply where functionally appropriate. Finally, the functional intent of GDC 41 ("Containment atmosphere cleanup"), GDC 42 ("Inspection of containment atmosphere cleanup systems"), and GDC 43 ("Testing of containment atmosphere cleanup systems") may apply as provided in SRP Section 6.5.3 with regard to containment-atmosphere cleanup by the HPS.

9.2.4 Research and Development

The HPS is not discussed in the Regulatory Technology Development Plan. This review has not indicated that any further development is required.

9.2.5 Safety Issues

A. <u>Role of the Helium Purification System as a Cleanup System</u>

The accident analysis in PSID Chapter 15 indicates that postulated power, flow, and reactivity transients do not significantly challenge the primary fission-product barrier provided by the fuel-particle coating. Other than by quality control in fuel fabrication, however, the MHTGR HPS provides a diverse backup for bad fuel batches, as well as the primary protection against any major fuel degradation due to an inadvertent ingress of chemicals or use of materials that could severely degrade the primary fission-product barrier. The level of safety significance of this backup role needs to be established. The PSSAR should either consider or justify not considering the HPS as a containment atmosphere cleanup system to accommodate unexpected or inadvertent fuel damage.

B. Completion of Design and Analysis

The PSSAR should provide a detailed design description and performance analysis of the HPS, including proposed acceptance criteria. A detailed engineering design description should be provided to demonstrate that all the design and safety objectives can be met.

C. System Ruptures

Results of analyses supplied in response to Comment 9-7 indicated that releases from "worst-case" failures in the HPS following normal operation of the plant would result in doses below the allowable exposure limits both at the site boundary and for allowing operator access to the remote-shutdown area. The response needs to be reconsidered in terms of rapid release from the low-temperature adsorber (LTA), which contains condensed fission-product gases.

D. Safety Classification To Ensure Primary-System Depressurization

In response to Comment 5-44, DOE stated that the primary system could be depressurized if necessary through the HPS during event category III sequences. It is the staff's position that the safety classification of those components of the HPS required for such depressurizations should be classified as safety related.

9.2.6 Conclusions

The PSID indicates that the proposed design is to be based on proven (Fort St. Vrain) technology. The conceptual design of the MHTGR HPS is conditionally acceptable, therefore, but an improved definition of its role and a more detailed design description and safety analysis are required.

9.3 Liquid Nitrogen System

9.3.1 Design Description and Safety Objectives

The conceptual design of the MHTGR liquid nitrogen system (LNS) is essentially an extension and further development of Fort St. Vrain design tailored for application to multiple reactor modules. It will provide liquid nitrogen for use in cooling the low-temperature adsorbers (LTAs) in the helium purification system (HPS) and for use in various instruments in the nuclear steam supply system 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) nitrogen-recondenser compressors. The peak load during the initial stages of a depressurization event can be accommodated without a second recondenser by using the excess storage capacity in the phase separator and storage tank. There are two full-capacity liquid nitrogen pumps, with one serving as a backup. These backup components can be isolated during normal operation for service or replacement.

9.3.2 Scope of Review

The staff review focused on the acceptability of the conceptual design with regard to meeting the radionuclide-control design criteria of 10 CFR Part 100 and the dose limits of 10 CFR Part 20. No independent calculations were performed.

The staff attempted to compare the brief description of the LNS given in Section 9.1.2.3 of the PSID with the acceptance criteria for systems that might be considered to be functionally equivalent to LWR systems, as described in appropriate sections of the SRP. Consideration was given to the acceptance criteria of SRP Section 9.2.2, "Reactor Auxiliary Cooling Water Systems," and SRP Section 9.3.1, "Compressed Air System."

9.3.3 Review and Design Criteria

Design criteria should be developed in a manner functionally equivalent to selected acceptance criteria of Sections 9.2.2 and 9.3.1 of the SRP as they apply to the operation of a cooling system for reactor auxiliary equipment that forms part of the primary coolant system pressure boundary, a cooling system for potentially important instrumentation, and a source of pressurized gas for other equipment that is yet to be fully specified. The brevity and incompleteness of the description of the conceptual design do not make it possible, however, to perform a detailed assessment of the conceptual design against the detailed set of acceptance criteria provided in the SRP for equivalent functions in LWRs. The design descriptions in the PSSAR should be written in a form consistent with the functional intent of the SRP.

DOE states that there are no "10 CFR 100 Design Criteria" for radionuclide control that apply to the MHTGR LNS; however, the LNS function is essential to effecting the radionuclide-control function of the HPS, which, as discussed in Section 9.2, has a 10 CFR Part 100 radionuclide-control function for certain event category III sequences.

9.3.4 Research and Development

The LNS is not discussed in the Regulatory Technology Development Plan, and this review has not indicated that any further development is required.

9.3.5 Safety Issues

A. <u>Completion of Design and Analysis</u>

The PSSAR should provide a detailed design description and performance analysis of the LNS. The integrity and leaktightness of the system should be demonstrated, and more information and analysis should be provided on accident sequences and radionuclide releases to the environment (for example, elaboration of the data given in PSID Table 9.1-10). Additionally, this should include consideration of LNS failure in conjunction with other postulated events, particularly seismic events, missile generation, or vulnerability, or with unanticipated abnormal conditions, such as a bad batch of fuel or chemical/materials interaction with the fuel.

B. Importance to Instrumentation and Dependent Systems

The PSSAR should identify and discuss the importance of the LNS to instrumentation and systems dependent on the LNS, with particular emphasis on HPS-depressurization performance, defense-in-depth considerations regarding moisture monitors in the analytical instrumentation and protection systems, and postaccident monitoring instrumentation requirements with regard to ensuring continued fuel integrity.

9.3.6 Conclusions

The PSID indicates that the proposed design is to be based on proven technology. The conceptual design of the MHTGR LNS is therefore conditionally acceptable, but more detailed design description and safety analysis are required. Based on resolution of the safety issues, it may become necessary to classify the LNS as safety related.

9.4 Reactor Plant Cooling Water System

9.4.1 Design Description and Safety Objectives

The reactor plant cooling water system (RPCWS) is described in PSID Section 9.1.2.4. The RPCWS removes waste heat from the following reactor-plant components: (1) the helium purification system (HPS) coolers and compressors, (2) the HPS regeneration coolers and compressors, (3) the main circulator (MC) motor of the heat transport system (HTS), (4) the moisture-monitor compressor modules, (5) the neutron control assemblies (NCAs), and (6) miscellaneous components. The waste heat is rejected by means of a heat exchanger to the service water system (SWS) described in Section 10.4. 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. A water-chemistry package is included for treatment when required.

During normal plant operation the RPCWS runs with one pump and one heat exchanger, with 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 main control room.

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 failed, or was not available, the plant would have to be shut down.

DOE has stated that the RPCWS does not serve any safety functions. It does, however, maintain operability of the HPS and the safety-related NCAs and also maintains operability of the HTS because of its function of providing motor cooling for the main circulator. Since it does not serve any direct safety functions it is not proposed to be safety related; however, it is stated to be of high reliability to meet the user requirements of high plant availability.

9.4.2 Scope of Review

The review focused on justification for consideration of the RPCWS as non-safety related when it provides cooling for the safety-related NCAs (that is, the control rod drive assemblies). The RPCWS was reviewed against the acceptance criteria of SRP Section 9.2.2, "Reactor Auxiliary Cooling Water System."

9.4.3 Review and Design Criteria

No safety-related criteria were applied to the RPCWS; however, in accordance with SRP Section 9.2.2, General Design Criterion (GDC) 44 ("Cooling water") applies as it relates to the capability to transfer heat loads from safety-related and important-to-safety equipment to a heat sink under normal operating conditions. Similarly, GDC 2 ("Design bases for protection against natural phenomena") would apply under Regulatory Position C.2, Regulatory Guide 1.29.

9.4.4 Research and Development

No development efforts are required for the RPCWS.

9.4.5 Safety Issue - Cooling of Neutron Control Assemblies

Since the RPCWS provides coolant for the safety-related neutron control assemblies (NCAs), their unavailability because of insufficient RPCWS cooling could adversely affect shutdown margins. DOE, however, in response to Comment 4-22 stated that it would take extended periods without cooling to affect these drives, and the control room indication of such conditions would be available to the operators. Furthermore, uncorrected RPCWS failure would more rapidly lead to overheating of the main circulator motor, and HTS failure would cause a reactor trip. Finally, even if the control rods ultimately would not trip, the reserve shutdown control equipment (RSCE) would be available for scram.

In evaluating DOE's position, the staff noted DOE's contention that the rod position instrumentation and indicators function only to monitor conditions in the reactor and can neither cause nor mitigate accidents. DOE's position that instrumentation and indicators are not safety related, however, is weak, since erroneous information could lead to false confidence and undesirable decisions by the operators. Furthermore, the expectation of correct operator response is inconsistent with DOE's claim that the reactor operators do not serve a safety-related function. Finally, to rely on prior failure of another component, the HTS, is a questionable practice, since it would be uncertain when and if the HTS would fail because RPCWS cooling to the HTS might be sufficient. For instance, flow blockage could be local to the coolant paths to the cooling system for the NCAs. The staff will complete its evaluation at a later review stage when further discussion can be held with DOE on this matter.

9.4.6 Conclusions

Most of the RPCWS does not cause safety concerns and can well be designed and operated as planned by DOE. The staff will complete its evaluation of the NCA cooling issue at a later review stage. If the RPCWS is determined to be a support system for a safety-related system, then it will have to be classified as a safety-related system.

9.5 Heating, Ventilation, and Air Conditioning

9.5.1 Design Description and Safety Objectives

The nuclear island (NI) heating, ventilating, and air-conditioning (HVAC) system provides for equipment operability, personnel comfort, and the monitoring and filtering of any potentially radioactive atmospheres.

Once-through, conditioned supply air will be provided for (1) the accessible portion of the reactor building (RB), (2) the reactor auxiliary building (RAB), (3) the reactor service building (RSB), and (4) the personnel services building (PSB). The radioactive waste management building will have a similar air-conditioning system, except that its exhaust will be filtered continuously. The HVAC system for the RB and RAB will have 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 filter trains, which will provide a prefilter, and a high-efficiency particulate air filter (HEPA), 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 and for the cooling water building, 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 hot-water-heated unit heaters. 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 not 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 would be provided during shutdown when access was 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 be provided with separate unit coolers using chilled water. These unit coolers will also control the relative humidity at less than 50 percent. The NI HVAC system will be controlled from the main control room, with local control being 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.

Since a failure of the HVAC system is not expected to cause any 10 CFR Part 100-related radioactive releases, it is considered to be non-safety related. For routine operation the filtering system of the HVAC system will be designed to meet the routine offsite release limits of 10 CFR Part 50, Appendix I, and the occupational doses of 10 CFR Part 20. In general, loss of the HVAC system would because for an orderly shutdown of a reactor module or of the plant, depending on the degree of failure. DOE states that none of the safety-related equipment relies on HVAC to perform its safety function.

9.5.2 Scope of Review

The review focused on the potential impact on safety if failures of portions or all of the HVAC system could become precursors or contributors to events leading to significant radionuclide release. The HVAC system is described in Section 9.2.11 of the PSID.

9.5.3 Review and Design Criteria

No safety-related design criteria have been applied to the HVAC system. The system will be designed, however, to meet criteria for routine emissions (10 CFR Part 50, Appendix I) and occupational exposure (10 CFR Part 20), as well as to meet user requirements. Specific resource allocations to meet applicable criteria will be provided as the design progresses. The radiation levels in all accessible areas are to remain below 1 mrem per hour to permit at least 40 hours of access per week. At a later review stage, the then-current LWR criteria for the HVAC system will be reviewed for MHTGR applicability.

9.5.4 Research and Development

No HVAC-related items are currently included in the Regulatory Technology Development Plan, nor do there appear to be any areas that should require further development efforts.

9.5.5 Safety Issue - Precursor of or Contributor to Events Leading to Significant Radionuclide Release

The staff agrees with the DOE position that the safety features of the MHTGR, particularly the passive RCCS as the final decay heat removal system, result in the assessment that failure of the HVAC system cannot reasonably be considered a precursor of or a contributor to events leading to significant release of radionuclides. This assessment should be confirmed, however, at a later review stage when more details of the design and role of the HVAC system are expected to be available. For example, the HVAC system may be required to meet GDC 3 ("Fire protection") to ensure that non-safety-related systems do not prevent functioning of safety-related systems.

9.6 Fire Protection

9.6.1 Design Description and Safety Objectives

The design of the plant fire protection system (PFPS) provides those features needed to rapidly detect, control, and suppress fires, including automatic detection systems, manual fire-hose stations, and portable fire extinguishers, as well as automatic water, carbon dioxide, and Halon subsystems. The PFPS interfaces directly with the nuclear island (NI) and protects the systems, structures, and components required to protect the public health and safety. The PFPS has backup, independent, motive power that will be available in the event of abnormal operating occurrences, including loss of all ac power.

Plant Fire Protection Water Subsystem

The plant fire protection water subsystem (PFPWS) consists of two fire pumps and controllers. The primary pump is electrically driven; the backup pump is diesel driven. The diesel-driven pump has a battery-powered starting system and gravity fuel-oil feed from an 8-hour-day tank. Two fire-water storage tanks, each of 300,000-gallon capacity, feed the pumps. Each pump is separately connected to an underground fire-water loop that encircles the NI and supplies water to the yard hydrant and hose house system and several fire protection piping systems within the plant's structure. Isolation valves allow for maintenance without interrupting existing protection.

The underground fire-water loop interfaces with and supplies water to the NI. The NI portion consists of yard fire hydrants, water spray, deluge and wet-pipe sprinkler systems, and wet-standpipe fire-hose stations. Standpipes and hose stations for safety-related buildings are connected to the yard main, independent of the connection to the non-safety-related, fixed water suppression system serving the same fire area.

Plant Fire Protection Carbon Dioxide Subsystem

A total-flooding plant fire protection carbon dioxide subsystem (PFPCDS), designed for double-shot capability, delivers carbon dioxide to the turbine-generator building and enclosure areas. This subsystem consists of a low-pressure, refrigerated carbon dioxide-storage tank, piping, nozzles, and controls for master and selection valves, as well as detection and audio alarms. Carbon dioxide is not used on the NI.

Plant Fire Protection Halon Subsystem

The plant fire protection Halon subsystem (PFPHS) is designed for double-shot discharge capability and protects electrical panel areas and local control rooms in the operations center, reactor building, and buildings and structures in the energy conversion area. The subsystem consists of dedicated main and connected reserve-cylinder banks, manifold piping, and applicator nozzles, as well as detection and audio alarms.

Plant Fire Detection and Alarm System

The plant fire detection and alarm system (PFDAS) will be available to detect and annunciate the presence and location of fire and/or combustion byproducts. The detection/alarm system will not be interfered with or affected by any other system. The detection/alarm system will be used in and around all systems, structures, and components required for the protection of the health and safety of the operating staff. Other areas that will be protected by the PFDAS include those in which radioactive materials will be handled in the reactor building, the reactor auxiliary building, the reactor service building, and the personnel services building. The non-Class IE uninterruptible power supply will permit the fire detection and alarm system to be operational during loss of all ac power.

9.6.2 Scope of Review

Sections 7.4.5, 9.1.3.1, and 9.2.4 of the PSID were discussed with DOE and reviewed within the framework of GDC 3, "Fire protection"; Branch Technical

Position (BTP) APCSB 9.5-1, "Guidelines for Fire Protection for Nuclear Power Plants" (NUREG-0800); and SRP Section 9.5.1. The conceptual nature of the design would not allow a detailed review at the compartment or component level. Additional reviews will be needed as the design evolves, especially with regard to the design and arrangement of redundant safety trains. Graphite fires were not considered in this section (see Section 15.2.6.2). The nature of the reactor cavity fire suppression system has not been determined.

DOE has stated that the LWR fire-protection programs are not applicable to the MHTGR and that the plant fire protection and fire detection and alarm systems are not safety related. Justification for regulatory deviation is based on DOE's position that a fire cannot cause loss of control of chemical attack, loss of the reactor-cavity cooling, or loss of function to control heat generation and that the systems do not perform any 10 CFR Part 100-related radionuclide-control functions. The NI portion will, however, meet the intent of GDC 3.

The primary focus of GDC 3 is the probability of fires and explosions and their effect on plant systems. Following the fire at the Browns Ferry plant, the NRC staff issued specific guidance for implementation of GDC 3 in BTP APCSB 9.5-1 (May 1, 1976).

9.6.3 Review and Design Criteria

Fire-protection criteria for advanced plants are currently under review with the general provisions currently contained in 10 CFR 50.48. It is expected that the PFPS will be reviewed under the criteria current at the time of the PSSAR review.

9.6.4 Research and Development

The staff finds at this stage of review that no special or unique fire-protection research and development will be needed for the MHTGR.

9.6.5 Safety Issues

A. Conformance With BTP APCSB 9.5-1 and SRP Section 9.5.1

Both BTP APCSB 9.5-1 and SRP Section 9.5.1 have a defense-in-depth philosophy not otherwise found in the broad GDC 3 guidelines. Based on past LWR experience, meeting the requirements of GDC 3 alone will not provide adequate fire protection for the MHTGR. From a defense-in-depth perspective, DOE's justification for deviation from regulatory criteria must show equivalence in the level of protection. The design has not matured to the point where equivalence can be shown.

B. Backup Fire-Suppression Capability

Backup fire-suppression capability is defined as fire-hose stations, portable fire extinguishers, and yard hydrants. To justify adequate backup fire-suppression capability, however, DOE must make an appropriate commitment to manual fire-fighting procedures and training.

C. Inadvertent System Actuation

Based on past LWR experience, emphasis on automatic rather than manual fire suppression raises special concerns with regard to inadvertent actuation. At a later review stage, DOE should address the potential impact of the inadvertent actuation of the fire suppression system on safety systems and components. In addition to the consequences, the mechanisms by which actuation is initiated (for example, excessive room-temperature smoke, steam, dust, and maintenance activity) should be considered.

D. System Interactions

The MHTGR conceptual design does not provide for investigation of fire-induced system interactions. As the design matures, the effects of fire-induced system interactions on multiple module control and shutdown systems will have to be evaluated.

E. Shutdown During Fire

The need for and role of remote shutdown during a fire will be determined at a later review stage.

F. Quality Assurance

A quality assurance (QA) program for fire protection should be part of the overall plant QA program. Specific criteria such as those in BTP APCSB 9.5-1 and SRP Section 9.5.1 should be met.

9.6.6 Conclusions

Meeting the intent of GDC 3 alone will not provide adequate fire protection for the MHTGR. The MHTGR should meet 10 CFR 50.48 requirements and show at the PSSAR review stage appropriate conformance with the fire-protection criteria current at that time. Equivalence to BTP APCSB 9.5-1 cannot be shown at this time because of the conceptual nature of the design.

The adequacy of the design with regard to fire and fire-mitigating activity, including inadvertent actuation, will play a substantial role in subsequent evaluations of the fire protection system. Through proper engineering design, safety-system separation, redundancy, and protection, a large portion of the BTP APCSB 9.5-1 criteria can be met.

A detailed fire-hazard analysis will help identify potential fire hazards and their effect on plant systems. A probabilistic risk assessment pertaining to fire should be performed early in the MHTGR design stage and revised periodically as the design progresses. In addition, specific QA criteria should be met during design and construction of the fire protection system.

The potential impact of fire-induced system interactions on the plant will have to be evaluated. The need for and adequacy of remote shutdown during postulated fires involving multiple trains of safety equipment (for example, control room fires) will also have to be determined.

9.7 Other Service Systems

Chapter 9 of the PSID includes descriptions of many other service systems that were not included in the staff's review. The principal systems not reviewed at this time are (1) portions of the fuel handling and storage system not included in the Section 9.1 review, (2) the reactor service equipment subsystems, (3) the hot service facility subsystem, (4) the helium storage and transfer subsystem, (5) the decontamination service subsystem, (6) portions of the mechanical service system not included in the HVAC review in Section 9.5, and (7) systems in the energy-conversion area, except as included in the fire-protection review in Section 9.6. At the PSSAR review stage, the staff intends to review all these subsystems using, as appropriate, the guidance provided in the LWR SRP. During the course of this review, it is anticipated that DOE will provide acceptable justification for exceptions to the LWR criteria and propose acceptable new or alternative criteria as conditions warrant.

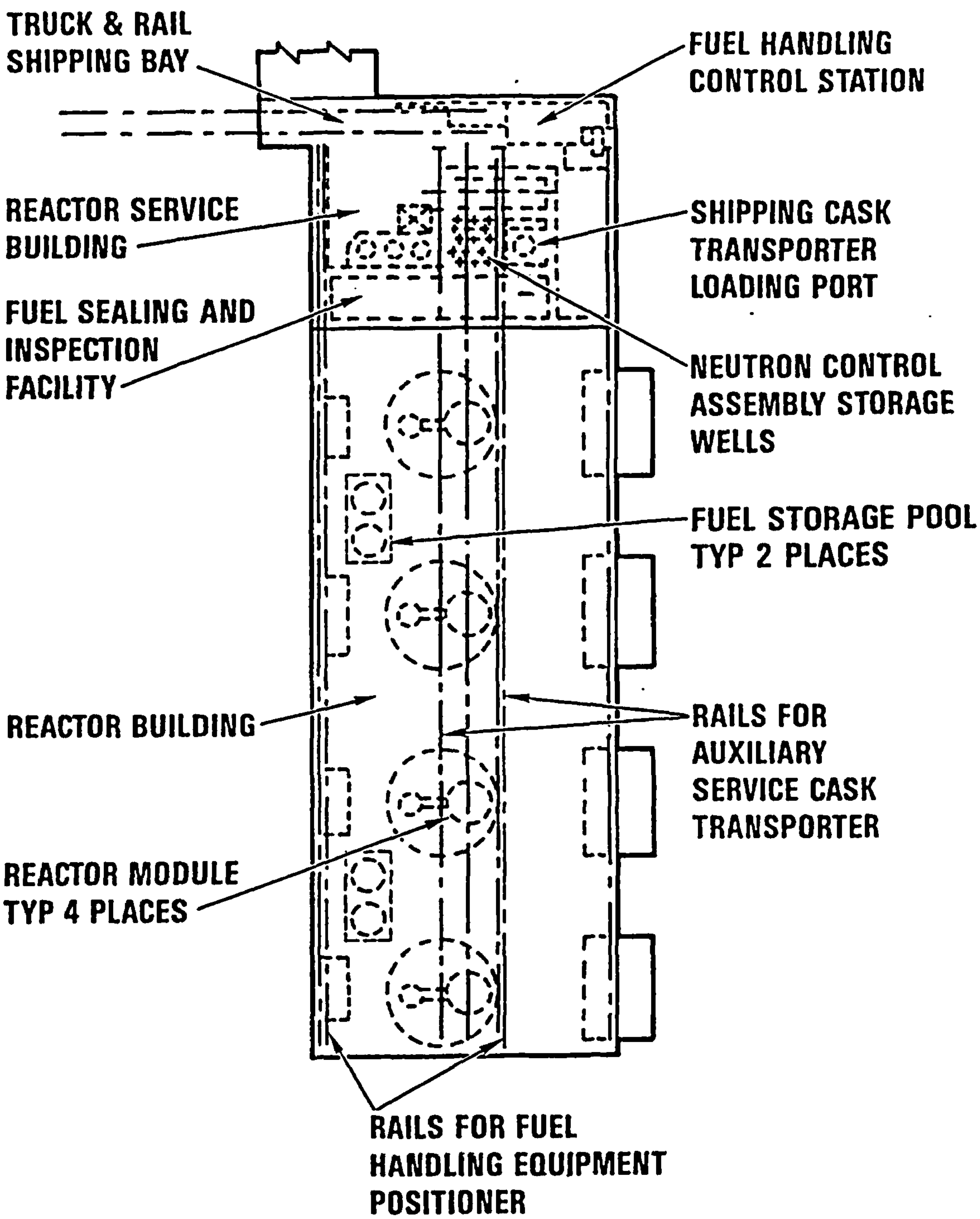

Figure 9.1 General arrangement of site fuel handling system
Source: DOE, 1986-3

10 STEAM AND ENERGY CONVERSION SYSTEMS

The systems reviewed in this chapter were selected from Chapter 10, "Steam and Energy Conversion Systems," of the Preliminary Safety Information Document (PSID) on the basis that their safety impact will be greater than that of other systems described in Chapter 10 of the PSID. At the time of the preliminary standard safety analysis report (PSSAR) review, all the systems will be reviewed to establish a safety level equivalent to that of similar light-water-reactor (LWR) systems.

10.1 Main Steam and Feedwater Supply Systems

10.1.1 Description and Safety Objectives

The main steam supply system (MSSS) is a piping system with the primary function of conveying superheated steam to the turbine-generators. Isolation valves are included in the MSSS design so that any one of the four steam generators in a four-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 MSSS 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 (FWS) system is a piping system that supplies water from the condenser to the economizer inlet of the steam generators. Condensate from the condenser is normally pumped first through a polishing demineralizer to adjust water chemistry and then through the feedwater heaters to the deaerator. The feedwater is then pumped at high pressure to the steam generators. Valves are included to isolate the steam generators and to connect the FWS system to the turbine bypass desuperheater and the steam and water dump tank.

Neither of these systems has a direct safety-related reactor-cooling function, since the reactor cavity cooling system (RCCS) is designed to act as the ultimate heat sink and is completely independent of the MSSS and the FWS system. Failures of these two systems must not, however, interfere with safety functions of the MHTGR operations.

10.1.2 Scope of Review

The scope of the staff review of the MSSS and the FWS system included only the effects of failures on the operability of plant safety systems. The safety related isolation valves on the main-steam and feedwater lines for each steam generator are considered to be a portion of the vessel system and are discussed in Section 5.3.

10.1.3 Review, Design, and Inspection Criteria

The physical integrity of the MSSS and the FWS system is important for normal operation. Since failure of these two systems has been considered in the safety design bases for the reactor building and its contents (see Section 6.2), these two systems need only be designed to meet appropriate industrial standards. General Design Criterion (GDC) 4, "Environmental and dynamic effects design basis," and 10 CFR 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants," are to be considered for the safety-related systems that could be affected by failures of the MSSS and the FWS system.

10.1.4 Research and Development

The Regulatory and Technology Development Plan (RTDP) does not identify any research and development programs for these systems. The staff agrees that safety research in this area is not needed. For adequate functioning, however, the MSSS and the FWS system will have to meet the objectives of a staff-approved preoperational and startup test program to be proposed at a later review stage.

10.1.5 Safety Issues

A. Power-Generation Stability

Although the power generation systems are not reviewed here as a whole, they are important contributors to the defense-in-depth aspects of core cooling. The remaining power generation systems are designed not to trip on power-system-component signals, such as single reactor module or turbine-generator sets. Further, the power generation system is not expected to trip on reactor trip or loss of offsite power. Electric power for uninterrupted forced cooling of the reactor is thus dependent on power-generation stability, which experience has shown is unreliable. Although provisions exist for alternate ac power and cooling by the RCCS, if needed, it is apparent that the power generation systems have an important role in providing defense-in-depth and the avoidance of RCCS use. Figure 10.8.3 of the PSID shows that for each turbine-generator unit, the bypass flow to the condenser is estimated at 50 percent of design flow.

B. Equipment Qualification Concerns for High-Energy-Line Breaks

The MSSS and the FWS system are potential sources of missiles, high-energy-fluid jets, pipe whip, and environmental conditions that could damage safety-related electrical and other equipment, and, therefore, these factors must be accounted for in the design of safety-related components. It should be noted that steam temperatures and pressures are higher than those for LWRs and that in regard to MHTGR conformance with GDC 4 and 10 CFR 50.49, this fact must be taken into account.

10.1.6 Conclusions

Unlike LWRs, where the MSSS and the FWS system provide essential cooling of the reactor during safety-related operations, the MSSS and the FWS system serve no direct safety functions. The effects of failures in these two systems must, however, be considered in terms of the consequences to safety-related equipment, taking into account that steam temperatures and pressures for the MHTGR are higher than those for LWRs. Conclusions on the overall acceptability of the MSSS and the FWS system, including consideration of the necessary amount of steam bypass flow to achieve power-generation stability, are deferred to a later review stage.

10.2 Startup and Shutdown Subsystem

10.2.1 Description and Safety Objectives

The startup and shutdown (SU/SD) subsystem is 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 is to provide for smooth operational transition for a module in the 0- to 25-percent (or 25- to 0-percent) power range. It is sized and designed to operate for a single module SU/SD 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 are used in conjunction with the SU/SD subsystem. The SU/SD subsystem is designed to deliver feedwater to the steam generator and steam to the turbine at the desired temperature, pressure, and flow, and within prescribed water-chemistry limits. The SU/SD subsystem does not perform any safety-related functions and is therefore not classified as safety related. In case of failure of part or all of the SU/SD subsystem, cooldown could be achieved by various other non-safety-related systems or by the safety-grade reactor cavity cooling system (RCCS). It must be assured, however, that failures of the SU/SD subsystem will not interfere with the effective functioning of safety systems.

10.2.2 Scope of Review

The staff reviewed Section 10.15 of the PSID and DOE's responses to Comments 10-5 and 10-6 to identify safety issues and the potential of the design to meet the appropriate criteria. No independent calculations were performed by the staff or its contractors.

10.2.3 Review, Design, and Inspection Criteria

The SU/SD subsystem supports non-safety-related functions during normal operating transients. Based on the staff's review of the acceptance criteria of the Standard Review Plan (SRP), only GDC 4 ("Environmental and dynamic effects design bases") would apply, as it relates to dynamic effects associated with flow instabilities and loads that are historically of concern in SU/SD situations and to failures resulting in missiles or adverse environmental conditions that could damage safety-related systems or components.

10.2.4 Research and Development

No work is proposed for the SU/SD subsystem in the RTDP, and none appears to be necessary. The functioning of the SU/SD subsystem will have to meet the objectives of a staff-approved preoperational and startup test program to be proposed at a later design stage.

10.2.5 Safety Issues

A. Consequences of Failure

The SU/SD subsystem should be designed so that any failures or misoperation would not result in damage to or interference with safety-related systems. It should be taken into consideration that the steam temperatures and pressures are higher than those for LWRs.

B. Thermal Shock

In response to Comment 10-6 on the effects of thermal shocks induced by SU/SD on the steam generator, DOE stated that this is a subject to be discussed further as the design progresses.

10.2.6 Conclusions

The conceptual design of the SU/SD subsystem is acceptable as non-safety related. Further analyses of its operational failure modes and integration into the power cycle will be needed in later stages of the design to support this conclusion.

10.3 Steam and Water Dump System

10.3.1 Design Description and Safety Objectives

The steam and water dump system (SWDS) is provided to contain the inventory of a steam generator in the event of a steam generator tube leak. Since the normal operating pressure of the secondary coolant is greater than that of the primary, a tube leak provides a path for steam 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 is actuated by non-safety-related portions of the plant protection and instrumentation system (PPIS) when a high level of moisture is detected in the primary coolant. The SWDS must be designed to contain the mass-and-energy inventory of the steam generator, as well as any primary coolant that leaks into the SWDS through a tube rupture. Since the primary coolant will have circulating activity, there is piping connecting the SWDS with the gaseous and liquid radioactive waste system (GLRWS) to ensure that no primary coolant is released directly to the environment.

The SWDS serves each steam generator module independently. The portion of the subsystem associated with each steam generator consists of a dump tank, two trains of dump valves, a drain pump, and piping and valves for interconnecting with the GLRWS. The steam generator is isolated by two power-operated valves mounted in series on each inlet and outlet of the steam generator. Dumping is executed by two parallel trains of dump lines, each equipped with two dual-actuated motor-operated valves mounted in series. The subsystem's dump and isolation valves are described as being powered from a reliable, but non-safety-related, power source.

The following instrumentation is to be provided for each of the four subsystem loops at the system-control station in the reactor buildings and in the main control room: (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.

Further, the SWDS, in conjunction with steam and feedwater isolation, also serves to limit the amount of positive reactivity that can be inserted by water ingress. Features of the SWDS's subsystems are described below.

Dump Tank

The carbon-steel dump tank is designed to contain the mass-and-energy inventory of the steam generator in its loop and must be sized accordingly. It is 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 steam generator inventory is 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 are motor operated when called upon by the PPIS and 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 are closed after the steam generator has emptied its inventory.

Drain Pump and Connecting Piping

The drain pump receives the liquid from the bottom of the dump tank and pumps it through connecting piping to the gaseous and liquid radioactive waste system (GLRWS). The connecting piping for the gases in the dump tank leads directly to the GLRWS.

The SWDS has important objectives during a steam-generator-leak transient. It limits the amount of chemical damage to the nuclear core during water ingress. Also, during a postulated tube leak, the valves and piping of the SWDS may carry radioactivity from the primary coolant and may act as a pressure boundary for the containment primary coolant. This containment function, however, is not proposed by DOE to be safety related, in keeping with the containment function assigned to the fuel particles and the radionuclide-retention function of the reactor building. Furthermore, the SWDS is not needed to ensure core cooling or to control radionuclides during event category II and III sequences.

10.3.2 Scope of Review

The staff reviewed the safety analysis presented in Section 10.16 of the PSID and focused on DOE's proposal not to require the SWDS to be safety related. No independent calculations were performed by the staff or its contractors on the performance of the components described in this section.

10.3.3 Review and Design Criteria

The staff contractor, Oak Ridge National Laboratory, surveyed potential criteria on the basis of comparisons of subsystem functional requirements with the acceptance criteria used in SRP Sections 5.2.5, "Reactor Coolant Pressure Boundary Leakage Detection," 5.4.6, "Reactor Core Isolation Cooling Systems [BWR]," 5.4.11, "Pressurizer Relief Tank," and 10.4.8, "Steam Generator Blowdown System [PWR]." The following general design criteria were identified as providing guidance for the review and design of the SWDS: GDC 1 ("Quality standards and records"), GDC 2 ("Design Bases for protection against natural phenomena"), GDC 4 ("Environmental and dynamic effects design bases"), GDC 5 ("Sharing of structures, systems and components"), GDC 27 ("Combined reactivity control systems capability"), GDC 29 ("Protection against anticipated operational occurrences"), and GDC 30 ("Quality of reactor coolant pressure boundary").

10.3.4 Research and Development

The Regulatory and Technology Development Plan does not identify any research and development programs for these systems. The staff agrees that safety research in this area is not needed. The adequate functioning of the MSSS and the FWS system will, however, have to meet the objectives of a staff-approved preoperational and startup test program to be proposed at a later review stage.

10.3.5 Safety Issues

A. Safety Classification

Although the staff will not require that all portions of the SWDS meet safety-related requirements, commitments must be proposed at a later design stage to ensure high-quality construction and availability when needed. These commitments should include the design specifications and standards that will be used and an approved program of preoperational and startup testing and technical specifications or equivalent administrative controls pertaining to inspection, maintenance, and out-of-service time limits.

B. Capacity

Although the capacity of the dump tank will be sufficient to hold the contents of a steam generator, the piping and valve sizing and capacity must be defined and justified in terms of a flow rate corresponding to an acceptable number of postulated steam generator tube failures.

C. Structural Failure of the Steam and Water Dump System

In the event of structural failure of the SWDS components, such as during a seismic event, the postulated release of radionuclides resident in the SWDS to the reactor building needs to be evaluated. The failure of any structural component of the SWDS will have to be evaluated in terms of GDC 4 and 10 CFR 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants."

D. Potential Damage to the Gaseous and Liquid Radioactive Waste System

An assessment should be made of whether the failure of the SWDS valves to close after a steam generator dump could cause an overpressurization or a significant radionuclide overload of the GLRWS. Further, the potential for significant damage to the GLRWS should be assessed for the case in which the level of cool water in the dump tank would be insufficient to absorb the energy inventory of the steam generator.

10.3.6 Conclusions

Based on the PSID analysis and the Probabilistic Risk Assessment (PRA), the staff concludes that the SWDS does not need to be classified as a safety-related system in terms of its need to serve a containment function, to cool the core, or to control radionuclides during event category II and III sequences. The failure of the SWDS to function as designed will, however, be reviewed at a later design stage to ensure that it will not have an impact on safety-related components. Also at that time commitments to achieve a satisfactory level of quality and availability will be reviewed.

10.4 Service Water System

10.4.1 Design Description and Safety Objectives

The functions of the service water system (SWS) are to remove waste heat from non-safety-related process systems located in various buildings on the nuclear island and to 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 used in the power-generation processes. It originates 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.

10.4.2 Scope of Review

The staff performed a review that focused on the consequences of SWS failure in regard to the defense-in-depth aspects of residual-heat removal from the reactor.

10.4.3 Review and Design Criteria

Based on staff review in relation to the acceptance criteria of SRP Section 9.2.1, "Station Service Water System," the reference LWR design criteria are GDC 2 ("Design bases for protection against natural phenomena") and GDC 4 ("Environmental and dynamic effects design bases"). These apply only as they relate to the assurance that potential failures do not adversely affect the operation of safety-related structures, systems, and components.

10.4.4 Research and Development

The Regulatory and Technology Development Plan does not identify any research and development program for these systems. The staff agrees that safety research in this area is not needed. The adequate functioning of the SWS will, however, have to meet the objectives of a staff-approved preoperational and startup test program to be proposed at a later review stage.

10.4.5 Safety Issue - Safety Classification

If at the preliminary or final design stage, safety analyses indicate that the reliability of the shutdown cooling system must be upgraded from non-safety related to safety related in order to meet risk goals, the design of the service water system will have to be upgraded to meet the acceptance criteria of SRP Section 9.2.1.

10.4.6 Conclusions

The conceptual design of the service water system is acceptable as non-safety related. The safety analyses for the preliminary and final designs of the MHTGR and refined PRAs need to provide sufficient bases to support this conclusion.

11 OPERATIONAL RADIONUCLIDE CONTROL

Chapter 11, "Operational Radionuclide Control," of the Preliminary Safety Information Document (PSID) contains seven sections, but only Section 11.1, "Radionuclide Design Criteria," is judged to be important at this conceptual design review stage. This section relates to information on fuel performance presented in Section 4.2, "Fuel Design," and to Section 15.5, "Siting-Source-Term Selection and Use," and is reviewed herein. The PSID material on liquid, gaseous, and solid radioactive wastes, together with the dose assessment of radionuclides discharged during normal operation to the environment (PSID Sections 11.2, 11.3, 11.4, and 11.7, respectively), will not be evaluated at this review stage because this material will not have a significant bearing on the MHTGR's reactor safety issues. Any safety issues that might emerge from review of these sections are expected to be resolvable by established means using available criteria. The PSID material on plant normal operation (PSID Section 11.6) is reviewed elsewhere in this SER (see especially Chapter 10 and Section 13.2), and the material on anticipated operational occurrences is considered at this review stage to be bounded by event category II and III sequences.

11.1 Radionuclide Design Criteria

11.1.1 Description and Safety Objectives

The radionuclide design criteria are stated by DOE to be the allowable levels of radionuclide accumulation in the primary-coolant circuit. This is essentially the inventory of radionuclides in the mechanistic siting source term (SST) described in Section 15.5. These criteria are defined as the circulating and plateout radioactivity in curies after 40 years of operation for each identifiable isotope and are stated in terms of the initial value and the values after 1 and 10 days of decay. These activities were derived by calculations that worked backward from the desired goal of meeting the protective action guidelines (PAGs) at the exclusion area boundary (EAB) for various postulated events. For these "back-calculations," assumptions were made concerning the following phenomena: (1) fission-product deposition and holdup in the reactor-building pathway, (2) deposition and holdup within the reactor vessel and in graphite, and (3) the fractional liftoff of radionuclides plated out (the liftoff fraction must be based on such factors as local flow velocities caused by depressurization events and augmentation by washoff from steam/water-ingress effects or by evaporation from primary-system surfaces as the temperatures of these surfaces become elevated), and (4) the nature and location of the plated-out radionuclides (including chemical and physical bonding to various surfaces, effects of reactor operating history, helium-purity levels, and the overall effect of 40 years of normal operation).

The radionuclide design criteria are used to determine the fuel-integrity requirements, mainly with respect to fuel-particle manufacturing quality standards. These could also affect the capacity requirements for the helium purification system. The factors affecting fuel integrity are discussed extensively in Section 4.2.

The radionuclide criteria for the circulating and plated-out radioactivities are presented in terms of both "design" and "maximum expected" values in PSID Tables 11-2 and 11-3, respectively. The "design" criteria are defined in the PSID as those maximum radioactivity levels that are allowable in the primary system to enable the plant to meet the most restrictive site-boundary criteria (that is, the PAG doses.) The "design" criteria set the fuel-integrity requirements for conservative (95-percent confidence) treatment of the postulated sequences in event category II. The "maximum expected" criteria are obtained by dividing the "design" criteria by factors of 4 or 10 for gases or metals, respectively. DOE uses these quantities as "uncertainty" factors to recognize its belief that current experience and expectations of successful research will justify the use of less restrictive assumptions than are currently used in the "back-calculations" and to illustrate the degree of margin with respect to its proposed source term. At present DOE states that the "maximum expected" criteria are to be used for dose estimates at the 50-percent confidence level. Such use would be consistent with the very rare event category III sequences. While the circulating radioactivity is included in the dose calculations, the inventory of radionuclides in the helium is about three orders of magnitude lower than the plateout inventory, and thus, the calculated dose is dominated by plateout and liftoff estimates.

11.1.2 Scope of Review

The review focused on the "back-calculation" method of determining the radionuclide inventory for the SST and the means to support the assumptions critical to the method's objectives. This review is related to material presented in Section 4.2, "Fuel Design," Section 15.2, "Description of Accidents Considered," Section 15.5, "Siting-Source-Term Selection and Use," and Figure 15.2, "Computer Codes Used in MHTGR Safety Analysis."

11.1.3 Review and Design Criteria

The setting of fuel-integrity standards by back-calculations from a desired goal of accident dose has no precedent in terms of light-water-reactor (LWR) criteria, nor has DOE proposed any specific criteria to be followed in this calculation or for the treatment of its uncertainties. The staff has requested DOE to propose such criteria at a later review stage and plans to review those criteria and possibly develop its own on the basis of precedents for the analysis and use of fission-product-transport data being developed in the course of LWR severe-accident studies.

11.1.4 Research and Development

Section 6 of the Regulatory Technology Development Plan (RTDP) describes the following technology development needs (TDNs) pertaining to the assumptions to be used in the back calculations:

6-1 Fission Product Transport in Reactor Building During Core Conduction Cooldown Transients

6-2 Fission Product Deposition Characteristics for Structural Metals

6-3 Fission Product Reentrainment Characteristics for Structural Metals

6-4 Fission Product Washoff Characteristics for Structural Metals

6-5 Effect of Dust on Fission Product Transport

6-6 Validation of Design Methods for Plate-out Distribution

6-7 Validation of Design Methods for FP [Fission-Product] Liftoff

6-8 Validation of Design Methods for FP Washoff

6-9 FP Diffusivities/Sorptivities in Graphite

6-10 Validation of Design Methods for Fission Gas Release

6-11 Validation of Design Methods for Fission Metal Release

6-12 Fuel Irradiation Proof Test

6-13 Fuel Compact Process Development

Although the listed TDNs were not reviewed in detail by the staff, they appear to be adequate to address the assumptions regarding the phenomena identified in Section 11.1.1. If the confirmation of any of these identified phenomena is not contained in the TDNs, a program for analyzing such assumptions should be added to the RTDP or plant-testing programs, as appropriate. At a later review stage, the staff will review progress on the TDNs and assess whether any changes in their scope, objectives, or experimental procedures may be necessary.

11.1.5 Safety Issues

A. Assumptions Used in Back-Calculations of the Radionuclide Design Criteria

DOE correctly recognized the need for substantial research to confirm the assumptions used to perform the back-calculations. DOE is optimistic that the research will confirm that the assumptions are conservative and that the "uncertainty factors" used to estimate the "maximum expected" radionuclide inventory will be supported. It must be recognized, however, that very few data exist to support most of these assumptions, and the success of the research program is essential to support the containment concept of the MHTGR.

B. Model for Back-Calculations

DOE has not presented for staff review a detailed model for the back-calculations or discussed quantitatively its uncertainties. This model should be presented as soon as practicable, although it may be necessary to include early research results to justify the model to be presented.

C. Design Basis for the Helium Purification System

In the description of the helium purification system (HPS) in Section 9.2.1, the staff did not include in its design basis the consideration that a duty of the HPS could be to maintain sufficiently low levels of circulating radionuclides in the primary system to ensure that plateout quantities will not exceed the assumptions used in the back-calculations. DOE should discuss this concern at a later review stage.

11.1.6 Conclusions*

Substantiation of the back-calculation method for establishing the siting source term and fuel-quality standards is essential in accepting the proposed containment design and emergency-planning actions for the MHTGR. DOE recognizes this by the research program described in the RDTP. As soon as practicable, DOE should provide for staff review a detailed model that includes a quantitative assessment of uncertainties. To justify the model presented, it may be necessary to include early research results. DOE should also propose review and design criteria for the model and its uncertainties.

*Indicates statement is particularly sensitive to change by evaluation of forthcomming DOE information.

12 OCCUPATIONAL RADIATION PROTECTION

Chapter 12 of the Preliminary Safety Information Document (PSID) provides information on radiation protection methods and on estimated occupational radiation exposures to operating and construction personnel during normal plant operation and anticipated operational occurrences (AOOs).

The radiation protection measures for the standard MHTGR are intended to ensure that internal and external occupational radiation exposure to plant operating personnel, contractors, administrators, visitors, and the general population as a result of station conditions, including AOOs, will be within the applicable limits of the top-level regulatory criteria and will be as low as is reasonably achievable.

The basis for the staff's acceptance of the material reviewed is that doses to personnel will be maintained within the applicable limits of the top-level regulatory criteria, which incorporate 10 CFR Part 20, "Standards for Protection Against Radiation." The MHTGR's radiation protection design and program features must also be consistent with Regulatory Guide 8.8, "Information Relevant to Ensuring That Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable" (Rev. 3).

The focus of this SER chapter differs from that of earlier chapters in that its objective is to provide guidance for the development of the preliminary standard safety analysis report (PSSAR) rather than to identify specific safety issues for conceptual design resolution.

12.1 <u>Occupational Radiation Exposure</u>

The staff's review focused on the general policy, design, and operational considerations for maintaining personnel exposure within the limits specified by the top-level regulatory criteria.

12.1.1 Policy Considerations

The general management policy for control of occupational radiation exposure at the MHTGR includes (1) minimizing the number of areas inaccessible because of high radiation levels during reactor operation, (2) selecting materials to minimize the production of radioactive materials, and (3) maintaining the annual average integrated dose to station personnel at less than 10 percent of 10 CFR Part 20 limits. This is consistent with the guidelines of Regulatory Guide 8.8.

In the PSSAR, DOE should include a description of the applicable responsibilities and the related activities to be conducted by the individuals having responsibility for radiation protection.

12.1.2 Design Considerations

The objectives of the radiation protection design are to minimize the necessity for and amount of personnel time spent in radiation areas and to minimize radiation levels in routinely occupied plant areas and in the vicinity of plant equipment requiring personnel attention. The design considerations for the MHTGR include plant layout and equipment design and location for the purpose of ensuring that occupational radiation exposures are within the limits set by the top-level regulatory criteria.

Some of the design considerations used to meet the plant objectives include modularization of radioactive components for ease of disassembly and removal to lower radiation areas for repair, redundancy of equipment, utilization of remote-viewing devices, location of equipment in low-radiation areas, separation of high-radiation sources and occupied areas, use of shielding around radiation sources, and provisions for venting, purging, and decontamination to reduce radiation levels in systems that may experience plateout. These design considerations conform to the guidelines of Regulatory Guide 8.8 and are acceptable.

12.1.3 Operational Considerations

DOE's operational considerations include the development and implementation of plant operating plans and procedures for radiation protection and exposure control, as discussed in Regulatory Guides 8.8 and 8.10, "Personnel Selection and Training" (Rev. 1). These operating plans and procedures will cover system operation, maintenance, surveillance, testing, fuel handling, emergencies, and administration, and will be prepared as the design proceeds. In the PSSAR, DOE should describe the methods used to develop these plans and procedures for ensuring that occupational radiation exposures are as low as is reasonably achievable (ALARA). It should also describe how such planning has incorporated information from operating-plant experience, other designs, etc. The information on occupational radiation protection contained in Section 12.1 of the PSID is acceptable for this stage of the design.

12.2 Occupational Radiation Sources

The sources of contained and airborne radioactivity used as inputs for the dose assessment and for the shielding and ventilation designs are described below, as well as the assumptions made by DOE in arriving at quantitative values of the contained and airborne source terms. The basis for acceptance in this review is that all sources of radiation that necessitate shielding, special ventilation, or access control are described to the degree needed for the shielding codes used in the design process.

12.2.1 Contained Sources

The principal source of radiation during full-power reactor operation is the core. Radiation sources include prompt neutrons and gamma rays from the fission process and secondary gamma rays produced in the fuel, reflectors, and structural materials. These radiation sources determine the reactor-cavity-shielding requirements, establish the radiological environmental conditions in the reactor cavity, determine the neutron streaming to adjacent equipment areas and through the reactor cavity cooling duct, and determine the activation of air constituents and structural materials in the reactor silo.

Other sources of radioactivity include the primary cooling system (due to fission products in the primary coolant and system plateout), the helium purification system (HPS), and the radioactive waste system. Listings of all the components containing radioactive sources that are inputs to the radioactive waste systems are provided in Chapter 11 of the PSID. The component dimensions and physical locations in the plant should be specified in the PSSAR so that all important sources of radioactivity can be located on plant layout drawings. Section 12.2.1 of the PSID contains tables listing the activities (broken down by energy group) of the core, spent-fuel elements, primary coolant, plateout sources, and the HPS.

12.2.2 Inplant Airborne Sources

The principal sources of inplant airborne radioactivity will be neutron activation of air in the reactor cavity and miscellaneous equipment leakages. Cooling air and air within the reactor cavity can be activated by the neutron flux from the reactor vessel. The dominant activation isotope of air within the reactor cavity is argon-41. This is also the only isotope that has a concentration higher than the maximum permissible concentration (MPC) limit of 10 CFR Part 20 at shutdown after 2 years of activation in the reactor cavity. This concentration will decay, however, to below MPC limits in less than 10 hours after reactor shutdown.

Equipment leakage is the main source of airborne radioactivity outside the reactor cavity. Equipment and valves for radioactive systems are designed and selected to minimize leakage. The heating, ventilating, and air-conditioning (HVAC) system is designed to control the spread of airborne activities into other plant areas by collecting and routing airborne equipment leakages to the appropriate ventilation treatment systems. Because of low primary-system activity and selected equipment and the HVAC system designs, airborne-radioactivity levels in plant areas should be maintained well below MPC levels during all modes of plant operation.

In the PSSAR, Section 12.2.2 should contain a tabulation of the calculated concentrations of airborne radioactive material by nuclides expected during normal operation and AOOs for equipment cubicles, corridors, and operating areas normally occupied by operating personnel. The models and parameters used for calculating these airborne-radioactivity concentrations should also be provided.

The information on occupational radiation sources contained in Section 12.2 of the PSID is acceptable for this stage of the design.

12.3 Occupational Radiation Protection Design Features

12.3.1 Facility Design Features

The acceptability of the facility design features of the standard MHTGR is based on DOE's application of the guidance contained in Regulatory Guide 8.8. The radiation protection design features are intended to help maintain the occupational radiation exposures below the goal set by the user and, thus, within the limits of the top-level regulatory criteria. The main and shutdown cooling circulators are designed to minimize inplace maintenance and can be moved to a low-radiation area for repair. They are shielded to reduce radiation levels during removal and can be tested and inspected remotely from a low-radiation area. The capacity for passive decay-heat rejection is an exposure-reduction design feature, since it eliminates the need for additional, active core cooling systems that would require maintenance in a radiation environment.

The MHTGR design provides for special remote-handling facilities, casks, and shielded storage wells for the following equipment and systems to minimize personnel radiation exposures: main and shutdown helium circulators, inner crossduct, steam generator tube bundles, control rod assemblies, spent-fuel elements, and radioactive filters and adsorbers.

The MHTGR design contains many features to minimize occupational radiation exposures. Filters that can accumulate high radioactivity levels are designed to be backflushed or replaced remotely. Pumps are equipped with mechanical seals, and associated piping is arranged to reduce servicing and repair or replacement time. Tanks are designed to minimize crud settling. Heat exchangers are designed to minimize leakage and are provided with adequate space for onsite maintenance and tube pulling. Remotely operated valves are used to minimize personnel exposures from valve operation. Instrument transmitting and readout devices are located in low-radiation zones. In the PSSAR, DOE should expand the description of the design features incorporated to facilitate system and component decontamination (such as piping taps and process points). These design features will not only facilitate decontamination during operation, but will also serve to maintain ALARA radiation doses during decommissioning operations.

In addition to plant equipment and components designed to comply with the guidelines of Regulatory Guide 8.8, the facility layout is designed to reduce radiation exposures. Valve galleries and equipment cubicles are provided with shielded entrances. Whenever practicable, radioactive pipes are separated from nonradioactive pipes and are located in shielded pipe chases. Penetrations through shield walls are designed to minimize radiation streaming. Major radioactive components are isolated and shielded in individual compartments. Viewing windows or devices are provided in rooms intended to house highly radioactive sources.

The design features incorporated in the standard MHTGR for maintaining ALARA occupation radiation doses during plant operation and maintenance conform with the guidelines of Regulatory Guide 8.8 and are acceptable.

12.3.2 Shielding

The shielding design for the MHTGR will be acceptable if the methods used are comparable to commonly accepted shielding calculations and assumptions and if the shielding serves to minimize personnel exposures.

The design objectives of the shielding for the MHTGR are to (1) ensure that radiation exposures to plant operating personnel, contractors, administrators, and visitors are below 10 percent of the limits of 10 CFR Part 20; (2) ensure sufficient personnel access and occupancy time to allow normal anticipated maintenance, inspection, and operations required for each plant equipment and instrumentation area; and (3) reduce potential neutron activation of equipment and mitigate the possibility of radiation damage to materials. The shielding thicknesses provided to minimize plant-personnel exposure were based on maximum equipment activities and were selected to reduce the aggregate, computed radiation level from all contributing sources below the upper limit of the radiation zone specified for each plant area.

Plant areas are divided into eight radiation zones. The dose-rate criterion for each of these zones is based on the expected occupancy and access restrictions for each zone. Each room, corridor, and pipeway of every plant building was evaluated for potential radiation sources during normal operation, shutdown, and emergency operations; for maintenance occupancy requirements; for general access requirements; and for material exposure limits to determine appropriate zoning. Plant layout drawings contained in PSID Section 12.3 depict the specific zoning for each plant area during normal full-power operation and for 24 hours after shutdown. Areas having radiation levels that could cause a whole-body exposure in any one hour in excess of 0.25 mrem are designated as restricted areas where access control is required. Any high-radiation area with a radiation level greater than 1000 mrem per hour is provided with a locked door or other permanent, positive access controls to prevent unauthorized entry into this area. In the PSSAR, DOE should provide a listing of all potentially accessible high-radiation areas having dose rates exceeding 50 rem per hour and should describe what controls (for example locks, administrative controls, area radiation monitors, and signs) these areas will have to preclude personnel entry. Stringent high-radiation-area controls should be provided for these areas because of the increased hazard. The zoning system and access-control features meet the posting-of-entry requirements of 10 CFR 20.203 and are consistent with the guidance in Regulatory Guide 8.8.

The PSID describes the shielding design used in each of the plant buildings that house radioactive components. In each case, the shielding is provided to attenuate direct radiation through walls and penetrations and scattered radiation to less than the upper limit of the radiation zone for each area. Multiple step-duct designs are provided in the reactor cavity cooling system to minimize neutron scattering to the operating-floor area, which is a zone II area (1 mrem per hour) during normal power operation. To minimize radiation exposure to plant personnel, additional shielding is provided under the reactor vessel at the shutdown cooling circulator, inside the shutdown cooling circulator heat exchanger, and above the main circulator.

In the radioactive waste management building, radioactive tanks are located in compartments that are separated from pumps and their associated equipment. Radioactive process pumps are located in separately shielded compartments, and valves for radioactive systems are located in shielded valve areas that are separated from pumps and tanks. Labyrinths are provided to minimize radiation streaming from highly radioactive components to normally accessible areas or corridors. Pipe chases with labyrinths are utilized for highly radioactive pipes. Remote handling is provided for radioactive filters and spent-resin processing equipment. The shielding is designed to maintain ARARA personnel radiation exposures and is acceptable.

The shielding design described in Section 12.3.2 of the PSID follows the guidance contained in Regulatory Guide 8.8 and satisfies the facility's design objectives. In addition to the information currently in PSID Section 12.3.2, the PSSAR should include a description of the codes used in the shielding calculations.

The PSSAR should contain the results of a design review of station shielding to ensure the accessibility of vital areas after an accident (in accordance with the criteria of Item II.B.2 of NRC report NUREG-0737, "Clarification of TMI Action Plan Requirements"). These results should include postaccident source terms, a listing of plant systems containing highly radioactive materials following an accident, a set of postaccident radiation zone maps depicting the radiation levels in various areas of the plant 1 hour after the accident, a list of the vital areas that will require continuous or frequent occupancy following an accident, and a summary of the integrated doses to personnel in the above-listed areas for the duration of the accident.

12.3.3 Ventilation

The ventilation system for the MHTGR is considered to be acceptable if it maintains airborne concentrations of radioactive material in areas normally occupied within the limits in 10 CFR Part 20 and if DOE has applied the guidelines in Regulatory Guide 8.8 or suitable alternatives.

The ventilation system for the MHTGR is designed to maintain inplant airborne-radioactivity levels in plant areas well below the limits of 10 CFR Part 20 during all modes of plant operation and to minimize the spread and exfiltration of airborne contamination. Equipment and valves for radioactive systems are designed and selected to minimize leakage. DOE plans to maintain ALARA personnel exposures by (1) maintaining air flows from areas of potentially low airborne contamination to areas of progressively higher potential airborne contamination, (2) exhausting a greater volumetric flow from potentially contaminated compartments than is supplied to maintain a negative pressure in these areas, and (3) processing air from potentially contaminated areas through filters and charcoal adsorbers to reduce airborne-radiation concentrations. These design criteria are in accordance with those in Regulatory Guide 8.8 and are acceptable.

To facilitate maintenance and inplace testing operations, the air cleaning system for the MHTGR is designed to be consistent with the guidance and recommendations of Regulatory Guide 1.40, "Design, Testing, and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants" (Rev. 1). The ventilation fans, coolers, and filters are provided with adequate space to allow easy access and permit servicing with minimum personnel-exposure time. The HVAC systems are designed to require low maintenance and permit rapid repair of components. HVAC systems that service nonradioactive systems or areas are located in low-radiation zones to permit unrestricted accessibility. The ventilation system for the standard MHTGR is designed to maintain personnel exposures well within the limits of 10 CFR Part 20 and is, therefore, acceptable.

12.3.4 Area Radiation and Airborne Radioactivity Monitoring Instrumentation

Area Radiation Monitoring Instrumentation

The area radiation monitoring system complements the personnel and area radiation survey provisions of the plant radiation protection program to ensure compliance with the occupational exposure limitations of the top-level regulatory criteria. It is designed to immediately alert plant personnel entering or working in normally nonradiation or low-radiation areas (1 mrem per hour) of abnormally high radiation levels that could result in inadvertent overexposures and inform main control room operators of the occurrence and approximate location of an abnormal radiation level in nonradiation or low-radiation areas. To meet these objectives, DOE plans to provide area radiation monitoring in areas where personnel have routine access and where there is a potential for personnel to unknowingly receive radiation doses in excess of defined limits in a short period of time because of system failures or improper personnel actions. The PSID provides a list of potential locations for area radiation monitors based on monitor-placement criteria provided in Section 7.4.2 of the PSID.

In the PSSAR, Section 12.3.4 should state whether the area radiation monitors in the vicinity of the fuel-storage areas meet the requirements of 10 CFR 70.24. It should also state whether and how the area radiation monitoring system conforms to the guidelines of Regulatory Guide 8.12, "Criticality Accident Alarm Systems," and Regulatory Guide 8.2, "Administrative Practices in Radiation Monitoring."

Airborne Radioactivity Monitoring Instrumentation

Airborne radioactivity monitoring instrumentation is used to (1) monitor and record concentrations of airborne radioactivity in the air within an enclosure by either direct measurement of the enclosure atmosphere or of the exhaust air from the enclosure, (2) monitor potential release paths to the environment, and (3) alarm on high radioactivity levels. Local alarms are provided to alert personnel in the area where the airborne-radioactivity concentration is at or above the setpoint value to ensure that the top-level regulatory criteria are met. The system provides a continuous record of airborne-radioactivity concentrations in the control room.

Combination halogen gaseous monitors are used where inhalation of airborne radioactive materials by plant personnel is a possibility. The sampling system for these monitors is designed and installed in accordance with the ANSI/ANS-N13.1 guide to sampling of airborne radioactive materials. Fixed airborne radioactivity monitors will be located to monitor (1) normally accessible personnel-operating areas in which there is a potential for airborne radioactivity; (2) exhaust ducts that serve an area containing processes which, in the event of major leakage, could result in plant concentrations approaching 10 CFR Part 20 limits; (3) outside air intake ducts for the operations center; and (4) exhaust to the environment. In addition, the PSID (Section 7.4.2.4.1) states that dilution from other exhaust ducts is considered when locating monitors in exhaust systems to ensure maximum coverage and still be able to detect 10 CFR Part 20 airborne-radioactivity limits in the area with the lowest ventilation flow.

Portable continuous air monitors (CAMs) will be used to monitor local areas where there is a possibility of airborne radioactivity during maintenance of radioactive systems. CAMs will also be used to monitor abnormal operations involving the spread of airborne radioactivity. To ensure that the fixed airborne radioactivity monitors are operating properly, periodic grab samples for particulates, iodine, and noble gases will be taken throughout the plant. All radiation monitors will be calibrated on a quarterly schedule.

In the PSSAR, the criteria and method for obtaining representative inplant airborne-radioactivity concentrations from the areas being sampled should be provided. Specifically, the airborne radioactivity monitoring system should be capable of detecting 10 MPC (maximum permissible concentration) hours of particulate and iodine radioactivity from any compartment that has a possibility of containing airborne radioactivity and that normally may be occupied by personnel.

With the exception of items not in compliance with 10 CFR 70.24 and Regulatory Guides 8.2 and 8.12, the objectives and location criteria for the area radiation and airborne radioactivity monitoring systems are in conformance with 10 CFR Parts 20 and 50 and Regulatory Guide 8.8. The material in Section 12.3 of the PSID pertaining to facility design features, shielding, ventilation, and area radiation and airborne radioactivity monitoring instrumentation (except as noted above) is acceptable. In the PSSAR, the plant layout drawings should show (in addition to what is currently shown) shield-wall thicknesses, controlled-access areas, personnel- and equipment-decontamination areas, personnel "dress-out" areas, personnel traffic patterns, location of airborne radioactivity and area radiation monitors, and location of the counting room.

12.4 Occupational Dose Assessment

The acceptability of the standard MHTGR dose assessment is based on the thoroughness with which DOE has provided occupancy factors, dose rates, and numbers of personnel required to perform job functions in various areas of the plant and on the methods used to perform the dose assessment.

The goal of DOE's dose assessment is to limit the collective annual exposure to the entire plant staff to an annual, average, integrated dose of less than 10 percent of 10 CFR Part 20 limits. DOE used two different methods, depending on the work category considered, in performing the dose assessment. An area-by-area and task-by-task method was used in estimating doses for preventive and corrective maintenance and inservice inspection. This is the method used in Regulatory Guide 8.19, "Occupational Radiation Dose Assessment in Light-Water Reactor Power Plants - Design Stage Man-Rem Estimates." In this method, maintenance and inspection tasks are assigned to the various plant areas, and occupancy times are developed based on the task's manpower, duration, and frequency. A general area dose rate for each plant area was then used to calculate the estimated person-rem per year. HTGR operating experience to date was used for these estimates, where applicable. In particular, Fort St. Vrain experience was used in estimating main circulator maintenance and removal, primary relief valve maintenance, helium purification equipment maintenance, and control rod drive mechanism maintenance. To estimate the doses associated with routine operations, waste processing, and refueling, DOE used time-averaging to estimate the amount of time workers will typically spend in different radiation zones. Although different from the method suggested in Regulatory Guide 8.19, this dose-averaging method takes into account the wide variety of activities in many different plant areas associated with these work functions. After calculating the estimated person-rem associated with each of the six work functions suggested in Regulatory Guide 8.19, DOE added a contingency of 20 percent to account for miscellaneous minor tasks not included and uncertainties in the numerical data.

In addition to using HTGR operating data, DOE used information presented in NRC report NUREG-0713, Volume 5, "Occupational Radiation Exposure at Commercial Nuclear Power Reactors - 1983," March 1985, as a source of historical exposure data. The resulting total, annual, collective-dose estimate from all six major work functions at the MHTGR is 149 person-rem (for a four-module plant of 350 MWt/125 MWe, each module with 80-percent availability). This estimate is well below the cumulative, average, annual operating dose of 616 person-rem for light-water-cooled reactors (based on data from 1974 through 1986) and is approximately 10 percent of the 10 CFR Part 20 limits for the standard MHTGR's estimated, maximum, permanent plant staff of 308. DOE plans to achieve this dose goal by using or improving on the existing and proven HTGR technology. DOE's assumptions on which its dose estimates for occupational exposures are based meet the intent of Regulatory Guide 8.19 and are acceptable. In the PSSAR, however, tables should be added to Section 12.4 that contain the input data (that is, task or plant area, average area dose rate, number of personnel involved for each task, frequency of task, exposure time, and dose per task or area) used to calculate the overall estimated dose for each of the six work functions.

12.5 Operational Radiation Protection Program

The PSID for the standard MHTGR does not contain a Section 12.5. As stated in the Standard Review Plan (SRP), Section 12.5 should describe the applicant's health physics program with respect to organization, equipment, instrumentation, facilities, and procedures. Since the PSID is a preliminary document to the PSSAR and since the PSID describes a standardized plant (versus a plant at a specified site), most of the level of detail included in Section 12.5 of the SRP is not warranted at this stage of the review. In the PSSAR, however, Chapter 12 should include (1) a description of the administrative organization of the health physics program, including the authority, responsibility, and training for each position identified; (2) the criteria for selection of portable and laboratory technical equipment and instrumentation for performing radiation and contamination surveys, area radiation and airborne radioactivity monitoring, and personnel monitoring during normal plant operation and AOOs; and (3) a description (including location) of the health physics facilities, access-control stations, laboratory facilities, decontamination facilities, and other contamination-control equipment and facilities.

12.6 Discussion and Conclusions

A measure of the expected low-level occupational exposure for the MHTGR is illustrated by Fort St. Vrain experience, which has demonstrated occupational radiation-exposure levels substantially less than those for light-water reactors (LWRs). For Fort St. Vrain, this may be attributed to the following factors: (1) the entire primary coolant system is within and shielded by a prestressed-concrete reactor vessel, (2) refueling is performed automatically, (3) relatively low quantities of both liquid and gaseous radioactive wastes are generated, and (4) a comparatively low quantity of radionuclides circulates with the helium coolant - this results in low contamination levels and low maintenance exposures for primary-system components. Significant areas where Fort St. Vrain experience is not directly applicable to the MHTGR are (1) the steel reactor vessel, as discussed in Section 5.2.5.C in relation to the frequency of inspection that may be caused by service level C and D occurrences, and (2) inspection and maintenance of components unique to the MHTGR design, such as the hot duct and the crossduct vessels.

DOE's estimate for total occupational exposure for the MHTGR is between four and seven times lower than for LWRs. This estimate is consistent with the operational experiences of Fort St. Vrain and Peach Bottom 1. Although the MHTGR differs in system design and plant layout from these earlier HTGRs, the staff believes that low MHTGR occupational exposure is achievable because of existing and proven technology, together with the features described above or those that will be described at the PSSAR stage of review.

13 CONDUCT OF OPERATIONS

13.1 Emergency Preparedness*

13.1.1 Summary

DOE described its emergency plan in the Emergency Planning Basis (EPB) Report, DOE-HTGR-87-001 (DOE, 1987-2). The major purpose of this report was to request NRC agreement that the emergency plan for the MHTGR contain no explicit plans or drills for rapid notification, sheltering, or evacuation of the public. Rather, if these actions became necessary, they would be performed on an ad hoc basis. This request is in accordance with the staff-proposed criteria that would not require preplanned notification, sheltering, and evacuation if the protective action guidelines (PAGs) of the U.S. Environmental Protection Agency were not exceeded for 36 hours following the initiation of any credible event. The details and the bases for the staff's proposed criteria are presented in Section 3.2.2.4 of this report.

The staff has reviewed DOE's request and considers such a change in emergency-planning policy as potentially acceptable on the basis of safety analyses performed by both DOE and the staff's contractors at Oak Ridge National Laboratory (ORNL) and Brookhaven National Laboratory (BNL). The staff's detailed considerations are given in the five subsections that follow.

13.1.2 Existing Emergency-Preparedness Requirements for Light-Water Reactors

Existing emergency-preparedness requirements for light-water reactors (LWRs) are found in 10 CFR Part 50.47 and Appendix E to 10 CFR Part 50. A key feature of the existing emergency-preparedness requirements is the need for plans and response capability to implement protective actions for the population within a plume exposure pathway emergency planning zone (EPZ) of about 10 miles in radius.

In the most concise terms, this means the ability to promptly evacuate or shelter that population near the reactor. All of the detailed emergency-preparedness requirements for offsite response flow from this key feature, that is, prompt notification, backup communications, dose assessment, environmental monitoring, medical services, training, and annual (onsite) and biennial (offsite) exercises. These requirements are summarized in Table 13.1.

Another feature of the existing emergency-preparedness requirements is the need for plans and preparedness to implement protective actions for an ingestion exposure pathway EPZ of about 50 miles. In the framework of the regulations,

*Indicates statement is particularly sensitive to change by evaluation of forthcoming DOE information.

these ingestion-pathway requirements generally are not discrete from the plume-pathway requirements. However, through implementing guidance, the ingestion-pathway requirements are directed to a different purpose (for example, embargo and modification of food production), and there is generally more time for implementation (hours to days).

For both the plume pathway and the ingestion pathway, the onsite plans of the utility and the offsite plans of the State and local governments are fully coordinated; however, they dovetail the closest in regard to notification, coordination of protective-action recommendations, and joint exercises. The staff's review focused on the offsite plans because this is the area in which DOE and the staff are proposing the furthest departure from existing requirements for LWRs.

13.1.3 DOE Proposal for Reduced Emergency-Preparedness Requirements for the MHTGR

In the Emergency Planning Basis Report, DOE developed its position with respect to emergency planning on the basis that the design features of the MHTGR, with its passive reactor shutdown and cooling systems and with core-heatup times much longer than those for LWRs, result in a system that is safe enough to warrant a reduction in the plume exposure pathway EPZ radius to the site boundary. Accordingly, DOE proposed that prompt public notification and provision for sheltering and evacuation of the general public not be included in the emergency plan.

In support DOE offered an analysis that considered low-frequency events in an approach similar to that in NRC report NUREG-0396, "Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Nuclear Power Plants." DOE's probabilistic risk assessment analyses for the MHTGR indicated that the MHTGR would not exceed the plume-exposure protective action guidelines (PAGs) at the site boundary for any transient or event with a mean frequency greater than 5×10^{-7} per plant-year. This result was also found for the staff-postulated bounding events discussed in Section 15.2.3.3. These conclusions, based on DOE's analyses, were tentatively confirmed by the staff's contractors at ORNL and BNL. The analyses showed that maximum fuel temperatures would not exceed the fuel-failure thresholds expected by DOE at any time and that the temperatures at 36 hours are well below the 60- to 100-hour maximum values computed. The staff believes that the analyses indicate sufficient margin so that the staff's proposed criteria could be met on the basis that the information provided by DOE at this stage of the review is later confirmed. At later review stages, the staff will make other and separate determinations based on improved descriptions of the MHTGR safety features, further safety analyses, the results of the research programs on fuel integrity, and specific siting considerations. The overall result for present consideration is that the MHTGR could conservatively meet a 36-hour criterion for not exceeding the PAGs.

It is this tentative conclusion that forms the conditional basis for the staff's proposals in Sections 13.1.5 and 13.1.6. The use of these staff proposals for a specific site is also conditioned on the successful resolution of the underlying siting and safety issues involved and, of course, resolution of the containment adequacy issue as described in the "Preface."

13.1.4 Relationship of Emergency Planning Zone Size to Emergency-Planning Policy

Although 10 CFR 50.47(c) states that the size of the EPZ may be determined on a case-by-case basis for gas-cooled nuclear reactors, the staff has concluded that this provision is only indirectly relevant to the emergency-planning considerations for the MHTGR. Rather, the staff has concluded that the DOE proposal for restricting the plume exposure pathway EPZ to the site boundary is equivalent to not requiring offsite emergency planning for the protection of the public. Since the current policy of the NRC is that offsite emergency planning is a requirement for the licensing and operation of a nuclear power plant, the staff has addressed the DOE proposal as a request for a change in this policy rather than an adjustment of the EPZ size. This is because an adjustment of the EPZ size, particularly a radical one like that proposed for the MHTGR, is in conflict with a stated objective of the current EPZ requirement in that the current 10-mile EPZ provides a substantial base for expansion of response efforts beyond the 10-mile boundary if this should prove necessary. This is explicit in the planning bases given in NRC report NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants." To date, proposals for smaller EPZs have not addressed this important issue.

Should DOE's research and final design development programs satisfactorily address the staff's concerns regarding the potential for large offsite releases, the staff concludes that such a change in policy could be warranted. Current emergency-planning regulations are based on an underlying assumption that a serious accident could occur and that such an accident could result in offsite individuals being exposed, in a relatively short time, to levels of radiation high enough to require medical care. Based on the staff review of the DOE submittal for the MHTGR, it appears that releases exceeding the lower-level PAGs of 1 rem to the whole body and 5 rem to the thyroid would not occur at all, or if they did occur it would not be for a few days, and that higher-level releases that could require the need for medical care as contemplated by the current regulatory policy would not occur at all.

The remainder of this section reflects the staff's evaluation and conclusions regarding the minimum emergency planning that could be approved should the final design of the MHTGR support such a change in policy. Consideration of site-specific parameters may require that additional requirements be imposed at a later time.

Because of the long times available, the staff concludes that any evacuation triggered by an MHTGR accident could be accomplished ad hoc, that is, by using State and local government plans that already exist for dealing with national hazards (for example, hurricanes, floods, fire, earthquakes, and technological hazards such as chemical accidents, explosions, and fires) to respond to potential MHTGR accidents.

Historically, ad hoc evacuations for such emergencies as hurricanes, chemical fires, and transportation accidents in the United States have taken from 2 to 8 hours including the time to notify the population. This is typically accomplished by route alerting using fire trucks and police cars, with door-to-door followup. Newspapers, radios, and televisions assist in the notification process. In many respects, the response to an MHTGR accident would be similar to the response to a hurricane, for which there is a long period to monitor the course of the event and to determine and implement protective actions.

As described in Section 13.1.6, the staff is proposing criteria that would ensure that at least 24 hours would be available for emergency response before any offsite protective actions became necessary. The staff believes that this is sufficient time for local agencies to take such protective actions (for example, sheltering or evacuation) using their existing emergency plans coupled with radiological emergency plans as described in Section 13.1.5.

13.1.5 Content of Emergency Plans for the MHTGR

Section 8 of the Emergency Planning Basis Report states that the MHTGR's emergency plan will be prepared later. However, DOE stated that it would not include offsite exercises and drills or prompt public notification but that it would include ingestion-pathway plans. The staff herein describes what it would propose to require for emergency plans for the MHTGR and other advanced reactors that meet the qualifying criteria in Section 3.2.2.4. The staff's proposal for these emergency plans is described in narrative form by comparing them with the current requirements for LWRs. In addition, the existing requirements and the proposed requirements are given in Table 13.1.

The requirements for onsite utility plans for the MHTGR (that is, notifications, exercises, and arrangements for requesting and using offsite assistance on site) would be essentially the same as the current regulations except where the onsite plans correlate with offsite plans. For example, exercises involving the plume exposure pathway would no longer be part of either plan.

The remainder of this section focuses primarily on offsite plans. First, for the MHTGR, the 10-mile plume exposure pathway EPZ would be eliminated and the 50-mile ingestion exposure pathway EPZ would remain. Further, the prompt-public-notification requirements in offsite plans would be eliminated for the MHTGR primarily because of the much longer times available to make notifications and to take protective actions (24 hours or more). The dose projections and assessment requirements in offsite plans for the plume exposure pathway would be eliminated because the much longer times available would permit an independent confirmation of the utility's projections by State and Federal organizations. Offsite environmental monitoring requirements for the plume exposure pathway would be eliminated for the same reasons; that is, the utility's monitoring provisions would suffice until others could be put in place. However, at a later review stage, it would be necessary for the utility to demonstrate through technical specifications or other acceptable administrative controls that the necessary equipment could be made available within a reasonable period and that personnel would be adequately trained for its use. For the ingestion exposure pathway EPZ, requirements for dose projections and assessment and environmental monitoring would remain.

Requirements in offsite plans related to arrangements for medical services for contaminated or injured members of the general public would not be necessary because of the lower releases and in any case could be determined as the need arose because of the longer times available. The present requirement in offsite plans for primary and backup communications would be retained because such communications must be in place before any accident occurs.

Training for response in the plume exposure pathway EPZ would not be required for offsite plans for the MHTGR because the extra time would permit instruction to be given, if necessary, to supplement the general training in emergency response that is part of State and local governments' normal programs. The requirement for training for response in the ingestion exposure pathway EPZ would be retained. The exercise requirement for State and local governments for the plume exposure pathway EPZ would also be eliminated; however, the exercise requirement for State and local governments for the ingestion exposure pathway EPZ would be retained. The current requirement for training and exercises for offsite emergency workers who would respond on site, such as, police, fire, and rescue personnel, is traditionally part of the onsite plan. This would remain a requirement for the onsite plan.

Finally, the ability to shelter and evacuate the general public would involve the use of present State and local government sheltering and evacuation plans for responding to natural and other technological hazards. That is, the existing State and local emergency plans for other hazards would be bolstered by the minimum additional offsite planning described herein.

13.1.6 Qualifying Criteria

Instead of accepting the DOE proposal for a plume exposure pathway EPZ at the site boundary based on a NUREG-0396-type analysis, the staff proposes to accomplish the same objective by using the criteria in Section 3.2.2.4 as the basis for qualifying for reduced offsite emergency planning. Although an offsite emergency plan for the ingestion exposure pathway EPZ would still be required, offsite planning would not have to include early notification, detailed evacuation planning, and provisions for training and exercising within a plume exposure pathway EPZ.

The criteria in Section 3.2.2.4 give credit for designs such as that of the MHTGR that provide a sufficiently long time before a significant radiation release. For designs such as these, the staff concludes that because sufficient time is available, reasonably timely notification of offsite authorities will permit effective protective actions without the level of planning currently required for LWRs.

The first qualifying criterion in Section 3.2.2.4 ensures that all events considered for design and siting purposes do not lead to offsite doses in excess of the PAGs early in the event sequence. Based on historical ad hoc evacuations in the United States (which have ranged between 2 and 8 hours), 24 hours is sufficient time for local agencies to take protective actions (for example, sheltering or evacuation), and in these cases planning does not substantially reduce the risk to the public. The 24 hours, combined with 12 hours for the plant staff to diagnose the event and attempt corrective action before initiating evacuation or sheltering, is the basis for the 36-hour criterion.

The second criterion in Section 3.2.2.4 ensures that events beyond those considered for design and siting purposes (of a frequency similar to those events considered in NUREG-0396 for LWR emergency-planning purposes) are considered for advanced-reactor emergency-planning purposes and that they do not contribute substantially to overall risk.

13.2 Role of the Operators

The role of the operators was reviewed in the context of the instrumentation and control design evaluated in Chapter 7; the operators' responsibilities as described in Section 13.2, "Description of Plant Operational Control," of the Preliminary Safety Information Document (PSID); the human-factors aspects of the major man-machine interfaces within the plant; and the operators' responsibilities with respect to the safety analyses described in Chapter 15. The staff has also reviewed the human-factors discussions and commitments provided in the document "MHTGR Assessment of NRC LWR Generic Safety Issues" (DOE, 1987-4).

The PSID states that the principal distinction between the responsibilities of licensed operating personnel and the responsibilities of other plant operations personnel is that licensed personnel are the only personnel permitted to manipulate apparatus and mechanisms that can directly affect the reactivity and power level of the reactor. Manipulation of apparatus and mechanisms that affect other nuclear-related chemical or physical processes by nonlicensed personnel is permitted only with the knowledge and consent of licensed operating personnel.

13.2.1 Description and Safety Objectives

The role of the MHTGR plant control room operators will differ from the role of control room operators in current LWR plants. DOE proposes that the operators will not perform any 10 CFR Part 100-related functions from the control room, since time periods of hours and days are available before manual actions are needed should the automatic safety-protection features fail to perform their functions. Any safety-related manual actions assigned to operators would be for low-probability events, for example, those identified in Chapter 15 as event category III. Thus the operators' roles will consist primarily of monitoring and releasing holdpoints so that automatic control can proceed. The operators can also take discretionary action, such as changing control setpoints, bringing alternate equipment into service, removing failed components from service, or performing administrative operations. If automatic controls were unable to return systems to predefined stable states, plant control would automatically revert to semiautomatic control. The control room operators, with guidance from the computerized control system, would take manual remedial actions to place the plant in a stable condition.

DOE stated that for the current status of the man-machine design, operator-workload analyses indicate that for the sequences analyzed, the MHTGR control-room-operator workload is less than one-half of the typical industry requirements for operator loading and also less than the actual operator workloads of existing LWR and HTGR plants. DOE proposes for the standard MHTGR 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 operators. These personnel will operate the plant through their interface with the plant supervisory control subsystem and those operator areas of responsibility outside the control room. The roving operators will monitor equipment and systems and provide for operation of local equipment in the plant complex.

Operator and machine tasks will be analyzed to confirm the size of the operations staff. Operator tasks and human-machine performance tests will be developed and validated using an interactive engineering simulation system.

The engineering simulation system will also serve for operator training, qualification, and examination. The design of the control room and the man-machine interface will include human-factors engineering.

The maintenance activities will be on a 24-hour-per-day basis to maximize the benefit of the online diagnostic system available through the automated control system. The surveillance portion includes the activities required to check and verify the satisfactory performance of various plant components. The operational support contribution is the time required from the operational staff to prepare, accomplish, or recover from maintenance or evaluation of plant-component performance.

The reactor module control subsystems will supply information and control capabilities to personnel with responsibilities for operations, test and calibration, engineering, maintenance (hardware and software), and management. Locations where the reactor module control systems have man-machine interfaces are the main control room, the remote-shutdown area, the plant protection and instrumentation system (PPIS) equipment room, the computer room, test and calibration stations, local control stations, and engineering offices. No single failure can eliminate information-handling functions because redundant capacity exists for both reactor-module data processing and storage. For balance-of-plant subsystems, secondary monitoring and control can be achieved from local panels to facilitate maintenance activities.

The data management subsystem (DMS) serves to provide plantwide data communication and centralized data processing. The DMS acquires, transmits, processes, records, stores, diagnoses, and distributes data and information for both onsite and offsite use and for immediate and future use. Distributed-data-communication controllers and high-speed digital computers perform the data-communication and data-processing functions. A distributed-communication network interconnects with the data-communication controllers. The network consists of multiple sets of optical-communication cables referred to as "data highways." The DMS network observes communications by detecting acknowledgement of readiness status for communication and monitoring digital-signal-transmission integrity. The network controllers schedule transmissions, select available communication routes, and determine and report if any communication errors occur. The DMS data processors accept system-user instructions to execute software programs and retrieve or store data. The data processors acquire data from the DMS communication network, store plant-process variables and status data, and record sequences of events. The data processors schedule execution of processing tasks and identify unauthorized interactions or data-security violations.

The safety protection information equipment consists of field-mounted electronic-multiplexer modules, redundant digital-data highways, redundant microprocessor equipment, and instrumentation displays in the remote-shutdown area and the PPIS equipment room. These displays will assist operators in verifying that the plant's safety-related systems are operable. Also, this display equipment provides a continuous, dedicated display of a minimum set of plant parameters or derived variables used by operators to evaluate the plant's safety status. These displays are also accessible in the main control room and other locations in the plant through the DMS.

The postaccident monitoring (PAM) instrumentation will provide data on plant variables needed by the operating personnel during and following an accident.

DOE proposes that these data (1) provide information required to permit the operator to assess that the reactor is safely shut down and is being cooled, (2) determine whether trip and other safety-related systems perform their intended functions, and (3) provide information to the operators that will enable them to determine the status of the radioactivity barriers.

The investment protection subsystem (IPS) provides the sense and command features necessary to sense plant variables, detect abnormal conditions, and initiate actions required to protect the plant investment. The IPS is not safety related, although it is a part of the PPIS and is separate and independent of all other plant instrumentation and controls. The IPS operator interfaces are in the PPIS equipment rooms in each reactor building and the remote-shutdown area in the reactor service building. The operator interfaces include color video displays, function input devices, and keyboards. In addition, the IPS sends data through an isolator to the data management subsystem for display by the plant supervisory control subsystem in the main control room. The remote-shutdown-area operator interfaces provide the reactor operators with the capability of initiating investment-protection trip actions and taking the necessary actions to shut down the plant from a position remote from the main control room. No manual inputs to the IPS are available in the control room.

13.2.2 Scope of Review

This review evaluated human-factors information presented in PSID Chapter 7 and Section 13.2, as well as DOE responses to staff comments on this documentation. The staff considered in this review that the proposed automated control of the four reactor modules by a single plant operating crew is a significant departure from past nuclear industry practice and experience.

13.2.3 Review and Design Criteria

The review was guided by General Design Criterion (GDC) 19, "Control room," the requirements for detailed control room design reviews given in NRC report NUREG-0737, Supplement 1, "Clarification of TMI Action Plan Requirements," and the guidelines contained in NUREG-0700, "Guidelines for Control Room Design Reviews." In addition, the requirements of Institute of Electrical and Electronics Engineers Standard 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations," were used with respect to the staff's position regarding manual initiation of reactor trips. Additional criteria may need to be developed to recognize the unique and advanced features of the MHTGR's instrumentation and control system design with respect to human-factors principles in response to Safety Issues E and F in Section 13.2.5.

13.2.4 Research and Development

DOE considers that the development of the hardware and software for a fully automated control system is "applications technology" and not a topic for inclusion in the Regulatory Technology Development Plan (RTDP). The staff viewed this position favorably after review of the DOE response to Comment 13-15, wherein DOE summarized the extensive and successful use of automatic control in U.S. and foreign reactors. Safety issues are identified in Section 13.2.5, however, that include the staff's concerns relevant to achieving an advanced and automatic control system in the MHTGR that is acceptable with respect to human factors. DOE is developing a human-factors engineering plan that includes a task analysis and a staffing-requirements analysis. The staff expects this plan would include the use of a plant-specific simulator. Because it has concluded that some operator functions are safety related (see Section 13.2.6), the staff requires at a later review stage that this plan be included as a section in the RTDP.

13.2.5 Safety Issues

A. Manual Means for Reactor Trip

The staff review of the man-machine interfaces indicated that all manually initiated reactor trips would occur through non-Class 1E devices and components. DOE maintains that there are no manual safety-related functions or tasks for human operators because reactor shutdown and shutdown heat removal are inherent in the design. The staff does not find, however, that this design for manual reactor trip is adequate and acceptable. If the automatic controls were unable to return systems to predefined stable states, plant control would revert to semiautomatic control. In these circumstances the control room operator, with guidance from the computerized control system, would take manual remedial actions to place the plant in a stable condition. The staff notes, however, that human error by the operator or error(s) in the computer's hardware or software may mislead the operators, and thus may cause plant conditions that approach unsafe operation. Although the reactor trip system should perform as designed, the addition of a manual Class 1E-qualified initiation system would provide operators with additional means to maintain safe operation and defense-in-depth for unanticipated control-system failures.

B. Completion of Control-System Design

In future phases of the MHTGR review, the staff will focus on several points. The plant control, data, and instrumentation system, for instance, requires a large amount of software. To ensure that the software is reasonably error free, the staff will require a structured verification and validation program. The validation of the individual component systems within the plant control, data, and instrumentation system, as well as the validation of the system as a unit, are important steps to ensure that operators are not misled. Furthermore, the results from data processing must be clearly labeled for each reactor unit and module to avoid confusing operators, software designers, and maintenance personnel. During the review of a completed design, the staff plans to evaluate the diagnostic features and operator aids within the control and instrumentation systems. Also, specific features of operator work stations, including input devices, display formats, and annunciation of malfunctions, will be evaluated. Finally, the stability margins of the plant's control system during low-power operation of a reactor will be reviewed. A control system with low stability may oscillate or cycle between limit settings; this would impair operator monitoring tasks. This review will make use of the task analysis and staffing-requirements analysis to be described by DOE at the next review stage.

C. Postaccident Monitoring and Communications

DOE's proposal that the operators' role is not safety related is not acceptable based on consideration of postaccident monitoring and offsite communication functions following the occurrence of an accident. At a later review stage, it will be necessary to develop the detailed role for operators in this regard for staff approval. The staff disagrees with DOE's proposed position relative to the general nonapplicability of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants To Assess Plant and Environs Conditions During and Following an Accident."

D. Accident Mitigation and Recovery Actions

In previous sections of this report, the staff established the potential need for manual actions under emergency conditions to provide cold reactor shutdown (Section 4.3.5.C), to depressurize the reactor vessel (Sections 5.2.5.G, 5.5.5.B, and 9.2.5.D), and to repair the passive heat removal system (Sections 5.5.5.B and 6.2.5.D). The staff judges all these functions to be safety related.

E. Defense-in-Depth From Control Room Operators

The control room operations crew is the only means for diagnosing and responding to unanticipated and "untrained-for" plant transients. The staff acknowledges that the MHTGR design contains many inherently safe systems. Because of these systems, it is difficult to identify deterministically a safety challenge to the plant. On the other hand, the staff has no measure by which to conclude that a perfect design exists. The control room operations crew is the last line of defense to an imperfect design. The staff requirements for safety-grade means to trip the reactor, to provide postaccident monitoring and communications, and to ensure operator availability for accident mitigation and recovery actions, are the staff's approaches to address this concern. The staff's building design requirements for operator protection are described in Section 6.1.2, "Safety Issue - Location of Control Room and Protection of Reactor Operators."

F. Review Plan for Advanced Control-System Technology

The staff encourages the use of digital computer systems and advanced technology, such as expert systems. Properly designed, the attributes of these systems will overcome many of the human-factors limitations experienced with the use of analog hard-wired technology. Failures in digital equipment, however, frequently manifest themselves in ways that differ from those encountered in most other devices. Also, many software design errors are subtle and difficult to identify and correct. An effective design verification and validation program should minimize software errors. Finally, expert systems are generally finite in scope. An expert system may have limited ability to recognize when it is operating outside of its field of knowledge. Therefore, an expert system should only serve as an operator aid; the operator will have the full responsibility for the plant. At this time, the staff does not have a review plan for man-machine interfaces based on digital computer systems and advanced technology. Efforts to develop such a plan are under way, however, and will include the problems discussed above and other problems relevant to the safe operation of the plant.

G. Task Analysis, Crew Size, and Training

At a later review stage, DOE will be required to describe in the RDTP the scope, objectives, and facilities to be used for the human factors engineering plan. The findings from this research, particularly the task-analysis portion, will be considered by the staff as the final confirmation of the MHTGR instrumentation and control system design and will be a determining factor in the selection of the operating-crew size and the effectiveness of its training program.

H. Major Operator Error

In its letter to Chairman Zech, entitled "Report on Key Licensing Issues Associated With DOE Sponsored Reactor Designs," the Advisory Committee on Reactor Safeguards (Kerr, 1988-1) pointed out that the accidents at both Three Mile Island Unit 2 and Chernobyl Unit 4 were caused in large part by "deliberate but wrong" operator actions and that advanced-reactor designs should be demonstrated in a systematic way to be less vulnerable to such maloperations. In anticipation of this concern, DOE provided a brief discussion (Neylan, 1988-2) to illustrate how the passive and inherent safety features of the MHTGR can provide an improved level of protection against major operator error. This discussion used the same systematic approach as that used in the reactor design and safety analysis; namely, consideration of the control functions needed to retain radionuclides in fuel particles: (1) remove core heat, (2) control heat generation, and (3) control chemical attack. Although this initial study gave some confidence that the potential consequences of various postulated operator errors were bounded by the existing safety analysis, the study performed was not deemed conclusive by the staff. Further study and review of the possible spectrum and consequences of major operator errors will be required at later review stages.

13.2.6 Conclusions

The staff accepts, with caution, the proposed fully automated control system and the control of the four reactor modules by a single plant operating crew. In addition, it is important to note that as part of the conceptual design effort, DOE and its contractors involved personnel with operating experience in the design and review effort. The staff believes that such involvement is an essential contribution to ensure a safe design and encourages continued efforts in this regard. The staff does not, however, agree with DOE's proposal that the role of the operator is not safety related and that the plant is not vulnerable to major operator errors. DOE contends that since its safety analyses for the proposed design do not assume that an operator takes any action during the early course of an accident, there are no safety-related operator functions. The staff's position is that the presence of operators provides the necessary safety functions and the lines of defense to (1) monitor and provide confirmation of plant response, (2) communicate plant conditions following an accident, (3) provide mitigating manual actions, and (4) initiate recovery actions. Although the staff fully supports a reactor design that strives to eliminate the need for operator action during the course of an accident, this review was performed, and future MHTGR reviews will be planned, on the basis that the operators' presence is safety related. To accept DOE's proposal would require demonstration that all MHTGR failure modes, initial conditions, and failure scenarios are completely known. The staff position is that only after extensive experience has been obtained from plant operation, including the demonstration of a plant's safety characteristics, could such a proposal be reconsidered. Accordingly, the staff basis for review is that the MHTGR design must make provisions for an accessible and habitable control room and a safety-grade shutdown and monitoring area (or areas).

Specifically, the staff requires a minimum of one Class 1E-qualified manual reactor trip system for the MHTGR. This system shall be continually manned or accessible and a man-machine interface that meets human habitability requirements in terms of radiation, seismic, and environmental qualifications must be provided. At a future review stage, DOE should propose for staff review a postaccident monitoring and operations facility that is consistent with the staff's event category III safety evaluation.

With this position, the staff recognizes that the one Class 1E manual trip system does not meet the single-failure criterion. It is believed that one Class 1E manual-trip capability is adequate because of the inherent safety features in the design of the plant and because of the automatic reactor trip system, which meets the single-failure criterion. Furthermore, there are other manual means by which to trip the reactor, although these are not Class 1E.

13.3 Safeguards and Security

13.3.1 Scope of Review

The staff performed a review that focused on the potential of the design to meet the existing requirements for protection against radiological sabotage contained in 10 CFR Part 73, including 10 CFR 73.1, 73.2, and 73.55, and Appendixes B and C, and interpretation of requirements given in Regulatory Guides 5.7, "Entry/Exit Control for Protected Areas, Vital Areas, and Material Access Areas," 5.12, "General Use of Locks in the Protection and Control of Facilities and Special Nuclear Materials," 5.44, "Perimeter Intrusion Alarm Systems," and 5.65, "Vital Areas Barriers and Emergency Access to Vital Areas"; in Review Guidelines 10, "Power Supply to Security Lighting," 17, "Definition of Vital Areas," and 18, "Protected Area Control Function in Bullet Resistant Structure," in the Standard Review Plan; and in NRC reports NUREG-0908, "Acceptance Criteria for Evaluation of Security Plans," NUREG/CR-0509, "Emergency Power Supplies," and NUREG/CR-1327, "Security Lighting Planning." Special attention was given to how the MHTGR would foster the objectives of the Commission's Severe Accident Policy Statement, which states:

> The Commission...recognizes the importance of such potential contributors to severe accident risk as human performance and sabotage. The issues of both insider and outsider sabotage threats will be carefully analyzed and, to the extent practicable, will be emphasized in the design and in the operating procedures developed for new plants.

Also, Generic Issue A-29, "Nuclear Power Plant Design for the Reduction of Vulnerability to Sabotage", is one of the medium-priority generic safety issues for which that policy statement expects new designs to demonstrate technical resolution.

In performing this review the staff reviewed relevant sections of the PSID, through Amendment 7, including Sections 1.3, 3.2, 4.1.4, 5.5, 6.1, 6.2, 8.10, and 13.3, and DOE responses to staff comments and requests for additional information. The staff also reviewed the discussion of Generic Issue A-29 provided in the DOE report, "MHTGR Assessment of NRC LWR Generic Safety Issues" (DOE, 1987-4).

In a letter transmitting Amendment 6 to the MHTGR PSID (Walker, 1987), DOE stated that it would limit responses to requests for additional information to those that would not require restricted safeguards information. Details of the security system would not normally be provided until submittal of security, contingency, and guard-training plans that would accompany a formal operating-license application, as required by 10 CFR 50.34 and 73.55. Nevertheless, where security details have been included in the PSID, they have been reviewed and evaluated. The need for additional information at some later design stage is also noted herein.

This review did not include an assessment of the proposed design's compliance with any applicable material control and accountability requirements. Such review may be necessary before licensing. Since the MHTGR fuel is uranium enriched to less than 20 percent U-235, which is not considered by NRC to have strategic significance, no licensing problems in this area are anticipated.

This safeguards and security review has been performed using criteria developed for light-water reactors. At a later review stage, the staff may have developed specific guidance for advanced reactors that would then be applied.

13.3.2 Design Description and Evaluation

Two separate but adjacent security areas, one for the energy-conversion area (ECA) and one for the nuclear island, will be in the owner-controlled area. The ECA contains power-conversion structures and equipment (including the turbine building, intake pumphouse and discharge structure, cooling tower basin and circulating-water pumphouse) and the operations center building. The operations center is proposed to include the reactor control room as well as security access control points and alarm stations. The inclusion of the reactor control room within the ECA portion of the operations center has been found unacceptable by the staff as discussed in Section 6.1.3. The ECA provides ordinary industrial-level security, with unalarmed physical barriers to channel cooperative individuals to access points.

Reactor systems and equipment containing radionuclides (including the reactor vessel, steam generators, and spent-fuel storage pools) are located in the nuclear island (NI), with each of four reactor modules housed in its own below-grade reinforced-concrete structure. Steam and feedwater tunnels and electrical cabling for instrumentation and control connect between the NI and the ECA. The NI security program consists of a nuclear-level physical security organization, a protected area, one or more vital areas within the protected area, physical barriers, controlled access points, detection aids, communication capabilities, a testing and maintenance program, and an armed response force.

In this review, no credit is given to ECA security or to plant equipment located outside the NI, under the assumption that these would be vulnerable to a threat given the capabilities defined in 10 CFR 73.1.

13.3.2.1 Physical Security Organization

A description of the NI physical security organization was not provided with the PSID. The PSID included, however, requirements for the physical security organization that essentially paraphrased the requirements in 10 CFR 73.55(b).

13.3.2.2 Physical Barriers

A description of the NI physical barriers was not provided with the PSID. The PSID included requirements for the physical barriers that essentially paraphrased the requirements in 10 CFR 73.55(c).

Protected Area

The PSID identifies the operations center and the NI warehouse buildings as part of the ECA, rather than part of the NI. Enclosure 4 of the staff's letter to DOE (Morris, 1987) noted that these buildings contain portions of the boundary between the less secure ECA and the NI protected area, and thus portions of those buildings should be considered to be in the protected area. Attention will need to be given to the physical barriers and intrusion detection systems at those boundaries, as well as the main steam and feedwater piping tunnel boundary.

Vital Areas

Enclosure 4 of the staff's letter also asked for the identification of systems and components, including piping runs and motor control centers, that would be considered vital in the sense of 10 CFR 73.2(i). DOE declined to present this safeguards information at this time. Based on the criteria provided in safeguards Review Guideline 17, "Definition of Vital Areas," protection of seismic Category I equipment as vital would be sufficient to protect against radiological sabotage. All this equipment is located within the NI protected area. Since the staff has determined that the control room must be in the nuclear island and classified as a vital area, barriers for the control room are required to be bullet resistant.

For vital equipment in the reactor building, the PSID requires access to be through doors or hatches that will be alarmed and have locks of substantial construction to offer penetration resistance and to impede both surreptitious and forced entry. Consideration may also need to be given to ensuring that vital equipment cannot be disabled from outside the vital area containing that equipment. Of possible concern in this regard are the reactor cavity cooling system (RCCS) air inlet and outlet structures. This concern is mitigated because the system is designed with multiple inlet and outlet ports and interconnected parallel flow paths to permit cooling even if any single duct or opening becomes blocked. No data have been presented, however, to show that sabotage of the RCCS in conjunction with other decay heat removal systems and inducement of a loss-of-offsite-power transient is beyond the capabilities attributed to the sabotage design-basis threat.

The conceptual design plot plan (see Figure 6.1) includes two fences surrounding the NI protected area. In Comment 13-10, the staff asked for confirmation that one of these two fences was not intended to be the vital-area barrier. The response from DOE did not preclude the use of fences as vital-area barriers. This would be unacceptable. The NRC regulatory position, as presented in Regulatory Guide 5.65, is that:

> ...access to vital areas requires passage through at least two physical barriers of sufficient strength to meet the performance

> requirements of 10 CFR 73.55(a). Accordingly, no accessible openings in vital areas should exist....[The vital-area] barrier should be constructed of materials that provide delay to forced entry. Such material should be resistant to cutting....

In addition, Review Guideline 17, incorporated by reference into the Standard Review Plan, defines a vital area as:

> Vital area means any area which contains vital equipment within a structure, the walls, roof, and floor of which constitute physical barriers of construction at least as substantial as walls described in [10 CFR 73.2] (f)(2).

Besides reactor equipment, onsite secondary power supplies for security equipment are required to be protected as vital by 10 CFR 73.55(e). In response to staff Comment 13-9, DOE stated that the dedicated security backup generator and the dedicated security uninterruptible power supply will be located within vital areas within the protected area. Although the PSID shows the emergency power source to be in the operations center, which is listed as being in the ECA, it is reasonable to consider that the portion of the operations center on the NI side of the access-control portal will be in the protected area. Since 10 CFR 73.55(c) requires passage through at least two physical barriers of sufficient strength to prevent access to vital equipment, the adequacy of the barriers to this equipment must still be determined.

13.3.2.3 Access Requirements

A description of the NI access controls was not provided with the PSID. The PSID included, however, access requirements that essentially paraphrased the requirements in 10 CFR 73.55(d).

13.3.2.4 Detection Aids

A description of the NI detection aids was not provided with the PSID. The PSID included, however, requirements for the detection aids that essentially paraphrased the requirements in 10 CFR 73.55(e).

These requirements include containing the central alarm station within a bullet-resistant structure located within a building in such a manner that its interior is not visible from the perimeter of the protected area. They also include the requirement that the central and secondary alarm stations be located so that a single act cannot eliminate the capability of calling for assistance or otherwise responding to an alarm. Locating all security services, including both these alarm stations, the arms room, and the security-force ready room, in the operations center building, as described in PSID Section 6.2.7, may not be compatible with that requirement.

The plant security system is supplied from normal ac power sources backed up by the station backup generators and a dedicated security backup generator. The electronics portions of the plant security system are supported by a dedicated uninterruptible power supply (UPS), which is also backed up by the station backup generators and the dedicated security backup generator. In response to staff Comment 13-9, DOE stated that the dedicated security backup generator and the dedicated security UPS will be located within vital areas within the protected area. Exterior lighting needed for security-alarm assessment is supported by the dedicated security diesel generator. Additional commitments may be needed, however, since NRC report NUREG/CR-1327 states:

> Generators cannot start fast enough and switch into the power grid to avoid a momentary void in current flow....The isolation zone can tolerate illumination lapses of up to 10 seconds. Therefore, either instant-start luminaires with a 10-second start time for the generator or an UPS with any luminaire is appropriate.

13.3.2.5 Communications

A description of security communications was not provided with the PSID. The PSID included, however, requirements for the security communications that essentially paraphrased the requirements in 10 CFR 73.55(f).

13.3.2.6 Test and Maintenance Requirements

A description of physical-security test and maintenance was not provided with the PSID. The PSID included, however, requirements for the NI physical-security test and maintenance that essentially paraphrased the requirements in 10 CFR 73.55(g).

13.3.2.7 Response Requirements

A description of the armed response force was not provided with the PSID. The PSID included, however, NI response requirements that essentially paraphrased the requirements in 10 CFR 73.55(h).

13.3.2.8 Employee-Screening Program

A description of the NI employee-screening program was not provided with the PSID. This would not be a factor in licensability of the MHTGR as a standard design.

13.3.2.9 Severe Accident Policy Considerations and Dependency on Physical Security System

In response to staff Comment 13-5, DOE described the design features that would make the MHTGR more inherently safe from radiological sabotage and less dependent on physical security for protection against such sabotage. Although DOE has not established MHTGR design criteria for protection against radiological sabotage, the inherent safety features of the MHTGR design provide advantages in protection against insiders and outsiders as compared with a current-generation LWR.

For protection against induced reactor transients, DOE cited the large negative temperature coefficient, the high-temperature stability of the fuel, the low power density, and the slow heatup rate of the graphite core. For decay-heat removal, there are a number of redundant systems, one of which, the reactor cavity cooling system, is a passive system that would be difficult for insider or outsider saboteurs, given the capabilities assumed in 10 CFR 73.1, to totally disable. Although there are some uncertainties about vessel peak temperatures and structural temperature units, none of the severe transients considered are expected to result in releases approaching 10 CFR Part 100 levels. In addition, because of the high-temperature stability of the reactor fuel and core geometry and the slow heatup rate of the massive graphite core, time on the order of days is available to take corrective action, if needed.

A potential disadvantage of the MHTGR is that its reactor building might not be as inherently resistant to forced penetration as a conventional pressurized-water-reactor (PWR) containment vessel. Since the MHTGR reactor areas of the reactor building are below ground, however, and it is anticipated that vent openings can be designed with barriers as necessary to detect and delay attempted penetrations, this may not be an exploitable disadvantage, depending on whether the access portals, including any provided for RCCS inspection, are as resistant to forced penetration as are PWR containment hatches.

13.3.3 Conclusions

The staff concludes that, despite a lack of detail regarding the security system in the PSID, there is nothing fundamental to the MHTGR conceptual design that would prevent compliance with the provisions of 10 CFR Part 73. However, although MHTGR safeguards against radiological sabotage are in an acceptable stage of development for a conceptual review, some changes to the conceptual plant layout could be necessary. For example, as stated in Section 6.1.2, the staff has determined that the operators in the control room must be protected. Thus, the control room is to be located within the NI protected area, rather than in the less secure ECA, and will have to have bullet-resistant barriers to protect the operators in it. The concentration of all security alarm stations, equipment, and personnel in one location (that is, the operations center) may need to be changed so as to ensure that a single adversary action could not negate the security force's effectiveness and that the plant's armed response force would be in a position to interpose itself between the adversary and the four reactor modules' vital equipment with greater confidence. Additional attention may also need to be given to ensuring that an attacker could not easily eliminate security lighting. Completion of a safeguards information security and contingency plan will be required at a future stage of review.

DOE declined to discuss at this conceptual stage the potential vulnerabilities of the MHTGR to radiological sabotage, which limited the staff's review of the extent to which the issues of insider and outsider sabotage threats have been emphasized in the design and operating procedures for the MHTGR, as called for by the Severe Accident Policy Statement. As discussed above, the MHTGR may have inherent advantages compared with current LWRs with respect to insider sabotage, because of the passive nature of its decay-heat-removal concept and the long times available to assess damage and take mitigation and recovery actions. Although there are some uncertainties about vessel peak temperatures and structural temperature limits, none of the severe transients considered are expected to result in releases approaching 10 CFR Part 100 levels. Further study and the review of the possible spectrum and consequences of deliberate errors by operators or maintenance personnel will be required at later review stages. With respect to outsider sabotage, the principal issue to resolve involves the potential for sabotage associated with the RCCS vents.

Table 13.1 Emergency-preparedness requirements for offsite response

Current offsite plans	MHTGR offsite plans
(1) 10-mile plume emergency planning zone (EPZ) and 50-mile ingestion EPZ	50-mile ingestion EPZ only
(2) Notification of the public within about 15 minutes	No requirement
(3) Dose-projection and assessment capabilities for plume and ingestion EPZs	Dose-projection and assessment capabilities for ingestion EPZ only
(4) Offsite monitoring for plume and ingestion EPZs	Offsite monitoring for ingestion EPZ only
(5) Arrangements for medical services	No requirement
(6) Primary and backup communications	Primary and backup communications retained
(7) Training required	For monitoring equipment
(8) Biennial exercises required, including periodic ingestion EPZ exercises	Ingestion EPZ exercises only

14 PROTOTYPE-PLANT TESTING

The staff's criteria for prototype-plant testing and the bases for these criteria are given in Section 3.2.3.3. For the MHTGR, DOE proposed a demonstration plant at an unspecified, typical utility site. The DOE-proposed demonstration plant would have as its primary purpose commercial demonstration with initial startup data to be used to confirm specific aspects of design. The DOE proposal does not call for tests of specific safety transients at this time. However, based on judgments of the adequacy of existing operating experience, the novel design features proposed, and the status of the present technology base, the staff requires that testing and operation of a prototype test reactor, located at an isolated site, be mandatory before design certification. Table 14.1 gives the areas where the staff is considering plant testing, including transients, to verify the safety of the MHTGR in order to support licensing. The testing program would not intentionally risk damage to the plant, such as elevating reactor-vessel temperatures into the service level C domain.

Most of these tests would be performed as part of the initial startup test program, although the reactor physics parameters would need to be measured at critical stages of the fuel cycle, including the equilibrium core. The prototype test would also be expected to verify many other important design and safety features not included in Table 14.1. In addition to the specialized tests, the MHTGR will be required to meet the intent of Regulatory Guide 1.68, "Initial Test Programs for Water-Cooled Reactor Power Plants." DOE did not provide information on plant testing in the Preliminary Safety Information Document or supporting documents. In response to staff needs, DOE will submit for NRC review a proposal for the conduct of the tests that may be required. This proposal should describe plans for plant testing that will include items pertaining to the research program and design confirmation, as well as descriptions of tests relating to plant safety operations. The staff requires that commitment to a satisfactory prototype testing program be a condition for final design approval and that the program be satisfactorily completed before design certification.

At present the staff envisions the minimum facility for a prototype test to be a single module, with associated instrumentation and controls and other systems important to safety at least through and including the steam generator. It would be built to the same standards and design as the plant to be certified; if it did not include the whole plant, it would have to simulate interface requirements. Some special instrumentation and test features may be required. The purpose of the test program would be to verify the analysis of the plant response to important plant transients and to generate sufficient experimental data to verify the analytical tools used for the safety analysis, particularly for the bounding events. The test program would be directed toward internal events and conducted in a stepwise fashion from low power and low decay-heat conditions to higher power and higher decay-heat conditions and, in some cases, from fresh-core to equilibrium-core conditions. To provide investment protection during certain tests and to provide greater confidence that damage regimes would not be entered, DOE may decide to provide auxiliary and backup systems that would not be normally required in the commercial design following successful prototype testing (for example, high-reliability diesel generators).

The specific tests required to support the MHTGR design will be determined during the review of the application for final design approval (FDA). In addition, the FDA will identify the specific testing required to support the licensing of an individual plant or class of plants referencing that FDA.

Table 14.1 Areas considered by the staff for plant testing

Topic	Major areas of interest	Reference sections
Reactor physics	Negative temperature coefficient (anticipated transient without scram), decay-heat rate, shutdown margins, water-ingress simulation	4.3.4, 3.3.5.1
Invessel-flow distribution and vibration testing	Confirmation of results of modeling tests	4.4.4
Reactor cavity cooling system (RCCS)	Full and partial loss of air flow for pressurized and depressurized cases; heat transmission to earth; temperature measurements of fuel and reactor-vessel cavity; demonstrated response to steamline-break environment	5.5.4
Various configurations of active heat removal systems	Response of fuel, reactor vessel, and cavity temperatures to temporary loss and restart of heat transport system and shutdown cooling system	5.2.5.C
Fission-product retention in reactor cavity	Depressurization rate, environmental conditions, usefulness of filters	6.2.5.A

15 SAFETY ANALYSES

15.1 Introduction

15.1.1 Scope and Objectives

DOE's reactor-safety analyses presented in Chapter 15 of the Preliminary Safety Information Document (PSID), in the Emergency Planning Basis (EPB) Report, in the Probabilistic Risk Assessment (PRA), and in response to the staff's comments and requests for additional information are reviewed and evaluated in this chapter. Offsite radionuclide releases and the siting-source-term (SST) selection and use are discussed in relation to staff-defined bounding events and the radiological dose guidelines given in 10 CFR Part 100, the protective action guidelines (PAGs) of the U.S. Environmental Protection Agency (EPA, 1980), and the guidelines of the NRC Safety Goal Policy (51 FR 28044). Discussions are presented regarding uncertainties, phenomenologies, and margins on the basis of the time-history evaluations over the event sequences. Evaluations of the four key policy issues - selection of events that must be considered in the design, siting-source-term selection and use, adequacy of the containment concept, and adequacy of offsite emergency planning - are made in accordance with an NRC policy issue paper, "Key Licensing Issues Associated With DOE Sponsored Advanced Reactor Designs," SECY 88-203 (July 15, 1988), that guided the overall approach to the safety analyses, including considerations for defense-in-depth.

This chapter also presents summaries of the staff's review of the PRA and independent analyses of selected safety issues performed by NRC contractors at the Oak Ridge National Laboratory (ORNL) and the Brookhaven National Laboratory (BNL). As stated in Chapter 11, evaluations of accidents involving releases from radionuclide inventories other than those contained in the reactor and the primary coolant system will be performed at a later review stage because such evaluations would not have an influence on the MHTGR's principal issues of feasibility. The staff recognizes that such accidents will have to be satisfactorily addressed at a later review stage and may affect overall plant safety. The staff's assessment of the MHTGR's ability to meet the dose and risk guidelines is presented in the form of concluding statements for this phase of the MHTGR review in Section 15.6.

15.1.2 Background

In performing the reactor-safety analyses, DOE and the staff developed a spectrum of event initiators and sequences from PRA considerations and engineering judgments that included the operating histories of light-water and gas-cooled reactors, earlier accident studies characteristic of the HTGR type of gas-cooled reactor, and the MHTGR's emphasis on prevention versus mitigation of accidents. The unique passive and inherent safety features proposed by DOE and being reviewed by the staff are, in summary:

(1) The fuel is to be considered highly stable, with essentially all fission products retained within the coated particles. Offsite doses would result mainly from releases of the "liftoff" of plated-out radioactive species within the primary coolant system and radionuclides circulating with the

helium coolant. Significant fuel failure would occur only at temperatures that substantially exceed 1600°C, which are prevented by the design of the safety systems.

(2) An effectively large negative reactivity coefficient will exist over all temperature ranges and at all points of the fuel cycle. This coefficient would not be materially altered by loss of the helium coolant or the ingress of water or air into the reactor.

(3) The small thermal rating and the core geometry will permit decay heat to be removed passively at adequate rates to ensure fuel and reactor-vessel integrity by conduction through the core and reflector to the reactor vessel and then from the reactor-vessel surface to a surrounding heat sink of passive design.

(4) No human actions will be needed to mitigate any postulated event sequence because of the full automation of the plant safety response and the passive and inherent safety features of the design.

The assignments of top-level regulatory criteria made by DOE for use in its safety analyses are identified in Figure 15.1, where the mean frequencies of event occurrences per plant-year are plotted against consequences in terms of whole-body gamma doses. Events are grouped into decreasing regions of frequency, which are titled "anticipated operational occurrences," "design bases," and "emergency planning." These regions are similar but not exactly equivalent to event categories I, II, and III described in Section 3.2.2.1. Figure 15.1 is a summary of the results of DOE's safety analysis, and most events plotted are discussed later in this chapter. It should be noted at this time that all the dose consequences reported by DOE are well below the PAG doses for sheltering of 1 and 5 rem for whole-body and thyroid doses, respectively, at the exclusion area boundary (EAB). DOE cites these analyses in support of its proposal not to require emergency planning for offsite evacuation or sheltering. At the next review stage it is anticipated that DOE will present its safety analysis in direct correspondence with the event-category nomenclature.

Consideration and evaluation of all postulated events in event categories II and III are needed for establishing a siting source term, siting criteria, operator training, emergency plans acceptable to the staff (including the size of the emergency planning zone), determining the adequacy of both accident-prevention and -mitigation systems, and compliance with Commission policies for severe accidents and safety goals. The choice and evaluation of the accident spectrum must include prudent assessment of uncertainties in plant design and operation, as well as in the safety analyses. Because many safety conclusions for the MHTGR must be developed from first principles, the major sources of uncertainties that must be considered are (1) incomplete or misunderstood basic physical and chemical phenomena, (2) modeling and analysis methodology, (3) materials selections and performance, (4) design errors, (5) human performance, and (6) quality assurance deficiencies. Furthermore, an additional degree of conservatism is warranted to account for the shift in emphasis from accident mitigation to accident protection and prevention and to account for the large degree of reliance on non-safety-grade equipment. A further discussion of uncertainties and their effects on the review conclusions is given in Section 15.3, "Probabilistic Risk Assessment."

15.1.3 General Approach

In Section 15.1 of the PSID, DOE described its overall approach to its safety analyses. This included a brief description of its methods, assumptions, and computer codes. In this description, DOE outlined the background supporting its analytical methodology, particularly the computer codes referenced. Most of these codes were developed before the MHTGR project, in connection with design work performed on the various large HTGR projects undertaken during the 1970's. Figure 15.2, adopted from the PSID, summarizes the DOE approach and identifies the accident categories and the computer codes relevant to its analyses and release calculations. The major codes used are SORS (fission-product release from fuel at elevated temperatures), RATSAM (transient pressures and flows, including liftoff phenomena), OXIDE (chemical reactions with water), and TDAC (release from the vessel and, presumably, the reactor cavity).

The staff and its consultants have reviewed the overall approach to the MHTGR safety analyses for events in the design-basis region and have found the DOE approach to be adequate and, in principle, sufficient to account for this class of events for all credible mechanisms for fission-product release and transport and for the calculation of releases and dose estimates. The staff and its consultants, however, have not reviewed the individual computer-code modeling assumptions and input data and the information supplied by DOE to support their validity. Rather, as described in Section 15.4 and Appendixes A and B, the staff's contractors have performed independent analyses that address areas in the safety analyses selected to be the most indicative of the MHTGR safety characteristics and its ultimate success in achieving the safety performance goals described in the PSID. The applicability of the contractors' analyses is limited, however, to a gross assessment of selected aspects of the bounding events listed in Table 3.7. As discussed in Sections 15.3 and 15.4, essentially all the contractor work pertained to events judged to fall below frequencies of 10^{-4}, or within or below the region entitled by DOE as "emergency-planning basis."

At the time of the construction-permit review, much additional effort will be required by both the applicant and staff in safety analysis. Particularly, detailed calculations will be needed with regard to external events, potential structural failures, and the transport of radionuclides from the vessel system and the reactor building. New information forthcoming from the Regulatory Technology Development Plan will require review and assessment. In addition, the conclusions of this staff review will be reevaluated and DOE's overall approach to the safety analyses will be reviewed again. Before a construction permit can be issued, it must be demonstrated that the safety analysis is comprehensive and sufficient and that all models that pertain to safety are adequate and supportable from appropriate phenomenological experience.

15.2 Accidents Considered

15.2.1 Anticipated Operational Occurrences

DOE considers anticipated operational occurrences (AOOs) as events that are expected to occur one or more times during the lifetime of the plant and that have frequencies of 2×10^{-2} per plant-year or more. They were analyzed to demonstrate compliance with 10 CFR Part 50, Appendix I, and 40 CFR Part 190.

At the present stage of review, the staff believes that AOOs are sufficiently bounded by the lower frequency events that are considered below. The proposed AOOs will be reviewed in detail at a later stage of review.

15.2.2 Licensing-Basis Events

The term "licensing-basis events (LBEs)" is used by DOE to include events within the "design-basis" region; that is, events with frequencies ranging from 2×10^{-2} down to 10^{-4} per plant-year. Events in this frequency range would be expected to occur in the lifetime of a population of plants and are similar to event category II. They were analyzed conservatively by DOE to a level of 95-percent confidence and were to be selected by engineering judgment complemented by PRA. Safety-related plant design features are to be provided by the design to prevent exceeding PAG dose levels at the exclusion area boundary.

The PSID discusses two types of events in this category. The first type, identified as design-basis events (DBEs), permits some availability and performance of normally operating or standby equipment regardless of its quality rating. These DBEs are listed in Table 3.2. The discussion of this type of event illustrates the full potential of the plant to respond to the list of postulated accident-initiating events and identifies the most probable plant response if one of these postulated events actually occurred. The second type of event corresponds to the initiating event for each of the DBEs and is identified by DOE as a safety-related design-condition (SRDC) event. The postulated SRDC events refer only to the availability and performance of safety-related equipment in the accident progression and consequence evaluation. These events require the performance of the safety-related equipment and are used by DOE to illustrate that this equipment alone can prevent radioactive release above the required limits. Table 15.1 presents both the DBEs and SRDC events. This table highlights the differences in sequences for the DBEs and SRDCs for a given initiating event and provides a column for comments that characterize both the DBEs and SRDCs in terms of the dose consequences, or the peak fuel temperatures when there are no offsite releases.

The first five events are known as "pressurized-conduction-cooldown events" that would have the following initiators: (1) loss of all ac power, (2) loss of main heat transport system (HTS) cooling followed by failure to trip (an anticipated transient without scram [ATWS] event), (3) control-rod-group withdrawal followed by the loss of the HTS, (4) rod-group withdrawal with the loss of both the HTS and the shutdown cooling system (SCS), and (5) safe-shutdown earthquake with loss of both the HTS and the SCS. For the SRDCs, removal of decay heat and the subsequent core cooldown are always performed by the reactor cavity cooling system (RCCS), while for some of the DBEs, decay-heat removal is achieved by the SCS. For all events described as "pressurized," there are no offsite doses, since the primary coolant boundary remains intact. Consequences are summarized in terms of peak core temperatures.

Events 6 through 11 are depressurized conduction cooldowns and, since the primary coolant boundary is violated, all SRDCs and some DBEs result in small offsite doses based on circulating radioactivity and the liftoff of plated-out fission products in the primary system. Events 6 through 9 describe various cases of steam generator tube leaks and equipment failures. Events 10 and 11 pertain to primary-system leaks from the steam generator vessel and reactor vessel, respectively. Event 10 analyzes a leak area corresponding to a rupture of the primary system pressure relief line. Event 11 corresponds to a ruptured instrument line. Although the PSID discusses consequences from moisture and air ingress for these events, these phenomena are sufficiently understood and bounded by lower frequency and more severe events so that a discussion of the nature and consequences of chemical attack is deferred to Section 15.2.4.

15.2.3 Events of Lower Frequency Than Licensing-Basis Events

Events in this frequency region have been studied by three different approaches: the emergency-planning-basis events (EPBEs) described in the Emergency Planning Basis (EPB) Report, the beyond-licensing-basis events (BLBEs) described in Appendix G to the PRA, and the bounding events (BEs) developed by the staff and listed in Table 3.7. These events and their safety-analysis approaches are considered to generally correspond to event category III and are discussed in the following three subsections.

15.2.3.1 Emergency-Planning-Basis Events Proposed by DOE

DOE proposed the emergency-planning-basis-event (EPBE) classification to identify accidents that would form the basis for emergency planning. The events and the emergency-planning basis are described in the EPB Report. DOE selected these events on the basis of PRA and estimated frequencies for EPBEs that range from 10^{-4} to 5×10^{-7} per plant-year, as shown in Figure 15.1. Their prevention and mitigation are based on the same safety-related design features and equipment as those provided for SRDC events. Both best-estimate methods (50-percent confidence level) and conservative methods (95-percent confidence level) were used for dose calculations.

In a manner similar to that for the LBE sequences, the three EPBEs are summarized in Table 15.2. DOE also considers DBE -7, -10, and -11 as EPBEs because they involve offsite doses, although these have been discussed already as being included in the LBEs. The staff has not reviewed or attempted to calculate independently the doses presented in the EPB Report in order to concentrate efforts on the BE studies described in Section 15.2.3.3.

15.2.3.2 Beyond-Licensing-Basis Events Proposed by DOE

DOE provided, in Appendix G to the PRA, "Assessments of Events Beyond the Licensing Basis," a means to address the potential consequences and risks of events it considers would occur with frequencies below the emergency planning zone limit in Figure 15.1; that is, less than 5×10^{-7}. Although the staff agrees that the probabilities of occurrence for events of this type are extremely low, it views the investigations of these low-frequency events as a necessary deterministic adjunct to the PRA to account for uncertainties and to recognize the importance of engineering judgment in final decisionmaking with respect to the key policy issues. The staff considers the Appendix G events to be within event category III. As for the EPBEs, the staff has not reviewed or attempted to calculate independently the doses presented in Appendix G in order to concentrate efforts on the bounding events discussed below.

15.2.3.3 Bounding Events Postulated by the Staff*

The staff developed the bounding events (BEs) from five separate general considerations: (1) events that have occurred or nearly occurred over the entire history of nuclear reactor technology; (2) events that take into account the MHTGR's design emphasis on accident prevention as opposed to mitigation, thus emphasizing the importance of the uncertainties affecting the successful functioning of these features; (3) events that test the MHTGR's passive and inherent safety features; (4) events that assume worst-case failure of non-safety-related systems; and (5) recognition that time will be available to permit recovery from initiating events and repair of vital systems.

The BEs address the following categories of events: reactivity additions; reactivity-insertion failures; station blackout; heat-removal failures; loss of coolant, including rapid depressurization; chemical attack from air, water, and steam; and external events (earthquake, flood, fire, wind, sabotage, and aircraft impacts) in a manner consistent with external events imposed on light-water reactors (LWRs). The list of the BEs is provided in Table 3.7.

The BE approach complies with guidance contained in the Commission's Safety Goal Policy Statement and accounts for uncertainties and differences from LWRs in event probabilities, equipment performance, and human factors. DOE was directed to assume failure of non-safety-related equipment (either as an initiator or in response to the initiating event) in a way that exacerbates the accident to the maximum degree physically possible, unless a lesser degree can be justified. This would account for uncertainties resulting from the use of commercial-grade procurement and construction and the lack of NRC inspection of and technical specifications on this equipment.

In recognition of the unique differences in MHTGR safety systems, the following human-factors aspects were considered in event-sequence-progression analyses.

(1) Time is available to permit recovery from initiating events if no plant damage has occurred (ATWS, station blackout, loss of all cooling, reactor cavity cooling system failure). In consideration of emergency-planning requirements as discussed in Section 13.1, credit for recovery actions is given if adequate recovery can be achieved within 36 hours after event initiation.

(2) The passive and inherent safety characteristics and the full automation of plant safety response substantially reduce the probability of human errors of omission or commission by operators or management during the event sequence.

DOE performed its analyses of the BEs on a best-estimate basis consistent with the above considerations. The general conclusion from DOE's analyses is that none of the BEs result in fuel-particle failure and that the fission-product retention of the intact fuel is sufficient to control radionuclide releases to

*Indicates statement is particularly sensitive to change by evaluation of forthcoming DOE information.

meet the protective action guidelines (PAGs) at the exclusion area boundary (EAB). DOE summarized its analysis of the BEs in a table, which is adapted in Table 15.3. This table shows, for each BE, the assessed frequency per plant-year and the thyroid and whole-body doses at the EAB at 36 hours and 30 days. For all the BEs, both thyroid and whole-body doses are seen to be very small and well within the PAGs for both the 36-hour and 30-day cases. For the related cases discussed in Appendix G to the PRA, the results are generally similar. The staff judges that these results show that the MHTGR has the potential to cope with extremely rare and severe events without the release of a significant amount of fission products. The staff believes also that at the conceptual review stage the selected BEs are sufficient to illustrate the safety behavior of the MHTGR for low-probability events. At a later review stage, the selection and details of the BEs will be reconsidered on the basis of improved knowledge of the MHTGR design, research findings, and expected improvements in PRA. At this later stage, the staff plans that its consultants will perform independent confirmatory studies.

With regard to BE-6, "Severe external events consistent with those imposed on LWR," the staff deferred review of the seismic integrity of the unique structural features of the MHTGR. Further, the staff also has not reviewed the effects of additional external events that could be postulated. The staff believes, however, that the MHTGR can be satisfactorily designed to protect against such events and thus the above judgment is not affected by the staff's deferral of the review of severe external events.

15.2.4 Residual Risks*

On the basis of presumed successful research and testing programs, information presented by DOE with respect to BEs, and the safety analyses performed thus far by the staff and its consultants, credible events that would exceed the PAG doses could not be identified. This result is in accord with the statement of the Advisory Committee on Reactor Safeguards (ACRS) (Kerr, 1988-2) given in Appendix C, page 4: "Neither the designers, the NRC staff, nor members of the ACRS have been able to postulate accident scenarios of reasonable credibility, for which an additional physical barrier to the release of fission products is required in order to provide adequate protection to the public." At a later review stage and in conjunction with updated PRA and research findings, and as further deterministic information becomes available, including more details on sabotage and external events, the staff will reconsider MHTGR risks. If any event or sequence appears to have a frequency of occurrence in the frequency range of about 10^{-7}, it will be examined from the standpoint of residual risk. It will be determined if this event or sequence should be placed in event category IV, as described in Section 3.2.2.1, or if some other action to preclude the occurrence, such as a design change, should be undertaken.

15.2.5 Integrity of Safety Systems

For the MHTGR to meet the PAG doses for the BEs as projected by DOE in Table 15.3, the integrity of certain key safety systems during thermal, structural, and/or

*Indicates statement is particularly sensitive to change by evaluation of forthcoming DOE information.

chemical attack must be ensured. These systems are the fuel particles, the core and core support system (especially the graphite portions), the reactor vessel, the reactor cavity cooling system, and the reactor cavity. Table 15.4 outlines for each of these systems the concerns and the review status and identifies the section in this SER where the safety system is discussed. For the systems listed, further information will be needed from DOE at a future stage of review. At that time resolution of the concerns will be based on the forthcoming additional information and the staff's assessment of the defense-in-depth requirements based on the overall safety performance capabilities of the MHTGR.

15.2.6 Chemical Attack

15.2.6.1 Combustible-Gas Generation

At temperatures greater than about 700°C, steam will react rapidly with graphite according to the reaction

$$C + H_2O = CO + H_2 - Q,$$

where the endothermic heat of reaction Q is 51,000 Btu/lb-mole of graphite. The concern is the generation of the combustible gases carbon monoxide and hydrogen and their subsequent combustion within the reactor building, along with the potential for the destruction of safety-related equipment and the aggravation of accident sequences in progress. Other concerns are the reaction of steam with fuel to enhance the release of fission products or the degradation of safety-related graphite structures by oxidation. These are discussed in Sections 4.2.5.I and 4.5.5.D, respectively.

In the PSID, DOE considered the evolution of combustible gases for licensing-basis events (LBEs) and, in particular, for DBE-7 (moderate moisture leakage without SCS cooling) in which 50 kilograms of graphite was estimated to have reacted as calculated by the OXIDE code. In response to Comment 15-2, DOE calculated for SRDC-6 (see Table 15.1) the releases of carbon monoxide and hydrogen from the relief valve to the steam generator cavity of the reactor building (values were estimated at about an order of magnitude greater than for DBE-7) and concluded that these releases did not constitute a combustion hazard.

Substantial further information and review are needed to resolve the combustible-gas concern. In particular, cases need to be studied where even more steam enters the reactor than in SRDC-7, the reactor depressurization occurs within the reactor cavity, and less favorable assumptions about combustible-gas concentrations are made than in the Comment 15-2 analysis. In addition, consequences of a combustible-gas burn or explosion should be explored with respect to safety-related equipment that could be damaged, such as the RCCS. These concerns and potentially others of a related nature will be addressed at a later review stage. The staff anticipates that the applicant will provide substantial additional information in these areas, and staff consultants will confirm the staff's review by performing independent calculations, which could include development from first principles of a new model for the graphite-steam reaction to confirm the OXIDE code. While combustible gas is a concern to be addressed fully during a later review stage, the staff believes the concern will be resolvable at that time and that it is not sufficient to affect any of the staff's conclusions presented herein.

15.2.6.2 Air Ingress and Graphite Fires

Graphite fires and significant releases of radiation occurred in the Windscale reactor in England in 1957 and during the course of the Chernobyl accident in the Soviet Union in 1986. These events led to substantial investigations into the nature of graphite fires in nuclear reactors and the conditions necessary to sustain combustion. Because of the availability of this information and the fact that the PSID and supporting material did not initially address the potential for graphite fires, the staff has elected to address this question partly on the basis of two recent documents developed under the auspices of NRC. These documents are NRC report NUREG-1251 (Draft), "Implications of the Accident at Chernobyl for Safety Regulation of Commercial Nuclear Power Plants in the United States," and NRC report NUREG/CR-4981, "A Safety Assessment of the Use of Graphite in Nuclear Reactors Licensed by the U.S. NRC."

In NRC report NUREG-1251, Chapter 6, "Graphite-Moderated Reactors," the emphasis is on Fort St. Vrain concerns, although the MHTGR is discussed as well. At the time of the Fort St. Vrain construction-permit review in the late 1960's, graphite fires were considered and determined to be credible only if a chimney effect could be established through the core, such as could be caused by penetration failures at both the top and bottom of the prestressed-concrete reactor vessel (PCRV). Consequently, NRC report NUREG-1251 states, in a section entitled "Fires and Explosions":

> The staff has reached the conclusion that the use of a helium coolant, the overall negative reactivity coefficient, completely diverse alternative shutdown and cooling systems, and the protection offered by the PCRV against reactor fires, internal postulated explosions, and fission-product release to the environs remove Fort St. Vrain from any vulnerability characteristic of the Chernobyl design. In assessing the potential for a graphite fire, the licensee was asked to consider the highly improbable simultaneous failures of penetrations both at the top and bottom of the PCRV which would cause a chimney effect for sustained air ingress. Although the staff believes that the occurrence of such an event is extremely improbable, it agreed with the licensee that if the need arose, the reactor building could be flooded with water to a level sufficient to defeat the chimney effect and subsequently terminate the fire.

The staff now believes that a similar conclusion can be reached for the MHTGR (see below) at a later review stage and it should not be necessary to envision the extreme of flooding the reactor cavity.

NRC report NUREG/CR-4981, in its detailed review of the extensive literature and experimental evidence relating to graphite fires, confirms the early Fort St. Vrain finding that only a chimney-type geometry could cause sustained burning in a graphite reactor. Of particular relevance were graphite-burning experiments performed in 10-foot-long channels at the Brookhaven National Laboratory in 1962. Here it was found that conditions for self-sustained burning could be met if graphite were heated to at least 650°C, but "...it is necessary for a geometry to develop to maintain an adequate flow of oxidant and removal of combustion products from the reacting surface. Otherwise, the reaction ceases."

The report concludes:

> After review and analyses of existing information on graphite burning, stored energy accumulations and releases, and cause of the Windscale and Chernobyl accidents, we have concluded that the above phenomena are sufficiently well understood to allow the following evaluation of U.S. research reactors and Fort St. Vrain.
>
> The conclusions of these analyses are that the potential to initiate or maintain a graphite burning incident is essentially independent of the stored energy in the graphite and depends on other factors that are unique for each research reactor and for Fort St. Vrain. However, in order to have self-sustained rapid graphite oxidation in any of these reactors certain necessary conditions of geometry, temperature, oxygen supply, reaction product removal and favorable heat balance must exist.
>
> The reactors considered in this review have all undergone safety evaluations and have been granted operating licenses by the NRC. There is no new evidence associated with the analyses of either the Windscale Accident or the Chernobyl Accident that indicates a credible potential for a graphite burning accident in any of the reactors considered in this review. Nor is there any new evidence that suggests that detailed case-by-case safety analyses of the role of graphite in NRC licensed reactors are warranted.

In addition to the chimney geometry, the potential for a graphite fire as a result of a guillotine-type, double-ended failure of the MHTGR's crossduct was analyzed independently by the staff's contractor at BNL. The results are reported in Section 15.4 and Appendix B. Briefly, BNL found for this geometry that natural-convection forces were insufficient to develop flow rates that would sustain a graphite fire. A similar result can be deduced from an analysis performed by DOE in PRA Appendix G.4 and in response to Comment 15-10. In these studies, crossduct failure was reported to result in slow oxidation that totaled less than 4 percent after 800 hours of continued air ingress. DOE stated that such time would be sufficient to terminate air ingress before significant structural damage to the core occurred and that the mean thyroid and whole-body 30-day dose at the EAB would not exceed the PAG sheltering limits. Doses were also reported by DOE not to exceed PAG limits when openings were postulated at the bottom and top of the reactor vessel due to resistance to flow through the long, narrow passages of the annular core and the assumption that ad hoc actions would be taken to stop graphite oxidation after about 72 hours.

In spite of the experience and logic available with respect to graphite-fire concerns, the staff has concluded that while a graphite fire leading to significant offsite radioactive release is an event of very low probability, an upper frequency limit for occurrence of such an event should be established and the nature of the ad hoc actions indicated by DOE should be defined. An objective of this study would be to determine whether graphite fires should be considered in event category III or IV, or whether they are below regulatory concern. The study should develop further information on in-reactor convection flows, constraints on air supply offered by the reactor cavity, and vessel-failure modes conducive to graphite fires. Of central importance is an improved understanding of the thermal and structural conditions to be experienced by the reactor vessel and the reactor cavity during the course of bounding events that could be conducive to graphite fires.

15.3 Probabilistic Risk Assessment

15.3.1 Basis and Specific Objectives of Probabilistic Risk Assessment

In accordance with NRC's Severe Accident Policy Statement, DOE's licensing plan for its standard MHTGR provides for the development at each licensing stage of a probabilistic risk assessment (PRA) to demonstrate that this new reactor design is acceptable in terms of severe-accident concerns. The conditions contained in the policy statement explicitly require the performance of a PRA and the consideration of severe-accident vulnerabilities to ensure that there is no undue risk to public health and safety. The guidelines to be addressed by the PRA are given in NRC's Safety Goal Policy (51 FR 28044) and Advanced Reactor Policy Statements (51 FR 24643). The Advanced Reactor Policy Statement requires that, at a minimum, the plant have the same degree of public protection as is required for current-generation light-water reactors (LWRs); however, enhanced margins of safety compared with the current LWRs are to be expected. The policy statement further affirms that the degree of protection afforded the public will be judged on the basis of the plant's design capability and margin to prevent and mitigate severe accidents. Accordingly, the staff's review of the accuracy and completeness of the PRA submitted in support of the MHTGR was a major contributor to the staff's overall findings. In response to the regulatory requirement for the performance of a PRA, the following four programmatic objectives were identified by DOE as having to be addressed:

(1) Provide a means of characterizing the safety of the MHTGR such that the conceptual design can be evaluated in a logical fashion.

(2) Provide the basis from which to select the MHTGR licensing-basis events (LBEs) to be evaluated in the PSID.

(3) Evaluate a wide spectrum of events with offsite doses to show compliance with protective action guidelines (PAGs) at the site boundary.

(4) Evaluate the MHTGR risk to the public using the limits in NRC's Safety Goal Policy Statement. The guideline values stated in the policy statement and 10 CFR Part 100 were proposed by DOE as design limits. NRC has not yet approved these guideline values as such, but has reviewed the MHTGR PRA for compliance with these limits.

Section 15.3.5 provides the staff's judgments as to how well these objectives were met by the PRA at this conceptual review stage. In this connection, it should be noted that (1) Science Applications International Corporation, under contract to NRC, reviewed the MHTGR PRA and provided significant commentary that was incorporated in this evaluation (Minarick, 1988) and (2) the results of the PRA evaluation should be considered in conjunction with the staff's evaluation of the PSID before arriving at any final conclusions relating to such broad safety issues as the adequacy of the plant's defense-in-depth.

15.3.2 Methodology and Uncertainties

The methodology used in the MHTGR PRA consisted of the event-tree/fault-tree approach commonly used to define risk-related sequences, their frequencies, and the offsite consequences. The uncertainties usually associated with quantitative PRAs were further exacerbated in this PRA by the paucity of design details and component-failure rates available at this conceptual stage of the design, and the almost exclusive use of untested passive designs and equipment to perform ultimate safety and protection functions. Specific methodology details and their uncertainties for the various phases of the PRA follow; however, before presenting them, the paramount importance of the reliability of the passive decay heat removal system (reactor cavity cooling system [RCCS]) and the performance of the fuel at high temperatures to the ultimate safety of the MHTGR needs to be reviewed in the context of these discussions.

The very high reliability proposed by DOE for the RCCS (that is, 10^{-6} per reactor-year coupled with an emergency-planning-basis event-frequency cutoff of 5×10^{-7} per plant-year) was a significant factor in limiting the number and type of severe-accident sequences that appear within the envelope of risks to be considered by the MHTGR design. Furthermore, for those sequences shown to have frequencies within this cutoff limit, the fuel-failure temperature based on phenomenological analyses was never reached, and therefore the releasable radioactive sources were limited to very small combinations of circulating and liftoff radioactivity. Accordingly, the importance of establishing the credibility of the proposed design for both the RCCS's reliability and the fuel failure performance is obvious. This is further exemplified by the fact that such historically important nuclear reactor severe-accident events as station blackout and anticipated transient without scram (ATWS) either have no important sequences or negligible offsite risks for this plant. Furthermore, from a bottom-line risk perspective, the heat removal system and fuel-failure temperature parameters greatly influence (1) siting considerations, including the question of offsite emergency evacuation planning; (2) quality classifications for plant equipment involved in potential sequences (for example, non-safety-related heat transport system [HTS], shutdown cooling system [SCS], and diesel generators); and (3) the need for a conventional reactor containment building.

15.3.2.1 Initiator Selection

The PRA states that the accident initiators developed for the MHTGR were derived by using a logic diagram that identified the risk-critical safety functions (that is, control of heat generation, removal of core heat, and response to chemical attack) needed to maintain control of radioactivity releases. Using this information, the critical systems and structures performing these functions were identified so as to be able to determine the appropriate set of initiating events that challenge these critical safety functions. This approach, when coupled with data from other PRAs, is generally considered to be the preferred method for initiating a PRA. The staff review of this effort indicates that the set of initiators was appropriate but incomplete to describe the potential risk associated with the MHTGR. Additional initiators that should be reviewed and analyzed include system-level failures (that is, loss of a dc bus and loss of service water), internal fires, internal floods, and unexpected environmental conditions that might impact the RCCS. Although it is admittedly difficult to address all potentially identifiable initiators at this stage in the design, nevertheless an element of uncertainty is introduced in the credibility of using the PRA results in those important areas relating to siting, system and component quality classifications, and mitigation systems. The specific initiators identified and analyzed in the PRA are (1) ATWS, (2) control-rod-group withdrawal, (3) loss of offsite power, (4) primary coolant system leaks, (5) steam generator leaks, (6) earthquakes, and (7) loss of HTS cooling.

15.3.2.2 Fault-Tree Analysis

Standard fault-tree development techniques were followed as part of the MHTGR's system-reliability analysis. The limitations on plant specifications at this time placed some constraints on reviewing these system models. For example, the success criteria and normal configurations of the service water system and the circulating water system were not defined. Also, the ac power bus loads were not yet developed, and specific valve types were not specified (for example, the shutdown cooling water subsystem inlet and outlet valves). Thus, failure modes and data analyses were uncertain.

Some events were not defined explicitly enough to quantify properly. Common-mode and common-cause events were not present explicitly in the models. Human-failure events were too vaguely described to determine whether they were assumed to occur before the event initiation or after. The use of the term "inadvertently" for human-failure events is not specific enough to allow quantification. Most restrictive in tracing the results of the PRA was the fact that there is no list of basic events that includes the occurrence probability associated with each event.

With regard to the reliability model (fault tree) for the RCCS, the PRA qualitatively discussed the chances that this highly important passive heat removal system could be made inoperative by flow blockages or internal cavity events challenging its structural integrity. The PRA arrived at a final RCCS unavailability value of 10^{-6} per reactor-year by simply assuming that the most likely way the system could be totally destroyed was by a seismic event of some magnitude; namely, 1.6 g. The 1.6-g value was estimated by simply multiplying the known fragility value for a similar piece of equipment at the Zion plant by the ratio of the safe-shutdown-earthquake (SSE) values for both plants. Hazard data were then used to determine the probability of the 1.6-g earthquake; this resulted in the assignment of an unavailability of 10^{-6} per reactor-year for the RCCS. Given the significant effect the RCCS's unavailability of 10^{-6} per reactor-year has on plant risk and the uncertainty in the method for arriving at it (for example, ratioing equipment fragilities against plant SSE design values), additional effort is clearly needed before this value, or perhaps any other value of such low magnitude, can reasonably be accepted.

Another critical system for which a fault tree was not developed and to which a very low unavailability (4×10^{-5} per reactor-year) was assigned is the reserve shutdown control equipment (RSCE). This system would seem to have a common-cause failure potential from at least the plant protection and instrumentation system (PPIS) and quite possibly, common-mode and common-cause potentials within the two sets of RSCE. Accordingly, considerable uncertainty appears to exist at this time in the estimates of major system reliabilities.

15.3.2.3 Event Trees

MHTRG event trees were constructed for each of the seven initiating events. The event trees were system based and included failures of those systems and functions that provide protection once an event has started. The PRA's description of the event trees indicates that the methodology used was consistent with that for other PRAs. The requirements for operator action and plant monitoring during each event sequence were not, however, identified or addressed. Release categories were assigned only to those sequences with frequencies greater than 10^{-8} per year. Core-damage sequences were not developed, only non-core-damage releases, primarily because the very low unavailability assigned to the RCCS caused potential fuel-failure sequences to be truncated by the 10^{-8} per reactor-year cutoff criterion.

The systemic event-tree structures used in the PRA appear to have been adequate to generate core-damage cutsets. The functions (branches) seem to relate properly, logically, and chronologically. The release-category binning scheme is consistent with current LWR PRA techniques. As structures, the event trees appear to be credible and complete for the initiator groupings analyzed and, in this context, only moderately contribute to uncertainty in the risk results.

15.3.2.4 Release Categories

The MHTRG risk analysis begins with the identification of the accident initiators and proceeds to the performance of the resulting plant responses, the development of the accident sequences based on the event trees, the quantification of the sequence frequencies and, finally, the development of radioactive-release categories into which each of the plant sequences can be binned. As described in Section 8 of the PRA, the release categories consist of four bins (DC, DF, WC, WF), chosen on the basis that fission-product releases would occur from either forced-convection plant-cooldown events (F) under dry (D) and wet (W) conditions, or conduction-cooldown events (C) under dry (primary-coolant leaks) and wet (steam-generator-tube failures) conditions. The largest evaluated releases, DC-1 and DC-2, which considered failure of all cooling systems including the RCCS, had thyroid doses stated to be 47 and 23 rem, respectively, which therefore would make emergency evacuation mandatory for sequences in these categories. For DC-2, it was assumed that the RCCS was recovered after 100 hours, giving the lower dose. In both cases the fuel did not reach its failure temperature, but a release of 0.02 percent of halogens was assumed. The PRA event trees identified only three severe-accident sequences for these release categories; however, their frequencies were stated to be below the sequence cutoff of 5 x 10^{-7} per plant-year. Of particular importance here is the fact that these DC-1 and DC-2 sequence results depended heavily on phenomenological and frequency-related assumptions that inherently have large uncertainties.

15.3.3 Results

15.3.3.1 Definition of Licensing-Basis Events

Table 3.2-2 of the PSID identified 11 design-basis events (DBEs) that meet 10 CFR Part 100 dose limits and have individual frequencies greater than 10^{-4} per plant-year. Appendix C to the PRA contains the event trees for the seven initiating events described in Section 15.3.2.1 and the remaining licensing-basis events (LBEs) that will be evaluated against the PAG limits and safety-goal criteria. As shown below, based on DOE's 5×10^{-7} per year sequence-truncation limit, all the beyond-design-basis-event (BDBE) sequences were for only those events in which the primary coolant system was breached:

Initiator	BDBE sequences identified
Primary-coolant leak	30
Loss of heat transport system cooling	1
Earthquake	2
Loss of offsite power	None
Anticipated transient without scram	1
Control-rod withdrawal	None
Steam generator leak	24

The frequency of each of these 58 sequences was determined, and based on their release characteristics, each was binned into one of the four release categories (that is, DC, WC, DF, WF). The overall plant risk and offsite dose results were then calculated for the DBE and BDBE sequences and evaluated against the applicable DOE licensing criteria for the MHTGR.

In addition to the LBEs discussed above, Appendix G to the PRA contains brief descriptions of several sequences identified as beyond-licensing-basis events (BLBEs) that were proposed to demonstrate that the residual risk below the 5×10^{7} per plant-year sequence cutoff is insignificant. At this time, it appears that important accident-related considerations for at least one of these events, loss of RCCS, have not been adequately treated because the consequences of potential failures of the reactor vessel and reactor building cavity were not considered. A discussion of these inadequacies and their importance to judging the MHTGR risk is provided in subsequent sections.

15.3.3.2 Comparison With Safety Goals

The quantitative guidelines defined in NRC's Safety Goal Policy Statement are indicators of risk of early and latent fatalities and a frequency limit on large radioactive releases. With regard to the early-fatality criterion, the PRA indicated that this indicator was met, since none of the LBE sequences produced a single acute fatality (that is, doses were estimated to be less than the threshold value of 300 rem to the whole body). Regarding the latent-fatality indicator, the whole-body and thyroid rem-per-year doses were summed for all sequences, and using low-dose-response model conversions, the latent-fatality risk was found to be 6×10^{-9} per year, which is well below the safety goal guideline of 1.9×10^{-6} per year.

15.3.3.3 Comparison With Protective Action Guidelines

For the large-release criterion, assuming that large release is meant to be either a surrogate for no early fatality or compliance with 10 CFR Part 100, the MHTGR would clearly meet either criterion if it can be agreed that the plant meets PAG values of 1 and 5 rem whole-body and thyroid exposures, respectively. At this time, the PRA states that the MHTGR will meet these dose limits. In

Chapter 9 of the PRA, it is stated that "admittedly this first assessment reflects little margin in meeting these stringent limits, however it can be concluded that the MHTRG safety design approach makes compliance feasible." In contrast, however, the staff points out that uncertainties exist in many areas including the prediction of sequence frequencies, fuel temperatures, fuel-failure thresholds, and fission-product transport that make the conclusion that the MHTGR should be able to meet the PAGs questionable and dependent on further safety analyses and the successful completion of the research and testing programs described elsewhere in this report.

15.3.3.4 Evaluation of Defense-in-Depth

The PRA established the spectrum of DBE and BDBE accidents and provided analyses to demonstrate that criteria such as 10 CFR Part 100, PAGs, and safety goals would be met. It did not, however, provide an explicit comparison with current-generation LWRs to demonstrate that those factors conventionally thought to be important attributes of a defense-in-depth philosophy, such as conventional containment and the quality classification of equipment needed to reduce challenges to the safety systems, are not necessary.

15.3.4 Insights

15.3.4.1 Significance of Major Design Features

The design of the MHTGR's passive decay heat removal system and its silicon carbide-coated fuel particles are intended as significant inherent safety features that will greatly enhance the overall reliability of the plant. These design features are intended to provide (1) a highly reliable means to remove the nuclear decay heat, (2) a negative fuel-temperature coefficient to shut down the plant in the event of a failure to trip, (3) a high fuel-failure temperature to minimize fission-product releases, and (4) large core thermal inertia to accommodate emergency actions. These very positive advantages of the MHTGR design, nevertheless, need very careful evaluation in light of their uncertainties and sensitivities, especially with regard to their role in the design tradeoffs that have been integrated into this plant's unique design philosophy.

15.3.4.2 Adequacy of Defense-in-Depth

The degree of defense-in-depth provided by the MHTGR design is difficult to judge, both from an absolute sense because of the lack of a consensus-type body of MHTGR specific safety criteria and standards, and from a relative sense because of the inconclusiveness of a quantitative comparison of risk with that of LWRs. It does appear, however, that the conventional LWR defense-in-depth approach may be reduced by the over-reliance on the many previously noted inherent advantages in this plant. For example, major MHTGR front-line safety systems, such as emergency ac power (diesel generators) and shutdown heat removal systems (shutdown cooling system) are proposed not to be designed to LWR safety-related standards. This is likely to result in a situation whereby the plant's safety-related systems may frequently and seriously be challenged because of a philosophy that places ultimate reliance on a minimum of such systems, rather than on a broad base of similarly designed preventive and mitigative systems. The severity and likelihood of these challenges are never more apparent than if, for example, there should be a loss-of-offsite-power transient or other failure in which the first line of defense (that is, non-safety-related equipment) should fail to operate. In such a case, a significant rise in reactor pressure vessel temperature of sufficient level and duration would occur so that ASME service levels C and possibly D could be entered, as discussed in Section 5.2.5. Accordingly, from a PRA perspective, the overall level of the MHTGR's defense-in-depth compared with that of an LWR may be less in this area and its adequacy will have to await further study of the design and standards to which this plant will be built.

15.3.4.3 Comparison With Safety Goals and Protective Action Guidelines

For all possible sequences, down to the 5×10^{-7} per year cutoff, confirmation that the safety-goal guideline and the emergency protective action guideline limits have been met is very difficult. Specifically, the PRA provides analyses showing that for those sequences identified, the safety-goal guidelines are not exceeded. These analyses are not sufficient in and of themselves, however, to conclude whether these guidelines would not be exceeded for all possible internal or external events down to likelihoods as low as once in 2×10^6 million plant-years. It is important to note in this regard that the difficulty of identifying all potential severe-accident events down to such low likelihoods also exists for LWRs. In the LWR case, however, the severe-accident analyses address core melt and containment failures for sequences with frequencies as high as 10^{-5} to 10^{-6} and, therefore, such sequences are presumed to be legitimate surrogate analyses for comparison with the safety goals.

It appears that an affirmative decision regarding safety-goal comparison must be based to a large extent on bounding-type analyses which can demonstrate that for major plant disruptions (that is, failure of the reactor cavity and vessel), the occurrence frequency is satisfactorily low or if the disruption occurs, fuel failures will not result in significant fission-product releases to the environment. In this regard, the PRA should provide a much more detailed accident analysis than that presented in the PSID for a loss of the reactor cavity cooling system (RCCS) in which the effects of very high temperatures on the pressure vessel (~900°F) and the reactor-cavity concrete (>500°F) are included. For example, at these temperatures it is possible that both the reactor pressure vessel and the cavity walls could fail and cause considerable geometric changes in the heat-transfer models being used to calculate the heat flow from the fuel to the earth.

With regard to the MHTGR's ability to meet the PAGs, similar difficulties exist in making this judgment. Furthermore, as discussed previously, the calculated offsite dose values could change dramatically if more detailed and different models for fuel failures and assumptions for RCCS unavailability were made. In addition, the 5×10^{-7} per year sequence cutoff limit may be too high in view of the fact that it is only a factor of 2 below the very uncertain RCCS unavailability value of 10^{-6} per year.

15.3.4.4 Sequence Uncertainties

The staff's limited review of the PRA accident-sequence analyses primarily focused on the important implications that major uncertainties associated with the seismic and loss-of-offsite-power (LOSP) sequences could have in regard to the MHTGR risk. With regard to the seismically induced accident, the PRA's event tree contains fragility and hazard data that predict the RCCS "threshold-to-failure" cliff somewhere in the range of 0.8 to 2.0 g. Since this "threshold to failure" might occur at somewhat lower g-forces because of the uncertainties in seismic-fragility techniques and data, it is conceivable for a seismically induced sequence that the plant could have a 10^{-6} frequency of (1) losing all its decay heat removal design-basis systems and (2) fuel failures with significant environmental releases. With regard to the LOSP-induced accident, it is conceivable that the frequency of a resulting station-blackout challenge in which the RCCS is totally relied on to dissipate the plant's decay heat could be quite high because of the uncertainties in estimating the reliability of the plant's front-line non-safety-related heat removal and emergency ac power systems. In addition to the difficulties of quantifying and accepting the frequency of such a major plant challenge, there exist the additional uncertainties related to the plant's response (for example, the reactor pressure vessel's integrity). The specific concern with the pressure vessel for a station-blackout sequence is its increased chance of eventually failing because of potentially frequent temperature elevations (450°F to 900°F) and its subsequent effect on the continuing sequence of events. Accordingly, additional information with respect to possible uncertainties in modeling and data is needed to support the PRA's credibility.

15.3.5 Conclusions

15.3.5.1 Findings

(1) Additional information, including final design information pertaining to the integrity of the passive heat removal systems, needs to be provided to conclude that the MHTGR will meet safety-goal criteria or the specific PAG criteria for no sheltering or evacuation.

(2) The adequacy of the level of defense-in-depth for the MHTGR cannot be judged solely through a comparison of probabilistic results with risk-based criteria; however, the level of defense-in-depth appears to be less in some areas than that required for LWRs. For example, additional DOE commitments with respect to the integrity, reliability, and availability of industrial-grade systems that prevent challenges to safety-related systems are needed, such as those requested for the shutdown cooling system in Section 5.4.5.

(3) The identification of the MHTGR's licensing-basis-event sequences may be incomplete, if for no other reason than that the PRA implicitly states that there are no fuel-failure sequences down to a frequency as low as 10^{-8} per plant-year. Confirmation of this low probability of fuel failure must be sought in a future PRA based on more detailed design information.

15.3.5.2 Recommendations

(1) Consider performing detailed accident analyses involving reactor-cavity and vessel disruptions to the point where fuel failures could develop. The potential release of fission products could be tracked through the core and cavity environment to the atmosphere. The results would be used to understand and evaluate uncertainties before making judgments as to whether

fuel failures with fission-product releases resulting in significant off-site doses could occur down to levels of 10^{-8} per plant-year. Similarly, detailed severe-accident analyses could be performed for the other events described in Appendix G to the PRA.

(2) Consider improving reactor cavity cooling system (RCCS) structural margins in excess of the current safe-shutdown-earthquake design value of 0.3 g so that the uncertainties in the probability of seismically induced failures would be lessened.

(3) Within the plant's safety-related seismic and quality-control framework, develop commitments to improve the integrity, reliability, and availability of industrial-grade equipment over that currently identified so that the overall plant defense-in-depth would be enhanced.

(4) Develop firm data bases for (a) fuel-failure temperatures, (b) reactor-vessel performance during RCCS heat-removal operation, and (c) concrete-vault performance for loss of RCCS performance so as to judge the credibility of sequence analyses.

15.4 Independent Analyses

The Oak Ridge National Laboratory (ORNL) and Brookhaven National Laboratory (BNL), as contractors to the staff, have performed studies and independent analyses of postulated events pertaining to the safety performance of the MHTGR. Both ORNL and BNL performed studies of core conduction cooldown with and without the functioning of the reactor cavity cooling system (RCCS). Additionally, ORNL addressed reactivity-insertion events and BNL addressed large air-ingress events. Summary reports of ORNL and BNL are provided in Appendixes A and B, respectively, which, in turn, provide references supporting each summary. The purpose of this contracted work was to provide an independent assessment of those passive and inherent safety features most essential to the staff's judgments regarding safety performance of the MHTGR.

Although these independent studies are not yet complete in terms of additional details to be considered, of the events to be studied, and of documentation of the methods used, the studies have provided important and critical insights into the safety behavior of the MHTGR. This includes confirmation that, in concept, the MHTGR has the safety characteristics described by DOE. Important differences exist between some of the contractor estimates and those of DOE, but it is the staff's belief that further analysis, research information, and refined and agreed-upon assumptions and methodologies will lead to fully adequate agreement in all substantive areas of difference at a later design stage. The work performed thus far is summarized below.

15.4.1 Conduction Cooldown

Conduction-cooldown events are identified by ORNL as loss of forced cooling (LOFC). In Appendix A, maximum and average temperatures for the core and the maximum vessel temperatures are plotted against time, in hours, in Figures A.1, A.2, and A.3 for the reactor depressurized, pressurized (RCCS fully operational), and depressurized (RCCS fully failed), respectively. Temperatures generally begin to flatten out after about 60 hours and reach peaks around 100 hours at values generally in agreement with DOE values, except for maximum vessel temperatures, which are somewhat higher. These discrepancies are being investigated. In Appendix B, a best-estimate calculation of the depressurized-core-cooldown case (RCCS operational), the average and maximum core temperatures, the maximum vessel temperatures, and central-reflector average temperatures are plotted against time in Figure B.1. Maximum values occur at about 60 hours into the event at temperatures very close to DOE values. Also on Figure B.1 is a heat-flow plot showing that the decay-heat power almost equals the power to the RCCS panel at 60 hours. The calculations were sensitive principally to the parameters of (1) the decay-heat rate, (2) the effective conductivities of the graphite in the core and the outer reflector, (3) the emissitivities of the reactor vessel and the RCCS outer panel, and (4) the insulation value used for the upper plenum thermal protection structure. The sensitivity to the decay-heat rate and the effective thermal conductivity of the core is shown in Figure B.2.

A discussion of the results of these independent conduction-cooldown calculations for RCCS performance in terms of safety issues and research needs is given in Section 5.5. For the case of sustained loss of the RCCS, Appendix B gives a description of the factors to be considered in the calculations for this event and estimates that fuel temperatures peak at about 80 hours, reactor-vessel temperatures reach 700°C and 800°C between 400 and 1200 hours, and several regions of the concrete in the reactor cavity can reach 700°C.

15.4.2 Short-Term Response to Flow and Reactivity Transients

The transients examined included loss of forced cooling without scram, moisture ingress, spurious control-rod-group withdrawal, control-rod ejection, and rapid core cooling without scram. The results for all transients were benign and generally consistent with DOE predictions. For the case of moisture ingress from the failure of a single steam generator tube, plotted in Figure A.4, the ORNL calculation gave lower temperature increases than DOE values because the collection of water in graphite pores was not assumed (see Section 4.4.5.D for discussion of this effect).

15.4.3 Conduction Cooldown Without Reactor Trip

Appendix B discusses the consequences of the depressurized-conduction-cooldown event when reactor trip is not achieved. The reactor is shut down initially by the negative temperature coefficient, but after about 40 hours, it becomes critical again because of the decay of xenon-135, and it continues to oscillate around a higher temperature and heat input than is the case where the reactor is maintained subcritical. A best-estimate calculation showed that peak core temperatures reached 1600°C at about 60 hours, reached a maximum of 1760°C at about 120 hours, and prevailed for hundreds of hours rather than decaying moderately. The vessel reached a temperature of 550°C. This event will be studied further, including the case when the RCCS has failed. Also, as stated in Section 4.4.5.C, the staff is requiring that reliable mechanical means be provided to ensure sustained reactor shutdown.

15.4.4 Large Air Ingress

Appendix B presents analyses of the consequences of the ingress of air to the reactor as a result of a double-guillotine break in the crossduct. The conclusion is the same as that DOE provided in Appendix G.2.3 of the PSID, which stated that a relatively small amount of graphite would be oxidized based on air supply and convection-flow geometry. The staff has estimated that no fuel damage would occur except for bounding event 5.

15.5 Siting-Source-Term Selection and Use*

DOE has proposed a mechanistic siting source term (SST) for site evaluations, in accordance with 10 CFR Part 100, on the basis that no substantial fuel failure will occur even when the reactor is subjected to the bounding events (BEs). The proposed SST, as described in Section 11.1, is that radionuclide inventory in the primary system derived from a small amount of initially defective fuel that can be augmented to only a small degree by the occurrence of certain BEs. The staff has accepted this source term for use in the MHTGR conceptual design review and has determined that it is in accordance with the criteria set forth in Section 3.2.2.2, "Siting-Source-Term Calculation and Use." Final selection of the SST for the MHTGR will depend mainly on the results of research programs described in Sections 4.2.4 and 11.1.4, prototype-plant testing as described in Chapter 14, and continued safety analyses to be performed by DOE, the staff, and the staff's contractors. The results of this effort are expected to confirm DOE's present SST or cause the development on a mechanistic basis of a suitable alternative. In either case it is expected to be a quantity that can be used as a replacement for the TID-14844 value used for LWRs (AEC, 1962). The SST, when finally developed, will contribute to final decisions affecting the requirements for reactor containment; the determination of the exclusion area, the low-population zone, and the population center distance; emergency-planning requirements; comparison with the safety-goal guidelines; and the treatment of multiple reactor units at a single site with regard to their interactions and degree of coupling.

DOE has developed support for the proposed SST from analyses of the licensing-basis events presented in the PSID, the PRA, the Emergency Planning Basis Report, and its "beyond-licensing-basis events" presented in Appendix G of the PRA. Further information has been and is being developed by the staff's consultants, as described in Section 15.4. Development of the SST is being approached mechanistically in all cases, an approach that DOE has confirmed is a fundamental objective in the MHTGR safety analysis and design. In its review of the mechanistic approach, the staff has concluded that, for plant designs with long response times and the capability to withstand many low-probability events, it is acceptable and preferred to develop mechanistic bases rather than to follow the customary approach of postulating a nonmechanistic source term, which could obscure important phenomenological considerations. Furthermore, the mechanistic approach can be viewed as a safety enhancement in that the limits of the MHTGR's hazards would be technically defined rather than encompassed within a envelope that has not traditionally required complete technical accounting and understanding of all bounding events deemed credible. The relationship between the selection of bounding events that must be considered in the design of the MHTGR and the use of the proposed mechanistic source term for siting evaluations is seen, however, to be of critical importance. The staff's approach to this concern is addressed in Section 15.2.4, "Residual Risks."

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*Indicates statement is particularly sensitive to change by evaluation of forthcoming DOE information.

The SST is being developed by calculations of releases in terms that consider a time history that defines the scope and extent of events that need to be considered and/or certain end-of-sequence reactor states at times when it is clearly evident that additional fuel failure and fission-product releases are not credible. It should be noted that development of the mechanistic SST requires that all credible event initiators and sequences be identified or bounded by phenomenological understandings. The staff prefers this approach because it helps to demonstrate the status of knowledge concerning radioactive-release risks of the MHTGR.

In summary, DOE supports its proposed SST from postulated events and safety-analysis models that consider mainly the release of radioactive material circulating with the helium coolant and radioactive material plated out on primary-system surfaces that is available for liftoff when the primary system is depressurized. Although fuel-failure mechanisms are also considered in these models, they are only those derived from that fraction of fuel particles manufactured with defective silicon carbide coatings or heavy-metal contamination outside the silicon carbide coating. The events and models take into account the initial fuel quality, normal operations, the effects of temperature elevation, depressurization rates, and chemical reactions (including hydrolysis of exposed fuel kernels, liberation of sorbed fission products by graphite oxidation, and steam-augmented release of plated-out fission products). The events and models considered do not, however, extend to the regime of gross coating failures and fission-product releases that could be caused by a combination of decomposition of the silicon carbide coating layer that could occur at higher temperatures, internal pressures of fission gases, and chemical attacks on the interior of the silicon carbide coating by certain fission-product species. All the models considered assume that the passive heat removal system is effective in keeping the maximum fuel temperature below a level that would initiate thresholds for such failure mechanisms. The radionuclide releases from all the postulated events were calculated to be low and, as reported in Tables 15.1, 15.2, and 15.3, are within the PAGs and, of course, 10 CFR Part 100 guidelines. This is also evident from Figure 15.1. Final staff acceptance of the DOE-proposed SST depends on the confirmation of the release characterics of the fuel and the ability of the passive heat removal system to restrict fuel-particle temperatures to below failure thresholds.

15.6 Conclusions*

The conclusions regarding the safety analyses, including the PRA review, are based on preliminary methodologies applied to a design in the conceptual stage of development. The conclusions take into consideration the safety issues and research needs identified elsewhere in this report, that suitable design criteria have not yet been established in several important areas and that additional and more detailed safety analyses need to be performed, including improved PRAs. The most important conclusions are based on the judgments of the response of the design features that (1) permit passive removal of decay heat, (2) provide inherent reactivity control, and (3) allow the fuel to retain its integrity at high temperatures and under conditions of chemical attack by

*Indicates statement is particularly sensitive to change by evaluation of forthcoming DOE information.

steam and air. The safety analyses were performed using both deterministic and PRA methodologies. Although the PRA provided important input into accident selection and defense-in-depth considerations, engineering judgment was necessary to give confidence that a sufficiently representative set of postulated transients and events had been selected for analysis. The findings and recommendations of the staff's PRA review, some of which are also apparent from the deterministic review, are given in Section 15.3.5.

The staff concludes that:

(1) The MHTGR potentially can provide a level of safety that is very high and can be judged at this time to offer qualitatively an overall enhancement of safety. In this sense, it meets the safety-enhancement objective encouraged in the NRC Advanced Reactor Policy Statement. In the course of this review, however, major safety issues have been identified that relate to changes needed in design selections and other major safety issues that will require successful analytical studies, research, and/or testing for resolution.

(2) The most important accident-mitigation features of the MHTGR, like other HTGRs, are (a) its slow response to core-heatup events because of the core's large heat capacity and low power density and (b) the very high temperature that the fuel can sustain before significant fission-product release occurs. Also, like other HTGRs, its major potential vulnerabilities derive from the need to protect metal components from exposure to hot helium at elevated temperatures during postulated transients and to protect hot graphite and fuel from uncontrolled access to air and moisture.

(3) The staff has judged that the siting source term can be based on a mechanistic analysis of fuel failure and radionculide inventory contained in the circulating helium or plated out within the primary system. Final acceptance of a mechanistically calculated source term is dependent on satisfactory accomplishment of research and development goals, satisfactory resolution of the safety issues and deferred items, and a prototype test program demonstrating that the combination of research and development findings and analytical predictions confirm the staff's detailed and overall safety conclusions for the MHTGR.

(4) Release estimates provided by DOE cannot be quantitatively confirmed at this time to meet the protective action guidelines (PAGs) at the site boundary because of lack of experimental data and validated methodologies. The staff believes, however, that the releases would actually be very low and that the MHTGR has the potential to meet the PAGs at the site boundary. At a later design stage, demonstration that the PAGs can be met will be necessary by taking into account an improved data base and a better understanding of fission-product transport and retention by the primary coolant system and reactor building.

(5) For certain severe low-probability events, time will be of particular importance and the staff believes that the releases will not exceed a small fraction of 10 CFR Part 100 guidelines and possibly the PAGs during the first 2 or 3 days after event initiation. The potential may exist for some event sequences, particularly those involving total reactor cavity cooling system failure, that the 10 CFR Part 100 guidelines could be exceeded at times by more than a few days unless precluded by human recovery actions.

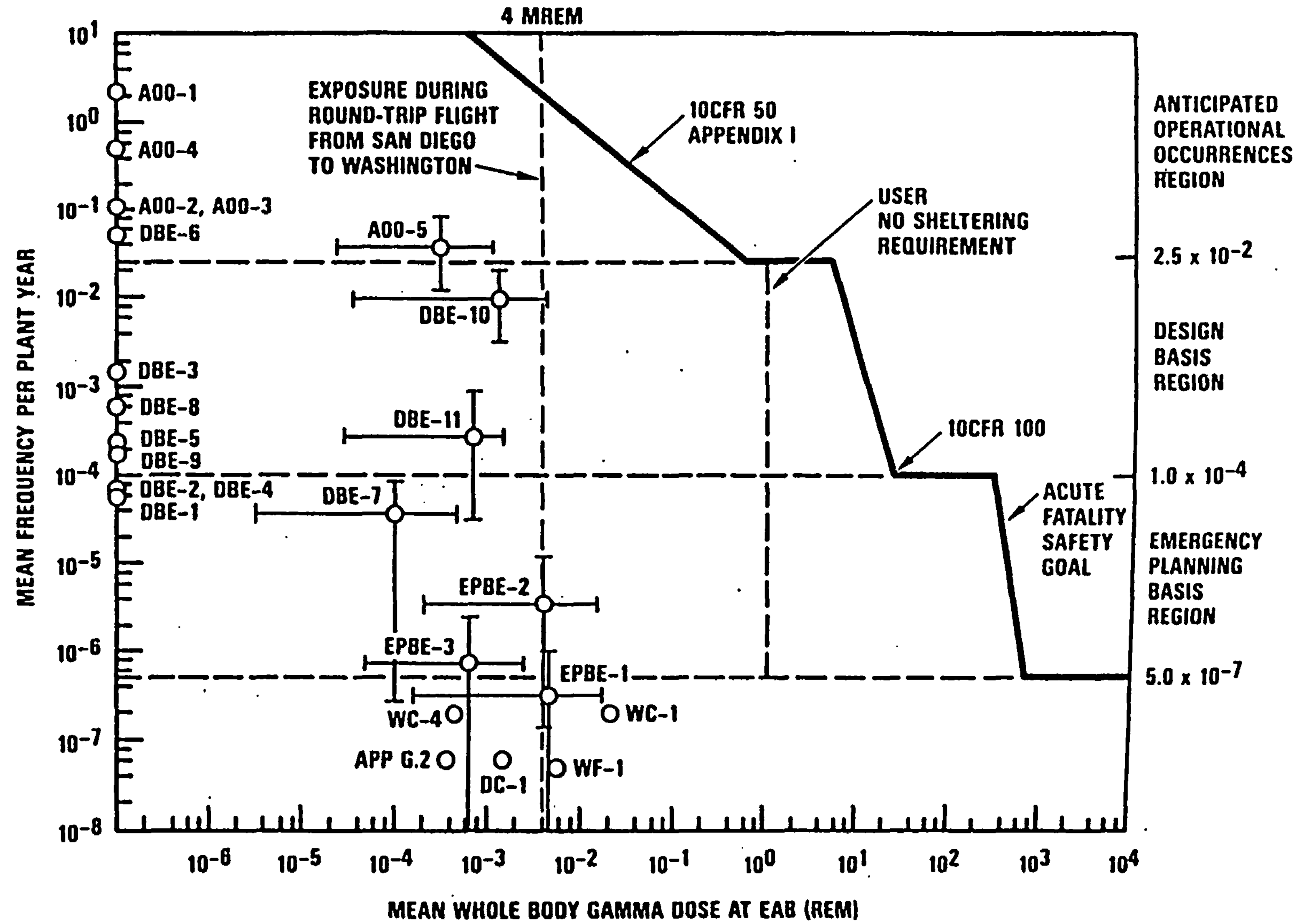

Figure 15.1 Assignment of top-level regulatory criteria and results of safety analysis
Source: DOE, 1986-3

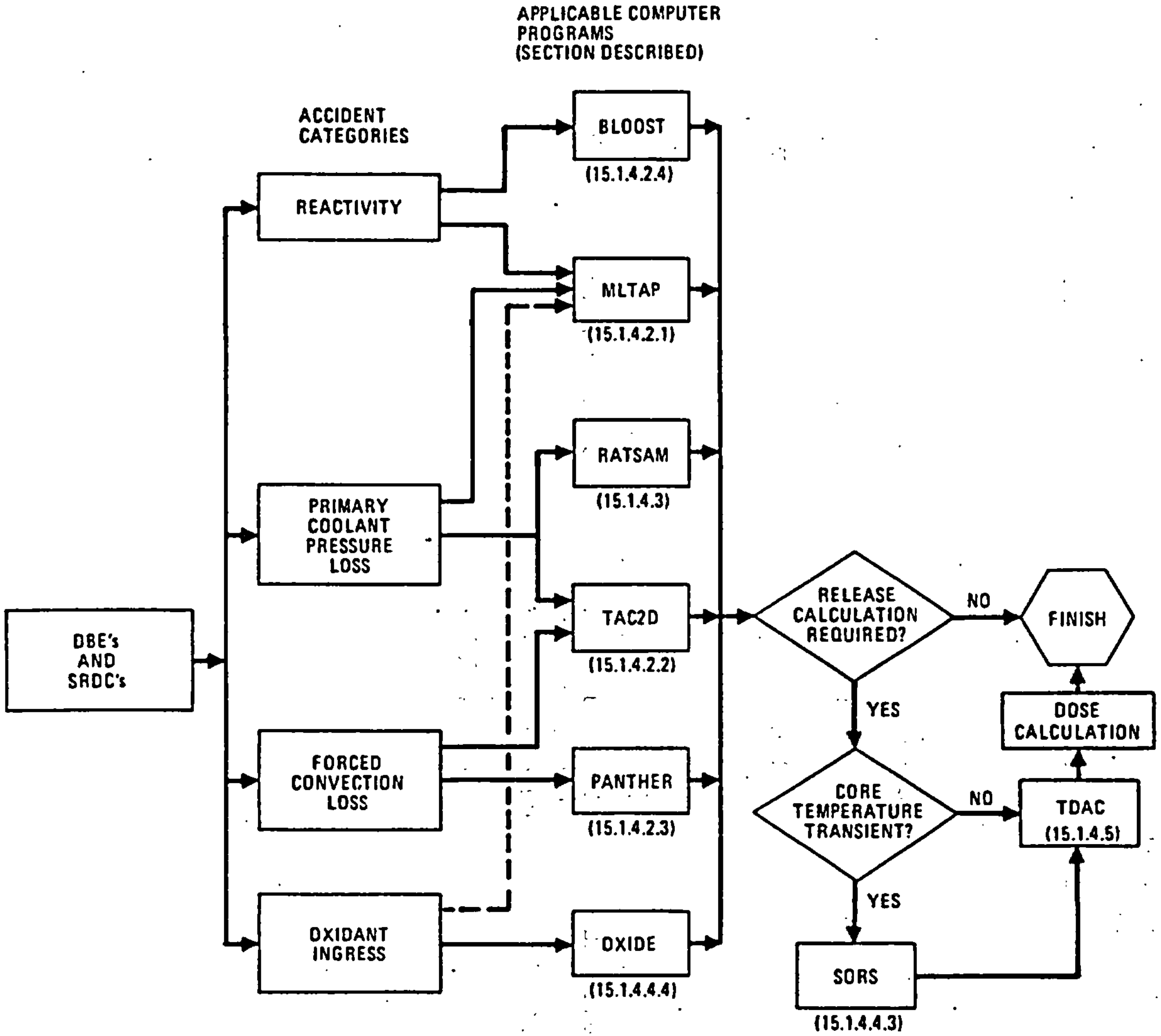

Figure 15.2 Computer codes used in MHTGR safety analysis
Source: DOE, 1986-3

Table 15.1 Licensing-basis events analyzed by DOE - design-basis events (DBEs) and safety-related design conditions (SRDCs)

Identifying number and type	Initiation	Sequence		Consequences and comments
		DBE	SRDC	
(1) Pressurized conduction cooldown	Loss of all ac power causes loss of forced cooling	Reactor trips on main-loop trip; core cooled by reactor cavity cooling system (RCCS)	Reactor trips on high power-to-flow ratio; core cooled by RCCS	Transient response essentially the same for both events; core temperature reaches a maximum of 1286°C after 100 hours; no fission-product (FP) release
(2) Pressurized conduction cooldown without control-rod trip (anticipated transient without scram)	Loss of heat transport system (HTS) followed by failure to trip reactor	Reactor trips on high power-to-flow ratio via reserve shutdown control equipment (RSCE); decay heat removed by successful shutdown cooling system (SCS)	Same reactor trip as DBE; core cooled by RCCS	Peak core temperature, DBE-lower than that for SRDC, SRDC-1296°C; no FP release
(3) Pressurized conduction cooldown with control-rod withdrawal	Withdrawal of maximum-worth control-rod group followed by HTS failure	Reactor trips on high power-to-flow ratio; core cooled by SCS	Same reactor trip as DBE; core cooled by RCCS	Peak core temperature, DBE-lower than that for SRDC, SRDC-1307°C; no FP release
(4) Pressurized conduction cooldown with control-rod withdrawal	Withdrawal of maximum-worth control-rod group followed by HTS and SCS failure	Reactor trips on high power-to-flow ratio; core cooled by RCCS	Same as above and for DBE-4	DBE and SRDC are the same; peak core temperature is 1307°C; no FP release

Table 15.1 (Continued)

Identifying number and type	Initiation	Sequence: DBE	Sequence: SRDC	Consequences and comments
(5) Pressurized conduction cooldown following safe-shutdown earthquake	0.3-g earthquake trips HTS and SCS; RCCS survives	Reactor trips on main-loop trip; core cooled by SCS	Reactor trips on high power-to-flow-ratio; core cooled by RCCS	DBE similar to DBE-2; SRDC same as DBE-1; no FP release
(6) Depressurized conduction cooldown with moderate moisture ingress	Moderate (5.7 kg/sec) steam generator tube leak	Moisture monitors detect leaks and trip reactor on outer control rods; HTS trips; steam inventory dumped to tanks, and steam generator isolated; core cooled by SCS	Reactor trip by outer rods on high power-to-flow ratio, steam generator isolated but steam not dumped; helium pressure rises; RSCE trips; relief valve opens; decay heat removed by RCCS	No offsite dose for DBE; relief-valve lifting causes exclusion area boundary offsite dose for SRDC, at 30 days, thyroid, 3.8 rem, whole body, 0.045 rem, highest doses for all SRDCs analyzed
(7) Depressurized conduction cooldown with moderate moisture ingress	Same as 6	Same as 6, except decay heat removed by RCCS	Same as 6	DBE has potential for small offsite dose if relief valve opens and closes; SRDC doses same as 6
(8) Depressurized conduction cooldown with small moisture ingress	Small (0.05 kg/sec) steam generator tube leak not detected by moisture monitors	Reactor and HTS trip on high pressure; decay heat removed by SCS	Reactor and HTS trip on high pressure; decay heat removed by RCCS	No offsite dose for DBE; SRDC doses bounded by SRDC-6

Table 15.1 (Continued)

Identifying number and type	Initiation	Sequence: DBE	Sequence: SRDC	Consequences and comments
(9) Depressurized conduction cooldown with small moisture ingress	Same as 8, but steam dump system valves fail to close	Same trip as 6; dump tank reaches primary-system pressure; decay heat removed by SCS; primary system remains pressurized	Trips and heat removal same as 6; no credit given for dump-tank presence and reactor slowly depressurized through steam generator to reactor cavity	No offsite release for DBE; SRDC doses bounded by SRDC-6
(10) Depressurized conduction cooldown with moderate primary-coolant leak	Moderate (81.9 sq cm) helium leak at top of steam generator vessel	Reactor and HTS trip on low pressure; decay heat removed by SCS; helium leaks from reactor building	Reactor and HTS trip on low pressure; decay heat removed by RCCS	Very small doses for both DBE and SRDC; doses bounded by SRDC-6
(11) Depressurized conduction cooldown with small primary-coolant leak	Small (0.32 sq cm) helium leak at top of reactor vessel; leak area sized to optimize conditions for maximum release	Reactor trips on low pressure; HTS fails to start; decay heat then removed by RCCS; building dampers open to relieve pressure	Reactor and HTS trip on low pressure; decay heat removed by RCCS; thermal transient same as 10	Low dose for DBE; for SRDC at 30 days, thyroid, 3.1 rem, whole body, 0.01 rem

Table 15.2 Emergency-planning-basis events (EPBEs) proposed by DOE

Identifying number and type	Initiation	Sequence	Consequences (50% confidence level)*
(1) Moisture inleakage with delayed steam generator isolation and without forced cooling	Moderate-sized steam generator leak (5.7 kg/sec)	Reactor trip and steam generator isolation are delayed until nearly 3000 kg of steam enters primary system; relief valve lifts and fails open; reactor depressurizes; decay heat removed by reactor cavity cooling system.	Cumulative releases to environment: Kr-88, 9.6 Ci; Sr-90, 3 x 10^{-2} Ci; I-131, 4.6 Ci; Cs-137, 3.3 Ci. 30-day mean exclusion area boundary (EAB) thyroid dose, 1 rem, whole-body mean gamma dose, 7 mrem.
(2) Moisture inleakage with delayed steam generator isolation	Same as EPBE-1	Same as EPBE-1, except cooling is provided by shutdown cooling system; releases are somewhat different than for EPBE-1 because of lower core temperature and forced-circulation-enhanced fission-product liftoff.	Cumulative releases to environment: Kr-88, 2.7 Ci; Sr-90, 8.3 x 10^{-2} Ci; I-131, 3.4 Ci; Cs-137, 1.7 x 10^{-1} Ci. 30-day mean EAB thyroid dose, 0.8 rem, whole-body mean gamma dose, 7 mrem.
(3) Primary-coolant leakage in four modules with neither forced cooling nor pumpdown	Earthquake causes small primary-coolant leaks in all four modules	Slow depressurization over period of 25 hours; hours; releases to environment by reactor building leakage, attenuated by plate-out, settling, and decay within building.	Both cumulative releases and offsite doses lower than for EPBE-1 and -2.

*At the 95-percent confidence level, all three EPBEs approach but do not exceed at the EAB the 5-rem protective action guideline thyroid inhalation dose commitment for sheltering.

Table 15.3 DOE-analyzed bounding events proposed by the NRC staff*

Bounding event	DOE-assessed frequency (per plant-year)	Thyroid dose at EAB (rem)		Whole-body dose at EAB (rem)	
		At 36 hours	At 30 days	At 36 hours	At 30 days
BE-1					
BE-1 A	2×10^{-12}	2.3	2.3	0.011	0.011
BE-1 B	2×10^{-11}	None	None	None	None
BE-1 C	4×10^{-13}	0.1	0.13	4×10^{-4}	4×10^{-4}
BE-2					
BE-2 A	5×10^{-5}	None	None	None	None
BE-2 B	1×10^{-7}	0.04	0.04	2×10^{-4}	2×10^{-4}
BE-3					
BE-3 A	2×10^{-7}	None	None	None	None
BE-3 B	4×10^{-10}	0.07	0.07	3×10^{-4}	3×10^{-4}
BE-4					
BE-4 A	$<2 \times 10^{-11}$	2.2	2.2	0.011	0.011
BE-4 B	$<8 \times 10^{-11}$	2.2	2.2	0.011	0.011
BE-5	$<1 \times 10^{-11}$	2.2	2.6	7×10^{-3}	9×10^{-3}
BE-6					
BE-6.1 (SSE)	1×10^{-4}	None	None	None	None
BE-6.2 (beyond SSE)	7×10^{-7}	0.19	0.22	1×10^{-3}	1×10^{-3}

*Listed in Table 3.7.
Source: DOE, 1986-3.

Table 15.4 Integrity concerns for key safety systems

System	Concerns	Review status	Reference SER section
Fuel particles	Fission-product-retention capability at temperatures of 1600°C and higher; effects of chemical attack at these temperatures.	Completion of research necessary to resolve concern.	4.2.4
Core and core support system	Structural failures and/or displacements due to very large seismic loading, thermal stresses, or combinations thereof; water or air ingress over time could diminish strength and eventually cause structural failure of graphites.	Information supplied by DOE not reviewed by experts because of funding limitations; review expected at time of construction-permit application.	3.5, 4.5
Reactor vessel	Decrease in tensile strength at elevated temperatures could lead to undefined consequences; frequency of vessel depressurization at elevated temperatures needs to be reduced.	Issues being addressed by ASME Code inquiry; staff decisions will consider deliberations and findings of inquiry.	5.2.5
Reactor cavity cooling system (RCCS)	Potential for failure or diminished capacity by means of cavity overpressure, a very large seismic event, or moisture blanketing by steamline break.	Defense-in-depth concerns may cause postulation of failure, even though remote.	5.5.5
Reactor cavity	Loss of concrete integrity by local overheating caused by RCCS failure.	Additional information needed to resolve concern.	6.2

16 TECHNICAL SPECIFICATIONS AND ADMINISTRATIVE CONTROLS

Information on technical specifications and other administrative controls was not presented in the Preliminary Safety Information Document. Such information should be fully available at the initiation of the preliminary standard safety analysis report review because in some cases technical specifications and other administrative controls will necessarily vary from the type of material submitted for light-water reactors and involve unique safety and research issues. For example, administrative controls will be necessary to address the integrity, reliability, and availability of equipment that is not classified as safety grade but is intended to provide defense-in-depth with regard to reducing challenges to safety-related systems, structures, and components (SSC). The need for such administrative controls was explicitly identified in Section 5.4.5.B with regard to the performance of the shutdown cooling system. Of course, information on administrative controls must be established for all safety-related SSC, as well as other SSC of industrial grade that provide defense-in-depth functions. Especially significant will be the administrative controls pertaining to the mechanistic source term such as those that control the plateout and the circulating radionuclide inventories. Such controls would be expected to consider and be in conformance with the research program findings pertaining to the "back-calculation" model discussed in Section 11.1.

17 QUALITY ASSURANCE COMMITMENT AND ACCEPTABILITY

Chapter 17 of the Preliminary Safety Information Document (PSID) defines the quality program requirements to be applied through the conceptual design phase of the MHTGR and describes generally how the program will be expanded during the preliminary and final stage (preliminary and final standard safety analysis reports [PSSAR/FSSAR]). The staff evaluation of this quality assurance (QA) program description is based on a review of this information, supplemented by discussions with the MHTGR plant-design control office (PDCO), and responses to Comments 17-1 through 17-8, which were included in Amendment 6 to the PSID, to determine the extent to which the QA program would comply with the requirements of 10 CFR Part 50, Appendix B, and appropriate Standard Review Plan (SRP) Chapter 17, Revision 2, criteria if implemented at the PSSAR/FSSAR phase, as described.

The applicable portions of Section 17.1 of the SRP are used for review and evaluation of the description of the QA program for design and construction of light-water reactors in each application for a construction permit, a manufacturing license, or a standardized-design approval. Section 17.2 is used for review of the QA program at the operating-license stage. While review of the PSID does not fall into any one of these categories, the SRP was used as a review aid by the staff, since no other guidance exists.

Under SRP Chapter 17, the QA program description is expected to describe in the applicable SAR how each criterion of 10 CFR Part 50, Appendix B, will be met. Acceptability of the program is judged, after review, to determine whether each of the criteria of Appendix B is acceptably addressed, the commitment to comply with NRC regulations and regulatory guides and the approach to meeting the QA criteria are adequate, and the organizational responsibilities and authorities are such that those performing QA activities have sufficient independence to carry out those activities without undue influence from those directly responsible for costs and schedules. The PSID identifies the overall QA program requirements applicable to the MHTGR program; responsibilities of the PDCO, prime contractors, and subcontractors; and DOE management responsibilities for ensuring quality; and describes the general approach to be used for development of a more detailed QA program through the preliminary and final phases. The QA program description does not at this time contain the level of detail needed for an SRP-type review, but the staff understands that DOE will provide such detail when a construction application is tendered.

The PSID states that the QA program fully complies with 10 CFR Part 50, Appendix B; is consistent with Regulatory Guide 1.28, Revision 3; and is applied to safety-related development activities. However, because the staff's review did not find DOE's safety classifications of structures, systems, and components fully acceptable (see Section 3.3), DOE will need to provide at the time of a construction-permit review a list of the specific structures, systems, and components to which the QA program applies, which includes modifications, in accord with the guidance provided in this document. Furthermore, the PSID does not illustrate compliance with Appendix B or the method to be used for complying with Appendix B at the preliminary and final stages. The PSID does state, however, that the full QA program will be applied by all program participants and subcontractors for the preliminary and final phases. For staff acceptance all Appendix B criteria must be applied, unless exceptions can be justified for individual cases. DOE identified seven regulatory guides it plans to consider, namely, 1.30, 1.33, 1.37, 1.38, 1.54, 1.94, and 1.116. This is acceptable provided the QA program description submitted with an application describes methods acceptable to the staff for meeting 10 CFR Part 50, Appendix B, including (along with consideration of the seven guides listed above) an explanation of the applicability of 10 CFR 50.55(a), 50.34(a)(7), 50.34(b)(6)(ii), 50.48, 50.49, and Appendix A, and Regulatory Guides 1.8, 1.26, 1.28, 1.29, and 1.39.

Table 17.1-1 of the PSID identifies the applicability of the 10 CFR Part 50, Appendix B, criteria to MHTGR program participants and subcontractor organizations for the conceptual design phase. The table appears to indicate the necessary elements for the scope of the conceptual design phase activities, as described by the PDCO, except for Criterion XV, "Nonconforming Material, Parts, or Components." Criterion XV should also be applied to the design portions of contractors' work. Although Criterion XVI, "Corrective Action," is applicable to these activities, as indicated in Table 17.1-1, it is essential to the quality of the overall design to ensure that nonconformances are properly identified and dispositioned so that appropriate corrective action may be implemented. Based on the PSID statement that the full QA program will be applied by all program participants and subcontractors for the preliminary and final phases, the approach is acceptable. At the time an application is tendered by DOE, the staff will expect the QA program description to include or reference the details of how the program participants and principal contractors will meet each criterion of 10 CFR Part 50, Appendix B.

The PSID makes a commitment to use Regulatory Guide 1.28, Revision 3, which by endorsing ANSI/ASME NQA-1 requires that persons or organizations responsible for ensuring that an appropriate QA program is established and for verifying that activities affecting quality have been correctly performed have sufficient authority, access to work areas, and organizational freedom to perform their functions. The NQA-1 criterion also requires that such individuals report to a management level where the required authority and organizational freedom are provided, including independence from cost and schedule considerations. Table 17.1-1 of the PSID applies 10 CFR Part 50, Appendix B, Criterion I, "Organization," to all program participants and subcontractors. The PSID indicates that this practice will continue through the preliminary and final phases. These general commitments are acceptable. At the preliminary and final phases, the application will need to detail the organizational structure and interfaces, as well as the performance of personnel and their capability to carry out their QA responsibilities.

It should also be noted that QA criteria will be applied to research, development, and testing programs, including work to be performed in foreign countries. Details of how U.S. QA requirements will be achieved in foreign countries will need to be developed at a later review stage.

Provided a future application for the MHTGR contains the QA program details described above, which the staff finds meet the applicable SRP criteria, the staff concludes that the QA program would support MHTGR design licensing.

18 REFERENCES

ACI-349 — American Concrete Institute, "Code Requirements for Nuclear Safety Related Structures," Detroit, Michigan, 1976.

AEC, 1962 — U.S. Atomic Energy Commission, "Calculation of Distance Factors for Power and Test Reactor Sites," TID-14844, March 23, 1962.

AISC-S326 — American Institute of Steel Construction, "Specification for the Design, Fabrication and Erection of Structural Steel for Buildings," Chicago, Illinois, November 1, 1978.

ANSI/ANS-8.1 — American National Standards Institute/American Nuclear Society, "Nuclear Criticality Safety in Operations With Fissionable Materials Outside Reactors," La Grange Park, Illinois, October 1983.

ANSI/ANS-13.1 — American National Standards Institute/American Nuclear Society, "Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities," La Grange Park, Illinois, 1982.

ANSI/ANS-19.1 — American National Standards Institute/American Nuclear Society, "Nuclear Data Sets for Reactor Design Calculations," La Grange Park, Illinois, July 1983.

ANSI/ANS-19.3 — American National Standards Institute/American Nuclear Society, "Determination of Neutron Reaction Rate Distributions and Reactivity of Nuclear Reactors," La Grange Park, Illinois, June 1983.

ANSI/ANS-19.3.4 — American National Standards Institute/American Nuclear Society, "Determination of Thermal Energy Deposition Rates in Nuclear Reactors," La Grange Park, Illinois, June 1983.

ANSI/ANS-19.4 — American National Standards Institute/American Nuclear Society, "A Guide for Acquisition and Documentation of Reference Power Reactor Physics Measurements for Nuclear Analysis Verification," La Grange Park, Illinois, June 1983.

ANSI/ANS-19.5 — American National Standards Institute/American Nuclear Society, "Requirements for Reference Reactor Physics Measurements," La Grange Park, Illinois, May 1984.

ANSI/ASME NQA-1	American National Standards Institute/American Society of Mechanical Engineers, "Quality Assurance Program Requirements for Nuclear Facilities," New York, N.Y., July 1986.
ASME Code Case N-47	American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section III, Division 1, "Class 1 Components in Elevated Temperature Service," New York, N.Y., February 1987.
ASME Code Case N-201-1	American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section III, Division 1, "Class CS Components in Elevated Temperature Service," New York, N.Y., July 1982.
ASME Code Section III, Division 1	American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, "Nuclear Power Plant Components," New York, N.Y., 1986 Edition.
ASME Code Section III, Division 2, Subsection CE (proposed)	American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, "Design Requirements for Graphite Core Supports," New York, N.Y., April 1984.
ASME Code Section XI, Division 1	American Society of Mechanical Engineers, Boiler and and Pressure Vessel Code, "Rules for Inservice Inspection of Nuclear Power Plant Components," New York, N.Y., 1986 Edition.
ASME Code Section XI, Division 2	American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, "Rules for Inspection and Testing of Components of Gas-Cooled Plants," New York, N.Y., September 1986.
Cleveland, 1988	John C. Cleveland (ORNL), "Modular High-Temperature Gas-Cooled Reactor Short Term Thermal Response," American Nuclear Society, Seattle Topical Meeting, May 1-5, 1988.
Dey, 1988	Moni Dey (NRC), memorandum to T. King and others (NRC), "Risk Benefits of Containment System for the MHTGR," May 3, 1988.
DOE, 1986-1*	U.S. Department of Energy, "Licensing Plan for the Standard MHTGR," HTGR-85-001, Rev. 3, February 1986.
DOE, 1986-2*	U.S. Department of Energy, "Top-Level Regulatory Criteria for the Standard HTGR," HTGR-85-002, Rev. 2, October 1986.

*Indicates that the document is classified as "applied technology" and is not in the public domain. Requests for these documents should be made through the U.S. Department of Energy, Washington, D.C.

DOE, 1986-3* U.S. Department of Energy, "Preliminary Safety Information Document for the Standard MHTGR," HTGR-86-024, Vols. 1-5, October 1986 plus 10 amendments through February 1989.

DOE, 1987-1* U.S. Department of Energy, "Probabilistic Risk Assessment for the Standard Modular High Temperature Gas-Cooled Reactor," DOE-HTGR-86-011, Vols. 1 and 2, Rev. 3, January 1987 through Rev. 5, April 1988. Vol. 2, proprietary.

DOE, 1987-2* U.S. Department of Energy, "Emergency Planning Basis for the Standard MHTGR," DOE-HTGR-87-001, February 1987.

DOE, 1987-3* U.S. Department of Energy, "Regulatory Technology Development Plan for the Standard MHTGR," DOE-HTGR-86-064, Rev. 1, August 1987.

DOE, 1987-4* U.S. Department of Energy, "MHTGR Assessment of NRC LWR Generic Safety Issues," DOE-HTGR-87-089, September 1987.

EPA, 1980 U.S. Environmental Protection Agency, "Manual of Protection Action Guides and Protective Actions for Nuclear Incidents," EPA-520/1-75-001, Washington, D.C., June 1980.

50 FR 32138 U.S. Nuclear Regulatory Commission, "Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants," August 8, 1985.

51 FR 24643 U.S. Nuclear Regulatory Commission, "Regulation of Advanced Nuclear Power Plants; Statement of Policy," July 8, 1986.

51 FR 28044 U.S. Nuclear Regulatory Commission, "Safety Goals for the Operation of Nuclear Power Plants," August 4, 1986.

52 FR 6339 U.S. Nuclear Regulatory Commission, "Emergency Core Cooling Systems; Revisions to Acceptance Criteria," March 3, 1987.

52 FR 34884 U.S. Nuclear Regulatory Commission, "Nuclear Power Plant Standardization, Policy Statement," September 15, 1987.

53 FR 32060 U.S. Nuclear Regulatory Commission, "Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Reactors," August 23, 1988.

*Indicates that the document is classified as "applied technology" and is not in the public domain. Requests for these documents should be made through the U.S. Department of Energy, Washington, D.C.

GA, 1985* GA Technologies, Inc., "US/FRG Accident Condition Fuel Performance Models," HTGR-85-107, San Diego, California, December 1985.

GA, 1986-1* GA Technologies, Inc., "Licensing Basis Event Selection Criteria," HTGR-86-001, Rev. 1, San Diego, California, February 1986.

GA, 1986-2* GA Technologies, Inc., "Application of Bridging Methods for Standard HTGR Licensing Bases," HTGR-86-017, Rev. 1, San Diego, California, February 1986.

GA, 1986-3* GA Technologies, Inc., "Bridging Methods for Standard HTGR Licensing Bases," HTGR-86-039, Rev. 2, San Diego, California, February 1986.

GA, 1987-1* GA Technologies, Inc., "Methodology for Development of Principal Design Criteria for the Standard HTGR," HTGR-85-166, Rev. 1, San Diego, California, January 1987.

GA, 1987-2* GA Technologies, Inc., "Safety-Related Structures, Systems, and Components for the Standard MHTGR," DOE-HTGR-87-003, San Diego, California, January 1987.

GA, 1987-3* GA Technologies, Inc., "Licensing Basis Events for the Standard MHTGR," DOE-HTGR-86-034, Rev. 1, San Diego, California, February 1987.

GA, 1987-4* GA Technologies, Inc., "MHTGR Core Nuclear Uncertainty Analysis," DOE-HTGR-87-085, San Diego, California, August 1987.

Gavigan, 1986 Francis X. Gavigan (DOE), letter to Themis P. Speis (NRC) transmitting Preliminary Safety Information Document, September 30, 1986.

GCRA, 1986 Gas-Cooled Reactor Associates, "Utility/User Requirements for the Modular High Temperature Gas-Cooled Reactor Plant," GCRA 86-002, Rev. 2, San Diego, California, September 1986.

Generic Letter 84-01 U.S. Nuclear Regulatory Commission, "NRC Use of the Terms 'Important to Safety' and 'Safety Related,'" January 5, 1984.

IEEE Std. 279 Institute of Electrical and Electronics Engineers, "Criteria for Protection Systems for Nuclear Power Generating Stations" (ANSI N42.7-1972), New York, N.Y., 1971.

*Indicates that the document is classified as "applied technology" and is not in the public domain. Requests for these documents should be made through the U.S. Department of Energy, Washington, D.C.

IEEE Std. 308 — Institute of Electrical and Electronics Engineers, "Criteria for Class 1E Power Systems for Nuclear Power Generating Stations," New York, N.Y., 1974.

IEEE Std. 344 — Institute of Electrical and Electronics Engineers, "Standard for Seismic Qualification of Electrical Equipment and Components for Nuclear Power Generating Stations," New York, N.Y., 1975.

IEEE Std. 383 — Institute of Electrical and Electronics Engineers, "Standard for Type Test of Class 1E Electric Cables, Field Splices, and Connections for Nuclear Power Generating Stations, New York, N.Y., 1974.

IEEE Std. 384 — Institute of Electrical and Electronics Engineers, "Standard Criteria for Independence of Class 1E Equipment and Circuits," New York, N.Y., 1981.

IEEE Std. 450 — Institute of Electrical and Electronics Engineers, "Recommended Practice for Maintenance Testing, and Replacement of Large Lead Storage Batteries for Generating Stations and Substations," New York, N.Y., 1975.

IEEE Std. 484 — Institute of Electrical and Electronics Engineers, "Recommended Practice for Installation Design and Instalation of Large Lead Storage Batteries for Generating Stations and Substations," New York, N.Y., 1974.

IEEE Std. 485 — Institute of Electrical and Electronics Engineers, "Recommended Practice for Sizing Large Lead Storage Batteries for Generating Stations and Substations," New York, N.Y., 1978.

IEEE Std. 603 — Institute of Electrical and Electronics Engineers, "Standard Criteria for Safety Systems for Nuclear Power Generating Stations" (ANSI N41.30), New York, N.Y., 1980

Kerr, 1988-1 — William Kerr (Chairman ACRS), letter to Honorable Lando W. Zech, Jr. (Chairman NRC), "Report on Key Licensing Issues Associated With DOE Sponsored Reactor Designs," July 20, 1988.

Kerr, 1988-2 — William Kerr (Chairman ACRS) letter to Honorable Lando W. Zech, Jr. (Chairman NRC), "Preapplication Safety Evaluation Report for the Modular High Temperature Gas Cooled Reactor," October 13, 1988.

Merrill, 1973 — M. H. Merrill, "Nuclear Design Methods and Experimental Data in Use at Gulf General Atomic," Gulf-GA-A12652 (GA-LTR-2), Gulf General Atomic Co., San Diego, California, July 1973.

Minarick, 1988	J. W. Minarick et al., "Review of the Standard Modular High Temperature Gas Gooled Reactor Probabilistic Risk Assessment," Science Applications International Corporation, Oak Ridge, Tennessee, March 1988.
Morris, 1987	Bill M. Morris (NRC), letter to Francis X. Gavigan (DOE) Enclosure 4, "Physical Protection Requirements for the MHTGR," July 6, 1987.
Moses, 1988	David L. Moses, "Technical Evaluation Report, Standard Modular High-Temperature Gas-Cooled Reactor (MHTGR)," Oak Ridge National Laboratory, Oak Ridge, Tennessee, April 1988.
Neylan, 1987*	A. J. Neylan (GA), letter to T. L. King (NRC) transmitting enclosure "MHTGR Heat and Mass Transfer Data," GA/NRC-006-86, April 3, 1987.
Neylan, 1988-1*	A. J. Neylan (GA), letter to P. M. Williams (NRC), "Fuel Specification and Performance Mechanisms," May 17, 1988.
Neylan, 1988-2*	A. J. Neylan (GA), letter to P. M. Williams (NRC), transmitting document, "Systematic Assessment of MHTGR Design's Insensitivity to Operator Error," June 30, 1988.
NUREG-0111	U.S. Nuclear Regulatory Commission, "Evaluation of High-Temperature Gas-Cooled Reactor Fuel Particle Coating Failure Models and Data," November 1976.
NUREG-0396	U. S. Nuclear Regulatory Commission, "Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Nuclear Power Plants" (U.S. Environmental Protection Agency, EPA 520/1-78-16), December 1978.
NUREG-0654	U.S. Nuclear Regulatory Commission, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants" (Federal Emergency Management Agency, FEMA-REP-1, Rev. 1), November 1980, p. 12.
NUREG-0700	U.S. Nuclear Regulatory Commission, "Guidelines for Control Room Design Reviews," September 1981.
NUREG-0713	U.S. Nuclear Regulatory Commission, "Occupational Radiation Exposure at Commercial Nuclear Power Reactors - 1983," Vol. 5, March 1985.

*Indicates that the document is classified as "applied technology" and is not in the public domain. Requests for these documents should be made through the U.S. Department of Energy, Washington, D.C.

NUREG-0737, Supplement 1	U.S. Nuclear Regulatory Commission, "Clarification of TMI Action Plan Requirements," January 1983.
NUREG-0800	U.S. Nuclear Regulatory Commission, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," LWR Edition, April 1984.
NUREG-0908	U.S. Nuclear Regulatory Commission, "Acceptance Criteria for Evaluation of Security Plans," August 1982.
NUREG-1226	U.S. Nuclear Regulatory Commission, "Development and Utilization of the NRC Policy Statement on the Regulation of Advanced Nuclear Power Plants," June 1988.
NUREG-1251 (Draft)	U.S. Nuclear Regulatory Commission, "Implications of Accident at Chernobyl for Safety Regulation of Commercial Nuclear Power Plants in the United States," Chapter 6, "Graphite-Moderated Reactors," August 1987.
NUREG-75/014 (formerly WASH-1400)	U.S. Nuclear Regulatory Commission, "Reactor Safety Study: An Assessment of Risks in U.S. Commercial Nuclear Power Plants," December 1975.
NUREG/CR-0509	U.S. Nuclear Regulatory Commission, "Emergency Power Supplies for Physical Security Systems," Oak Ridge Y-12 Plant, Oak Ridge, Tennessee, October 1979.
NUREG/CR-1327	U.S. Nuclear Regulatory Commission, "Security Lighting Planning," Mason Co, Inc, Lexington, Kentucky, April 1980.
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, Upton, New York, September 1987.
SECY 86-368	U.S. Nuclear Regulatory Commission, "NRC Activities Related to the Commission's Policy on the Regulation of Advanced Reactors," December 10, 1986.
SECY 88-202	U.S. Nuclear Regulatory Commission, "Standardization of Advanced Reactor Designs," July 15, 1988.
SECY 88-203	U.S. Nuclear Regulatory Commission, "Key Licensing Issues Associated With DOE Sponsored Advanced Reactor Designs," July 15, 1988.
Walker, 1987	Lloyd P. Walker (GCRA), letter to distribution (NRC and others) transmitting PSID Amendment 6, July 30, 1987.

APPENDIX A

SUMMARY OF ORNL INDEPENDENT ANALYSIS IN SUPPORT OF SAFETY EVALUATION REPORT

S. J. Ball, J. C. Conklin, and J. C. Cleveland
Oak Ridge National Laboratory
Oak Ridge, Tennessee

1. Introduction

This SER section summarizes the independent analyses done at ORNL on both the postulated long-term core heatup accident scenarios and the reactivity insertion accident scenarios. More details on the code development and the analysis descriptions may be found in backup ORNL/TM reports (Refs. 1 and 2). The purpose of this section is to give brief overviews of the codes and the methodology used, and to summarize the more significant findings. Recommendations for additional work are included.

2. MORECA Code Description

The MORECA code was developed to perform independent analyses on a broad range of Modular HTGR long-term core heatup accident scenarios. MORECA is based on the ORNL ORECA code, which was developed under NRC research program sponsorship, and which has been in use at ORNL and elsewhere since 1975 (Ref. 3). ORECA has been used in accident studies requiring core thermal analysis of Fort St. Vrain (FSV), the 2240-MW(t) design, and several other HTGR designs. ORECA-FSV has been partially verified for numerous cases vs. data and GA codes. ORNL has an ongoing (but "slow-paced") verification test arrangement in place with FSV. Verification of other versions of ORECA has been limited to comparisons with GA and BNL code calculations. These activities are continuing to verify model applicability to wider classes of transients and accidents.

The MORECA model for the core uses a point heat capacity node for the 66 fuel and 139 reflector elements (vs. one node per 7-element region in ORECA) in each of the 14 axial regions. The core is thus represented by 205 times 14 equals 2870 nodes. This finer structure was thought to be appropriate because of the high sensitivity of low fuel failure rates to time-at-temperature transients in the range near 1600°C. This structure also allows for investigations of azimuthal asymmetry, a feature that other current MHTGR core codes do not have. The PSID function for decay heat is used for the reference case calculations, and is considerably more conservative than the current "best estimate" function (Ref. 4).

Variable core thermal properties as supplied by GA were used for the reference case calculations (Ref. 5). These properties are functions of both temperature and radiation damage. Fully-irradiated thermal properties are used for the fuel, the inner reflector, and the ring of outer reflector elements adjacent to the fuel. Currently, the MORECA model does not include effects of annealing, which increases the thermal conductivity of the fuel and adjacent reflectors as the core heats up during the hypothetical accidents.

Coolant flow in the core is modeled over the full ranges expected in both normal operation and accidents, including pressurized and depressurized, forced and natural circulation, upflow and downflow, and turbulent and laminar flow. Flow in each of the fuel elements is modeled explicitly. Flow in the inner and outer reflectors is assumed to be uniformly distributed in the spaces between blocks, starting out with a user-input fraction of the total forced-circulation flow.

Other features of the current MORECA model are summarized as follows:

1) The core barrel and vessel are each represented by 7 axial X 4 radial (quadrants) nodes, plus nodes corresponding to the regions opposite the inlet and outlet plenums. The "roof" and "floor" heat shields are each represented by 205 "coverplate" nodes in the reference model. (This is somewhat of an overkill carried over from a previous model in which individual coverplate failures were of interest. "Coverplate" failure is not an issue here because the shields are made of the high-temperature material Alloy 800 instead of carbon steel. A simpler, more appropriate model has been developed subsequently.) The insulation resistance and radiation shielding of the upper plenum insulation cover is also modeled explicitly.

2) The Reactor Cavity Cooling System (RCCS) model incorporates detailed heat transfer and natural circulation cooling calculations for panel nodes corresponding to adjacent vessel nodes. Independent flow and heat transfer (radiative and convection) equations for each of 4 quadrant panels allow simulation of the full range of expected performance, and degraded states including partial and total system failures.

3) A Shutdown Cooling System (SCS) model has been installed but not yet fully implemented. Use of the SCS model is of particular interest for investigating scenarios in which forced circulation flow is restored after long heatup periods during which no circulation was available. In some HTGR designs this can become an operation limiting situation due to the possibility of damage to components downstream of the hot core outlet gases. The SCS inlet path has been designed to withstand such high temperatures, but independent calculations are planned as a check on the margins available.

4) A time-at-temperature fuel failure model developed for the larger HTGRs (Ref. 6) is used to predict fuel failure fractions. Another fuel failure model has been developed from later Goodin work which provides more accurate results in the lower-temperature, lower failure rates regimes, and is in the process of being added to MORECA (Ref. 7). This task was given a low priority because in all bounded accident cases simulated to date, the maximum fuel temperatures have not reached temperatures high enough (long enough) to result in fuel failures. In some extremely low probability cases in which a complete functional failure of the RCCS is assumed, the predicted maximum fuel temperatures do reach failure levels, and the improved model will be of more interest.

3. Summary of MORECA Runs and Findings

There are two general classes of heatup accidents studied using the MORECA code in which the RCCS is assumed operational. The first is the rapid depressurization and immediate Loss of Forced Circulation (LOFC) with scram, with no subsequent primary coolant system forced cooling. This case corresponds to the SRDC-11 case in the PSID. In the reference case depressurization LOFC calculation (Fig. A1), peak temperatures are reached after 4-5 days. There is no fuel failure, as the maximum peak fuel temperature (1482°C, 2699°F) is well below the 1600°C nominal "limit". The maximum vessel temperature (478°C, 893°F) is below the 1000°F extended code limit for a depressurized vessel. These results are generally in good agreement with PSID values except for vessel temperatures, where the PSID's maximum was less than 427°C (800°F). Reasons for this discrepancy are being investigated.

The second class of heatup accident with RCCS is the pressurized LOFC with scram, which corresponds to the DBE-1 case in the PSID. Results are shown in Fig. A2. The maximum fuel temperatures predicted are even lower than those in the depressurized LOFC case, and concern for any fuel damage is nil. The primary concern is for vessel temperature (maximum 469°C, 876°F), which exceeds the 800°F extended code limit for a pressurized vessel. The corresponding PSID prediction, using the GA PANTHER code, was 400°C (750°F). Some of the discrepancies were found to be due to simplifications in the PANTHER code that GA plans to address in the next stages of the design; however, some others have not yet been resolved. As in the PSID calculation, the MORECA prediction of maximum primary system pressure (7.05 MPa, 1022 psia) was not high enough to actuate the relief valve (7.18 MPa, 1041 psia); however, the MORECA assumptions of steam generator cavity temperatures, which have a significant effect on pressure, were quite simplified and arbitrary. The extent of the over-temperature at pressure predicted here would not be expected to cause a vessel failure; however, considering the uncertainties involved in the temperature predictions, means should be provided to depressurize, and vessel temperature monitoring should be provided. Monitoring would provide a basis for regulators to judge if restart following an LOFC should be allowed.

3.1 Variations

Many variations of these two classes of accidents were studied to observe sensitivities of the severity of the predicted results to both parametric (modeling) and operational assumptions.

Of the many parametric variations in the "reference" depressurized and pressurized LOFC cases, three were found to be of major significance in determining the safety related outcome of the predictions: (1) Assumptions of fuel and reflector thermal conductivities; (2) Use of the conservative (PSID) afterheat relationship vs the "best estimate" curves; and (3) Variations in assumed RCCS performance, including effects of assumed emissivity values that have a direct effect on transfer of heat from the core blocks to the RCCS panels.

The reference case assumption for reflector conductivity is that only the central reflector and first ring of elements surrounding the fuel suffer significant radiation damage (along with the fuel itself). However, for the case of relatively unirradiated (or annealed) elements, the thermal conductivities would be considerably higher. Data on effective fuel and graphite conductivities are typically difficult to quantify due to effects of impurities, geometries, gaps, thermal radiation effectiveness, and annealing that may take place during measurements. Hence we have assumed that there may be wide variations in the core conductivity values, due both to data uncertainties and actual changes due to operating history.

Typically, increasing the fuel and outer reflector conductivities will enhance heat transfer to the RCCS heat sink in LOFC heatup accidents, resulting in lower peak fuel temperatures. Results showed that several-hundred-degree variations in peak fuel temperatures were possible due to reasonable variations in assumed conductivities. While the "low end" values of conductivity were used in the reference case (resulting in acceptable peak fuel temperatures for the limiting-case depressurized LOFC), it is seen as essential that the conductivity relationships be carefully verified to provide assurance of negligible fuel failure. The maximum vessel temperature prediction is also affected by core thermal conductivity assumptions. While it was expected that increased core conductivities would result in higher peak vessel temperatures, in fact the opposite was true, at least for the cases where the axial conductivity was assumed to increase along with the radial. Increased conductivities (favorably) changed the times at which the peak temperatures occurred, and made the temperatures more uniform (axially and circumferentially), thus reducing the gradients.

Use of the "best estimate" afterheat curve (vs. the reference case, considerably more conservative PSID relationship), results in predicted peak fuel temperatures about 150 to 250°C lower for the depressurized LOFC (depending on other parameter assumptions). There is less of an effect for the pressurized cases. Peak vessel temperatures for the best estimate afterheat cases are typically about 50°C lower. Use of the Fort St. Vrain FSAR afterheat curve gives results nearly identical with those that use PSID values.

While the performance of the RCCS during postulated heatup accidents has relatively little effect on peak fuel temperatures, it can have a significant effect on peak vessel temperatures. For example, for a depressurized LOFC in which the RCCS was assumed to be failed totally for a one day period after the LOFC and scram resulted in a maximum fuel temperature increase of less than 20°C over the case of no RCCS failure. Assuming emissivity values of 0.5 (vs 0.8 in the reference case) for the RCCS panels and vessel walls increases the predicted peak fuel temperature in depressurized LOFCs by only about 30°C, but the peak vessel temperature increases by about 120°C. Hence it is important that the critical emissivity values be maintained in the 0.8 range. In depressurized LOFCs where one of the 4 quadrant RCCS panels is substantially blocked (friction factor times 200), the maximum fuel temperature goes only about 10°C higher than without the blockage. The vessel temperature opposite the failed panel, however, will exceed its design limit in 1 to 2 days. Hence, the RCCS performance monitoring must be such that it could detect such partial RCCS failures (especially for pressurized LOFCs) so that suitable corrective actions (such as depressurization) could be taken.

Besides the three major (important) variations noted, many other variations were studied which were all shown to have less significant effects on the safety-related outcome of the accidents:

(1) An arbitrary cooldown period following the scram, which makes the effective "initial condition" temperatures of the core lower; or conversely, an assumption of arbitrarily degraded RCCS panel performance for a relatively short period following the scram, increasing the "initial" core temperatures. While these variations had only a relatively small effect on maximum fuel temperatures, localized or intermittent failures in the RCCS heat removal function had significant effects on maximum vessel temperatures;

(2) Variation in the assumed initial reflector bypass flow fraction, as noted previously. In earlier MORECA calculations of pressurized LOFCs in which thermal insulation in the upper vessel region was omitted, a large (~10%) assumed bypass flow resulted in significantly higher maximum vessel temperatures, compared with assuming no bypass flow (as is done in the PSID). However, after adding the insulation, the maximum vessel temperatures for the pressurized LOFC appeared in the area adjacent to the fuel, and assumed bypass flow fraction variations had little effect on maximum vessel temperature. Maximum fuel temperatures are affected by bypass flow, but stay well below failure limits in all cases;

(3) Variation in the assumed initial and shutdown peaking factors, both axially and radially. This variation addresses the difference between the power distribution during operation (as given in the PSID, and as used in the reference calculations even after a scram), and the power distribution that is "smeared" out considerably, which more realistically models post-scram gamma heating. An interesting aspect of this particular sensitivity study was that in the pressurized LOFC case where a uniform post-scram power distribution was assumed, the nonuniform azimuthal temperatures persisted throughout the accident as a result of the initial nonuniform fuel temperatures and natural convection flow patterns set up at the start;

(4) Variations in RCCS flow loss coefficients (i.e. for increased friction factors or partial blockage) and air side heat transfer coefficients. Variations over relatively wide ranges had minor effects on RCCS heat removal performance; and

(5) Variations in outdoor temperature (RCCS inlet air temperature). The reference case assumed 29°C (85°F), while the maximum design temperature is 43°C (110°F). Peak vessel temperatures increase about one degree for every two degree rise in ambient.

3.2 Complete RCCS Failure

A "complete" failure of the RCCS is currently seen as a non-mechanistic failure, since no reasonable total failure mechanisms have been postulated. In the current calculation, the RCCS structure with its insulation between the riser and downcomer is assumed to be in place, but there is no air flow. Conduction and thermal radiation to the concrete silo is modeled simplistically, and credit is taken for the concrete heat capacity. No credit is taken for heat losses to the upper and lower heads. The results are shown in Fig. A3. Although the peak fuel temperature of 1606°C (2923°F) exceeds the 1600°C "limit", the predicted fuel failure is not significant. The vessel temperature, however, exceeds code values in about one day, and reaches dangerously high temperatures within two to four days.

4. Short-Term Response to Flow and Reactivity Transients

The short-term thermal response of the MHTGR was analyzed for a range of flow and reactivity transients. The transients examined included LOFC without scram, moisture ingress, spurious control rod group withdrawal, control rod ejection, and a rapid core cooling without scram. Certain actions of the plant control and protection system were assumed not to function in cases where very low probability events were analyzed. The computer code used for these investigations is a coupled neutron-kinetics heat transfer model of an average fuel, moderator, and coolant region. The computational methods were developed originally for pebble bed HTGRs, and the resulting code predictions compared well with experimental measurements (Ref. 8).

To the extent possible, the calculations were made independent of DOE input. However, values for certain parameters (particularly temperature coefficients of reactivity, reactivity worth of moisture, moisture ingress rates, and control rod worths) were obtained from the PSID and other DOE sources. These parameters are crucial to the favorable outcome of the current predictions.

The results of all of the postulated accidents were mild. In the LOFC without scram, the rise in maximum and average fuel temperatures is less than 150°C, well below any level of concern. In the moisture ingress accident due to an offset break in a single steam generator tube, no scram on high power-to-flow or high moisture and no flux controller action were assumed. Endothermic cooling by steam-graphite reactions was also ignored. The results (Fig. A4) were milder than those of the PSID, primarily because it was assumed here that the steam doesn't collect in the core (PSID assumption), it carries right on through. An assumed trip of the main circulator after about 2 min limits the gradual power excursion. These results are of course sensitive to the assumptions of reactivity worth of injected steam and the total quantity of injected steam (ingress rate), so larger reactivity insertions would result from higher worths or more ingress (e.g. from a tube header break).

In the postulated control rod group withdrawal event, the group of highest worth (~$4) was withdrawn in about 4 min without the normal flux controller action but with a scram on high power-to-flow. The maximum power increase was to less than 160% and the fuel temperature rise was small.

In the rod ejection accident (which is non-mechanistic), a highest worth rod (~$2) is rapidly removed. While the average fuel temperature rise is small, there is the possibility of significant overheating of some local fuel; hence this accident should be precluded by design. The point kinetics model used is not considered to be conservative for rapid, local reactivity insertions.

In a rapid cooldown ("cold slug") accident, the purpose was to investigate the susceptibility of the MHTGR to startup accidents. Although the power peak is high (2 to 4 times normal full power), the maximum fuel temperatures remained within normal bounds.

5. Conclusions

From the LOFC heatup accident analyses, it is evident that the current MHTGR design is not susceptible to significant fuel failure from postulated accidents even from very low probability or even from certain drastic, non-mechanistic events. The ORNL results generally corresponded well with independent calculations by DOE contractors and by BNL. Considering the fact that these are calculations of the most serious types of accidents that can be reasonably postulated, the fact that there is such good general agreement indicates that the analyses are relatively straightforward and therefore credible. The one major area of concern was with possible vessel overheating, and that would not be considered an immediate safety concern unless RCCS or partial RCCS failures occurred. Sensitivity studies showed that the most crucial safety-related parameter or operational uncertainties were the core thermal conductivities, the afterheat curve, and the effective RCCS heat removal performance.

For the reactivity insertion studies, it can be concluded that given the current nuclear parameter functions as input, the results of the postulated accidents are quite acceptable. Independent checks (and further work in an R&D program) should be made on the reactivity worth of steam in the core, the effect of core moisture on control rod worth, and mechanisms for more massive moisture ingress rates.

REFERENCES

1. S. J. Ball, J. C. Conklin, "Modular HTGR Heatup Accident Analyses", ORNL/TM-11088 (in preparation).

2. J. C. Cleveland, "Modular HTGR Short-Term Thermal Response to Flow and Reactivity Transients", ORNL/TM-11087 (in preparation).

3. S. J. Ball, "ORECA-I: A Digital Computer Code for Simulating the Dynamics of HTGR Cores for Emergency Cooling Analyses", ORNL/TM-5159 (1976).

4. HTGR-86-109 (DOE).

5. A. J. Neylan (GA), "MHTGR Heat and Mass Transfer Data", GA/NRC-006-86 (4/3/87), letter to T. L. King (NRC).

6. D. T. Goodin (GA), A 16508, October, 1983.

7. D. T. Goodin (GA), "Fuel Failure Model for MHTGR", HTGR-85-107, Rev. 1 (DOE).

8. J. C. Cleveland, "ORNL Analyses of AVR Performance and Safety", IAEA-TECDOC-358 (May 1985).

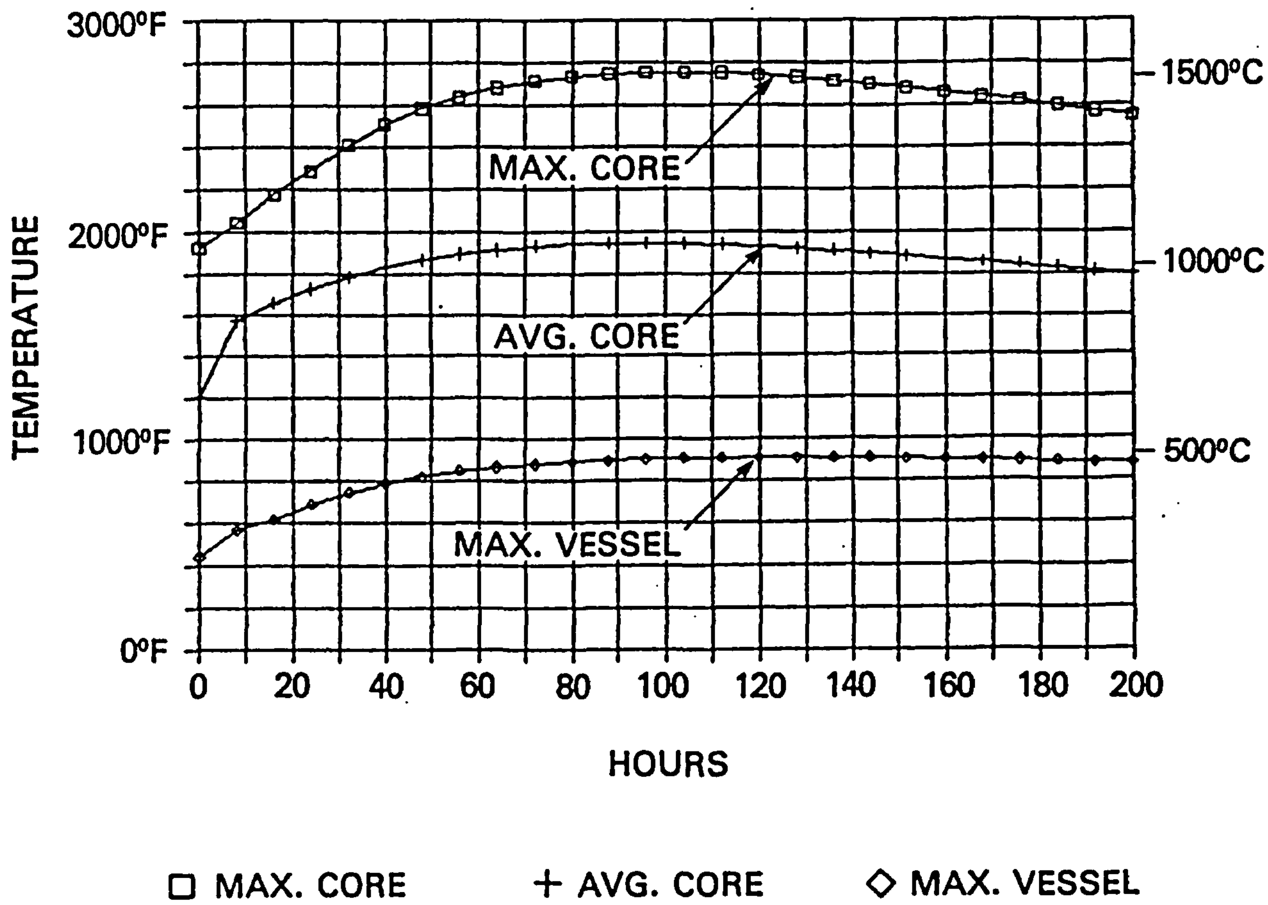

Figure A.1 Depressurized conduction cooldown with RCCS temperatures of core and vessel vs. time (MORECA reference case)

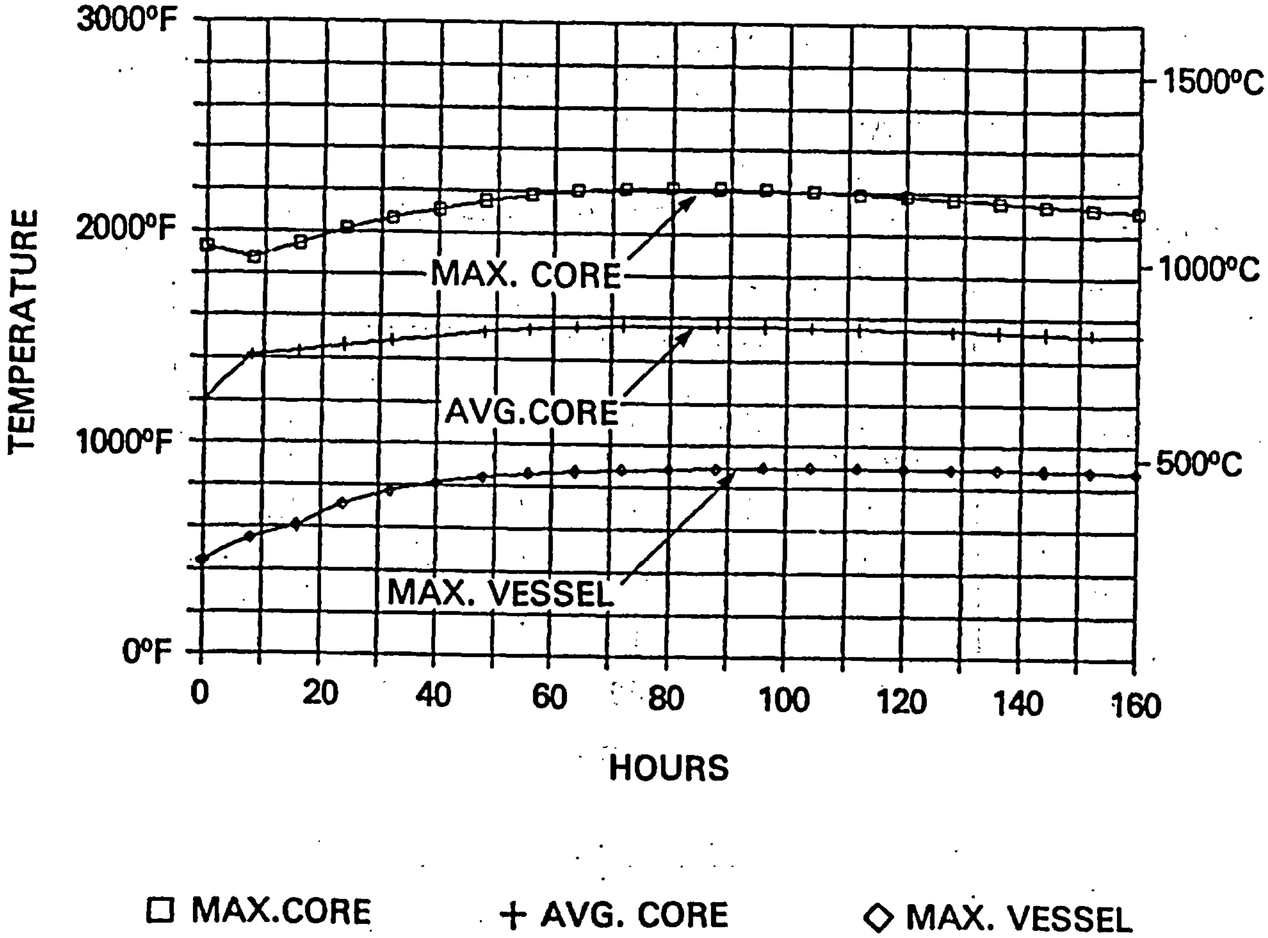

Figure A.2 Pressurized conduction cooldown with RCCS temperatures of core and vessel vs. time (MORECA reference case)

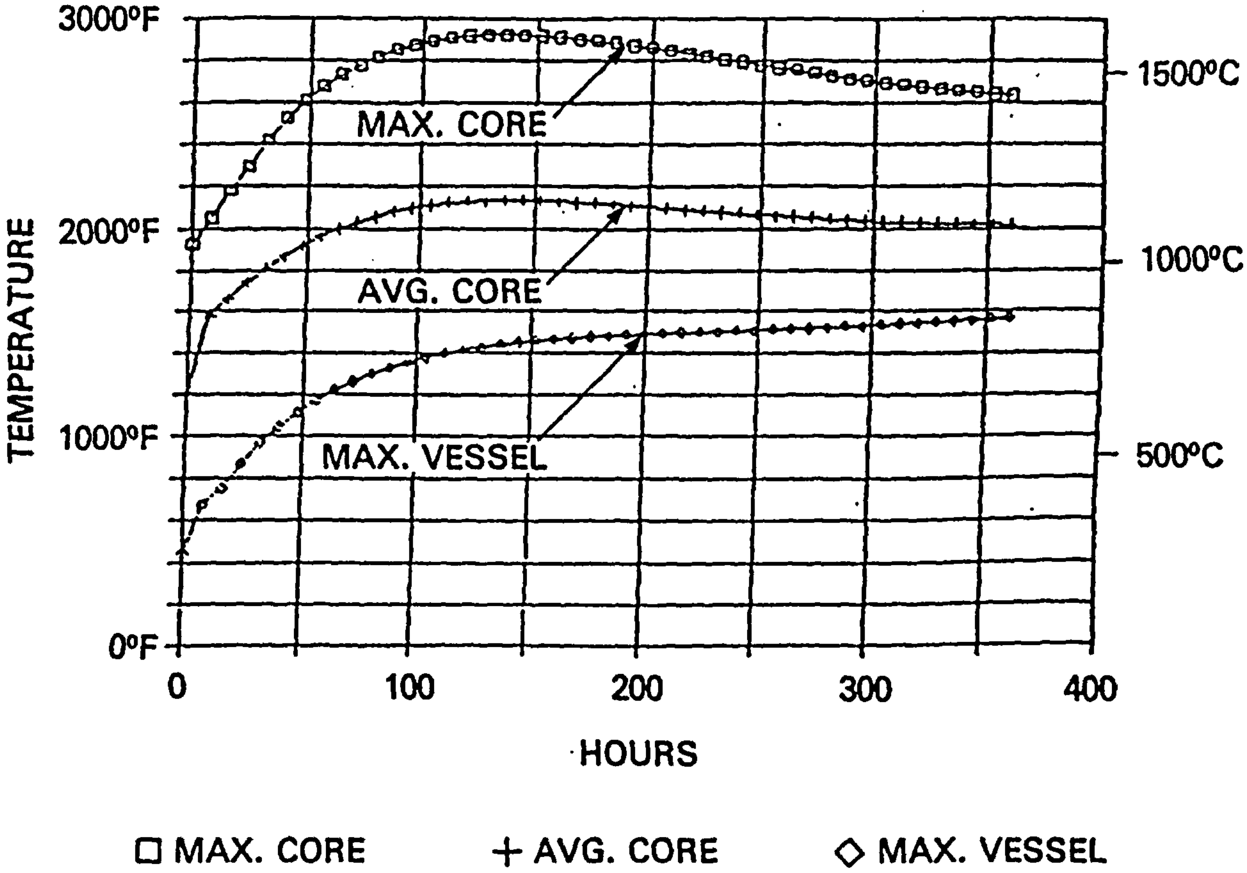

Figure A.3 Depressurized conduction cooldown without RCCS temperatures of core and vessel vs. time (MORECA reference case)

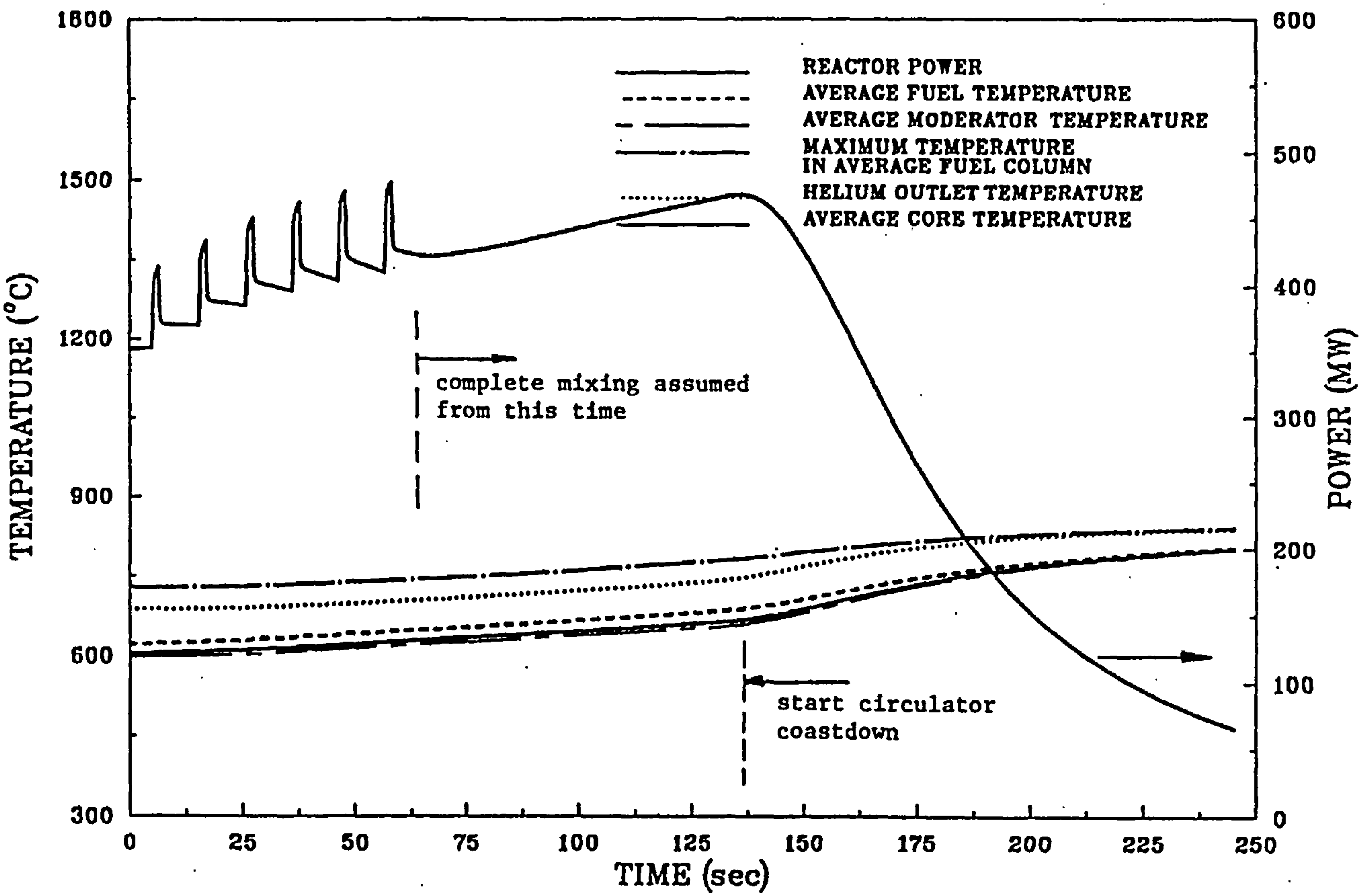

Figure A.4 MHTGR response to single steam generator tube break

APPENDIX B

SUMMARY OF BNL INDEPENDENT ANALYSIS IN SUPPORT OF SAFETY EVALUATION REPORT

Peter G. Kroeger
Department of Nuclear Energy
Brookhaven National Laboratory
Upton, New York 11973

SUMMARY OF

BNL INDEPENDENT ANALYSIS IN SUPPORT OF MHTGR SAFETY EVALUATION REPORT

Peter G. Kroeger

Department of Nuclear Energy

Brookhaven National Laboratory

Upton, NY 11973

1. SAFETY EVALUATION OF THE MHTGR DURING DEPRESSURIZED CORE HEATUP TRANSIENTS WITH FUNCTIONING RCCS

The scenarios considered in this section assume scram, depressurization, and loss of all forced circulation to occur at the beginning of the accident, with conduction and radiation heat transfer from the core to the passive RCCS, which continues to function normally. Corresponding events are considered in Chapter 15 of the PSID in DBE-11 and SRDC-6 to 11.

During normal full power operation the RCCS continually removes about 0.8 MW from the reactor vessel. In the early phases of the accident scenario the decay heat exceeds the heat removal by the RCCS, and the excess energy is stored in the core, resulting in a gradual core heatup. After 60 to 70 hr the RCCS heat removal exceeds the decay heat, and the system begins to cool down. In the best estimate case, a peak fuel temperature of about 1370°C is reached after 55 hr and a peak vessel temperature of about 425°C occurs after 91 hr. These best estimate temperatures are lower than those cited in the more conservative PSID evaluations. Typical results of reactor temperatures and heat flows for each transient are shown in Figure 1.

The major emphasis of our analysis was to independently verify the PSID evaluations and to identify the parameters which, within their uncertainty bounds, could have a significant safety effect on the accident transient. Peak fuel and vessel temperatures during the transient were the output parameters of primary concern. Excessive fuel temperatures can lead to fission product release. Vessel temperatures in excess of the maximum allowable ASME code values could prevent future reuse of the pressure vessel.

Numerous parametric variations on parameters such as in-core gaps between fuel elements, initial graphite irradiation damage, air inlet temperature to the RCCS, as well as thermal emissivities of the reactor and RCCS materials have shown that variations in these parameters have no major impact on the peak fuel temperatures. The vessel and RCCS thermal emissivities did have a significant effect on the vessel temperatures, indicating that this parameter should be controlled during manufacture and operation, primarily by avoiding any polishing or painting of the steel surfaces.

The two parameters having the most significant impact on the fuel and vessel temperatures were the decay heat and the effective thermal conductivity of the fuel elements and reflector blocks. Parametric evaluations were performed in order to establish the effect of these two parameters on peak fuel and vessel temperatures. The results are shown in Figure 2.

Current DOE data appear to indicate that very few, if any, fuel failures are likely to occur in the 1600 to 1800°C temperature range. Nevertheless, a value of 1600°C has frequently been cited as the threshold below which one is assured of no additional fuel failures, and no fission product releases beyond the circulating and plated out inventory. At temperatures of 2200°C and above, massive fuel failures would be expected.

The results in Figure 2 show that a 30% increase in decay heat or a 37% reduction in effective thermal conductivity would be required in order to reach 1600°C peak fuel temperatures. Significantly larger margins exist before the 2200°C threshold would be reached. A 27% increase in decay heat was found to cause peak vessel temperatures of 480°C, the value beyond which the restart capability of the vessel might be compromised. Thus, in operation with RCCS, significant performance margins exist before fuel failures and additional fission product release would be expected. However, the evaluations show that a high confidence in the decay heat function and effective core thermal properties is required to assure that vessel temperatures do remain within safe bounds.

2. SAFETY EVALUATION OF THE MHTGR DURING DEPRESSURIZED CORE HEATUP TRANSIENTS WITHOUT FUNCTIONING RCCS

The passive RCCS has a very low failure probability, and even in case of catastrophic failures, only parts of the system would be likely to fail, resulting in partial flow blockages and/or partial loss of draft. Parametric evaluations of RCCS performance have shown it to be highly "self-adjusting" (large increases in flow resistance lead to some flow reduction and higher air exit temperatures, with a relatively small loss in total energy removed). Nevertheless, as a limiting case, depressurized core heatup without cooling by the RCCS is being considered in this section.

In order to protect the surrounding concrete surfaces, the RCCS design includes thermal insulation. Additional shielding and thermal insulation are provided at the top and the bottom of the reactor cavity. This thermal insulation is the most significant heat transfer barrier in any heatup scenarios without functioning RCCS. The failure assumed here is a most conservative case, in that it postulates a worst case combination of:

1. Eliminating all air flow by blocking all flow passages completely, while
2. keeping all thermal insulation in place.

Adding more conservatism, for our Base Case evaluation, a concrete of relatively low thermal conductivity and a poorly conducting soil (clay) were assumed. Several parametric variations in concrete/soil properties and configurations were evaluated. A corresponding case is considered in Appendix G, Section G.2 of the PRA report for the MHTGR.

Our analyses found that the peak core temperatures exceeded those for the corresponding cases with RCCS by about 35°C only, and were essentially independent of concrete and soil conditions, since these structures were still relatively cool at 78 hr, when the core temperatures peaked. However, the vessel temperatures eventually reached levels between 700 and 800°C, typically peaking between 400 and 1,200 hr, i.e., weeks after the onset of the accident. Poorer concrete and soil conditions affected the peak vessel temperatures slightly, but greatly slowed down the ultimate cooldown. Several regions of the concrete walls of the reactor silo reached temperatures as high as 700°C. Thus, at least partial failure of these structures, weeks after the onset of the accident, is not precluded.

Parametric variations of decay heat and core effective thermal conductivity (with RCCS failed) gave only slightly smaller margins than the corresponding cases with RCCS as shown in Figure 2: a 27% increase in decay heat and a 33% reduction of the core effective thermal conductivity were required to reach peak fuel temperatures of 1600°C. However, unacceptable vessel and concrete temperatures are possibly reached. A 40% increase in decay heat brings the peak vessel temperature to 1000°C (however, only after 6 weeks). While there is no specific vessel failure temperature or failure mode, mechanistic accident scenarios can be envisioned here, during which some fuel failures occur around 100 hr, and subsequent vessel failures occur after several weeks, when core temperatures have already returned to the 1200 to 1300°C range.

To establish whether the reactor cavity could be designed to withstand even these core heatup accidents without functioning RCCS, an evaluation was made for a case of best estimate rather than conservative concrete and soil properties, and without the thermal insulation within the RCCS (this insulation is not really required for the RCCS to function properly under normal or accident conditions). In this case, the vessel temperatures peaked about 100°C lower than in the preceding cases, and the peak concrete temperatures at critical areas peaked near 250°C. One local peak concrete temperature at the side wall surface reached 560°C. Thus, a "hardened" reactor silo design with significantly lower vessel and concrete temperatures may be achievable with appropriate design modifications, i.e., elimination or reduction in insulation and proper concrete selection.

In summary, the decay heat and thermal conductivity margins for fuel failures are very close to the corresponding cases with RCCS functioning.

However, higher decay heat levels can significantly impact on the peak vessel and concrete temperatures, and some structural failures of these components at very long times are possible.

3. EVALUATION OF LARGE AIR INGRESS SCENARIOS

For significant amounts of air to enter the core large failures of the primary loop pressure vessel system must be postulated. These could be either in the form of multiple reactor vessel failures, or in the form of a cross duct double-guillotine break. The latter was assumed here.

In either case, the total gas flow through the core after such a break is limited by the friction pressure drop through the 16 mm diameter and approximately 10 m long coolant holes in the core.

Assuming an unlimited supply of pure air and no recirculation between the gasses exiting and entering the vessel at the break, the core inlet flow ranged from an initial value of 700 kg/hr to about 260 kg/hr for most of the 10 day transient evaluated (for 50 volume % mixtures of helium and air the flow rates were about one third of the above values). Varying the chemical reaction rates and the gas species diffusion coefficients by several orders of magnitude, it was found that all entering air will oxidize, exiting almost exclusively as carbon monoxide, and any uncertainty in reaction rates or diffusion coefficients will only affect the length of the reaction zone. The corresponding graphite oxidation rate was about 60 kg/hr for most of the transient. The thermal contribution from this exothermal graphite-air reaction to the core heatup was small, amounting to only about 10% of the nuclear decay heat.

As the air volume in the reactor and steam generator cavities is generally limited, significant air inflow could last but a few hours, with the inflow being originally a helium air mixture, gradually being replaced by a $He/CO/N_2$ atmosphere. Early during such a scenario, local burning of the exiting CO in the reactor cavity is not impossible, and this could possibly continue for a few hours. For the graphite oxidation to proceed to the point that structural damage inside the core would become possible, an unlimited air supply would have to be available for many days. It should be noted that the air flow into the core and the corresponding amount of graphite reacted, as given here, are larger than those reported by the DOE team. This is apparently due to our use of a finer nodalization in the computation of the downward flowing gas temperatures at the core barrel. While our conclusions are relatively insensitive to these differences in air flow rates, it appears that our results would be the more accurate ones.

4. EVALUATION OF MODERATE WATER INGRESS SCENARIOS

Considering the moderate steam generator break of SRDC-6 (single off-set tube rupture) the long term consequences of graphite oxidation during the subsequent depressurized core heat transient were evaluated.

Subsequent to the shutdown of HTS and/or SCS, their respective flow valves are in a closed position. If they were hermetically closed, only internal in-core recirculation of the He/H_2O mixture of about 18 volume % H_2O would be possible, resulting in very small in-core flow rates of about 0.5 kg/hr. As both valves are designed to permit some bypass flow in their closed position, initial estimates indicate a net circulation between steam generator and core of about 3 kg/hr, which is very minor. However, after the first few hours, the core temperatures are sufficiently high that all H_2O entering the core will react (endothermic), oxidizing about 1 kg/hr of graphite.

The gas exiting the core would have a 30 volume % concentration of water gas ($CO + H_2$). However, it could leave the primary loop only after passing through the steam generator and relief valve train, where it would be strongly diluted. Therefore, it is very unlikely that any combustible mixture could enter the reactor building.

Thus, no serious safety consequences from this accident scenario have been identified. Extension of this work to include large water ingress rates is planned.

5. DEPRESSURIZED CORE HEATUP ACCIDENT SCENARIOS WITHOUT FORCED COOLING AND WITHOUT SCRAM

The case of a depressurization accident without scram and without any forced cooling, but with functioning RCCS was investigated, using the reactivity feedback coefficients from the PSID for an EOC condition and best estimate cross section data supplied by GA. A similar case is presented in Section G.1 of Volume 2 of the PRA report for the MHTGR.

The reactor was found to shut down within about two minutes, due to the negative Doppler feedback coefficients. The power generated during this initial period amounted to about 40 full power seconds, resulting in an average active core temperature rise of about 100°C.

Recriticality due to Xenon decay was observed at about 50 hr, with power spikes occurring about one per hour, with an initial peak of 17 MW, decaying to a final steady level of about 1.2 MW.

Beyond about 120 hr an equilibrium condition was observed, where the positive reactivity due to low Xenon concentration just balances negative reactivity due to elevated fuel temperatures.

The peak core temperatures for this best estimate evaluation reached 1600°C at about 60 hr and peaked at 1760°C at about 120 hr, prevailing at this level for hundreds of hours rather than decaying moderately fast, as in the corresponding accident with scram. Thus some fuel damage and fission product release after 60 hr must be expected. Vessel temperatures of about 550°C would preclude reutilization of the vessel.

Further investigations will consider the case without functioning RCCS, and the sensitivity of the results to variations in the core and reflector temperature coefficients and the cross section data, in particular since in these accident scenarios the peak core temperature is strongly dependent on the Doppler feedback coefficients.

6. FUTURE WORK

The results reported above present a summary of the current state of our ongoing independent analysis. Several further evaluations are planned, or may be added as additional items of concern are identified.

The following further evaluations are currently planned:

- Extend the evaluation of accident scenarios without RCCS to include the case without scram, and to obtain more details on peak concrete temperatures in the silo.
- Include fission product release modelling in the reactor vessel and gas and fission product transport through the reactor building, including initial models for such effects as hydrolysis, plate-out, and lift-off.
- Re-evaluate the large air-ingress scenarios for the case of best estimate core heatup accidents, which may result in somewhat higher air

inflow and graphite oxidation rates than the ones obtained initially for conservative decay heat data with a correspondingly hotter core.

- Extend the case of core heatup without scram to investigate the effect of uncertainties in the reactivity and cross section data currently supplied by DOE.
- Extend the water ingress scenarios to include events with more massive water ingress.
- Evaluate in more detail the effects of water vapor and CO_2 in the reactor cavity on reduction of RCCS performance and peak vessel temperatures.

Reference: Peter G. Kroeger, "Safety Evaluation of MHTGR Licensing Basis Accident Scenarios" (in preparation) (NUREG/CR-5261, BNL-NUREG-52174)

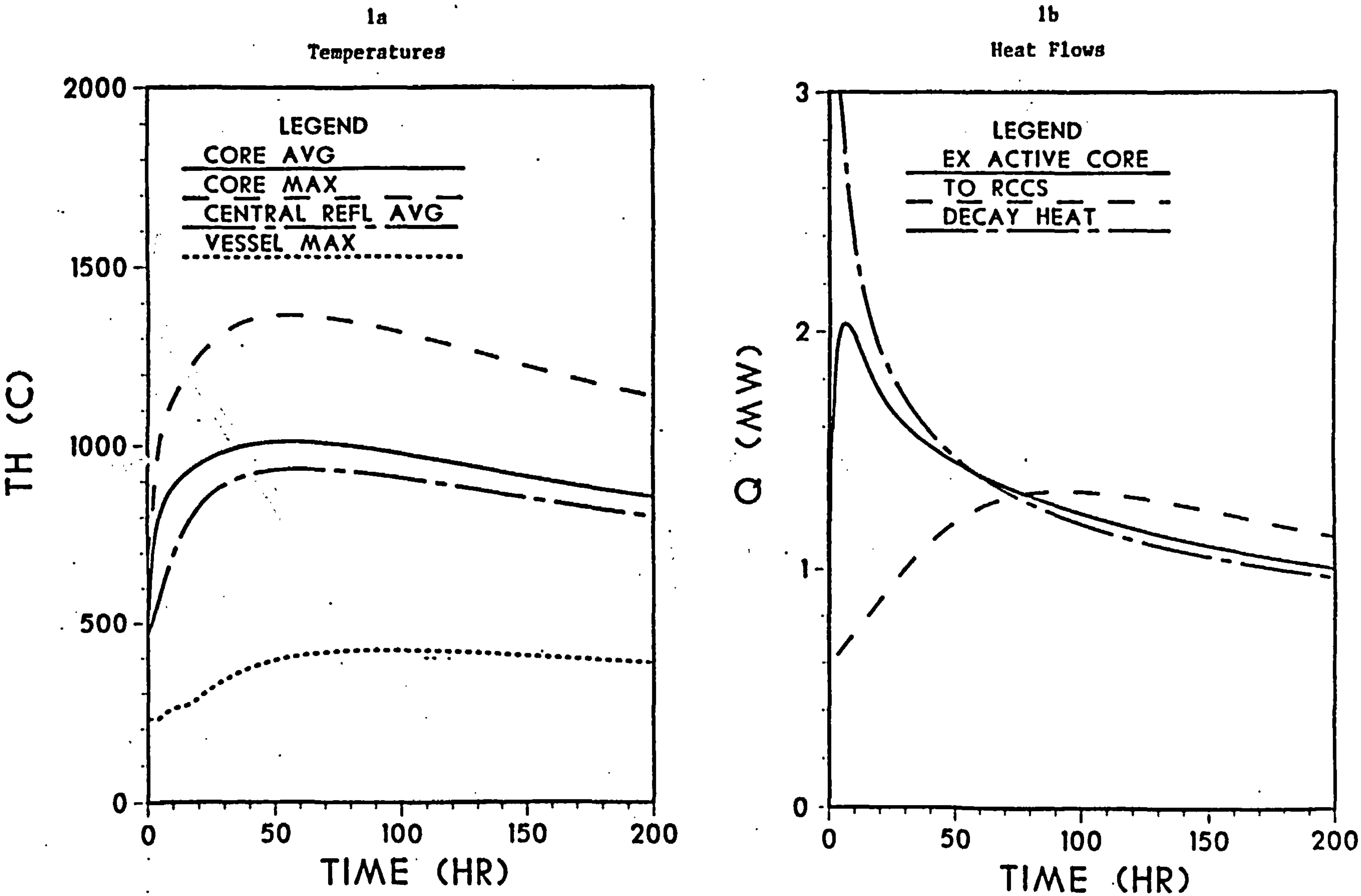

Figure B.1 Core and vessel temperatures and heat flows during best estimaste depressurized core heatup transient

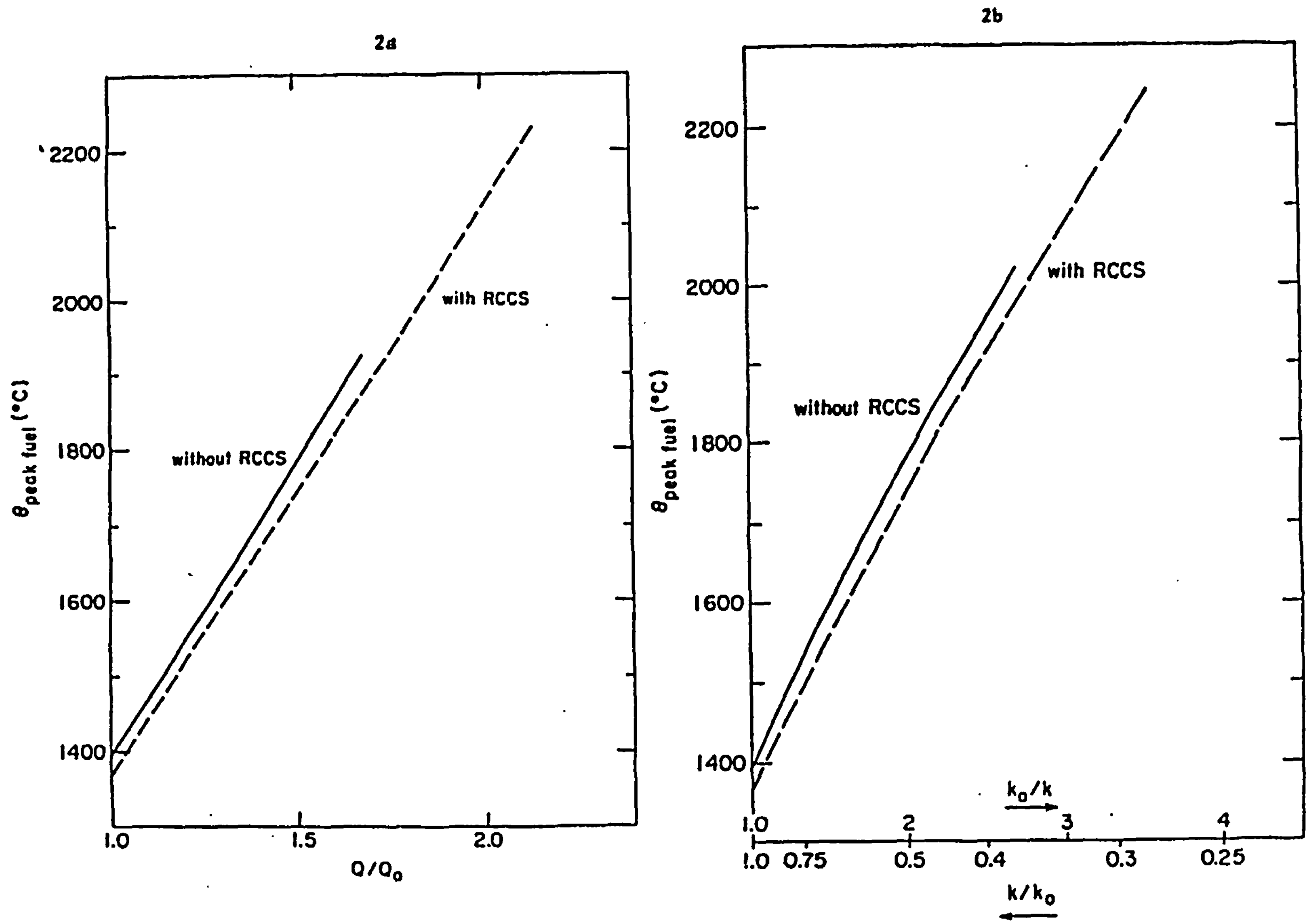

Figure B.2 Peak fuel temperatures during depressurized core heat accident scenarios without force cooling, with and without functioning RCCS at increased levels of decay heat (Q) above its best estimate value (Q_o) and at decreased values of core effective thermal conductivities (k) as compared to the best estimate value (k_o)

APPENDIX C

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
LETTER REPORT

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

October 13, 1988

The Honorable Lando W. Zech, Jr.
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Zech:

SUBJECT: PREAPPLICATION SAFETY EVALUATION REPORT FOR THE MODULAR HIGH TEMPERATURE GAS COOLED REACTOR

Introduction

During the 342nd meeting of the Advisory Committee on Reactor Safeguards, October 6-7, 1988, and in previous meetings of the Committee and our Subcommittee on Advanced Reactor Designs, we reviewed a draft of the subject Safety Evaluation Report (SER). During these meetings, we had the benefit of discussions with representatives of the NRC staff and its consultants, with representatives of the Department of Energy (DOE), and representatives of General Atomics, the chief design contractor for the Modular High Temperature Gas Cooled Reactor (MHTGR). We also had the benefit of the documents referenced.

The MHTGR concept is a product of a joint DOE/industry program to develop a design for a nuclear power plant using HTGR technology and having important inherently safe characteristics. The NRC staff is reviewing the concept under the advanced reactor policy to help assure that the final design will develop along lines acceptable to the NRC.

The draft SER indicates that the staff believes the conceptual design is generally satisfactory and that work directed toward eventual certification should continue. The staff has provided a number of conditions along with this endorsement and also believes that a continuing program of research and development will be necessary to support final design and eventual licensing.

We are in general agreement that design and development should continue along the lines outlined by the NRC staff. We can agree to moving forward, however, only because we understand that an NRC endorsement at this time does not imply a final commitment either to the general design or to its details. We believe that ongoing research and development can resolve important safety issues before licensing. We have a number of comments discussed below about the design.

Key Features of the MHTGR

The MHTGR differs in important ways from existing light water reactor (LWR) plants and from previous gas cooled reactor plants, including several new safety characteristics. The goal of the designers is that the improved safety features will more than make up for the absence of others (e.g., containment). They believe the MHTGR design will provide a plant that is safer than LWRs.

Safety of the MHTGR is keyed to properties of its unique fuel particles. Millions of these microspheres of enriched uranium oxycarbide, each the size of a grain of sand, are in the reactor core. Each fuel particle is coated with four successive protective shells that includes a buffer layer of a porous carbon and then bonded with others into a fuel rod which is, in turn, sealed in vertical holes in graphite blocks. These graphite blocks provide neutron moderation and are the chief structural material in the core.

The maximum fuel particle temperature in normal operation will be about 1150°C. An expected very small fraction of defective particles will cause a measurable, but acceptably low, level of chronic fission-product activity in the coolant and reactor systems.

So long as the particles are maintained below 1600°C, fuel, transuranics, and fission products will be retained by the particle coatings, with very high efficiency. At temperatures above about 2000°C, failures of particle coating will become significant, and above about 2300°C the coatings will fail completely. All other safety features of the reactor systems are designed to assure that particles will remain below 1600°C over a wide range of challenges and circumstances.

It is expected that temperatures can be maintained below 1600°C, in any conceivable reactor transient, because of two favorable characteristics of the reactor core: (1) Strong negative reactivity changes with increased temperatures in fuel or moderator and (2) Large thermal inertia of the core and fuel structure.

It is also expected that temperatures will be maintained below 1600°C even with loss of normal decay heat removal because of the following important features:

(1) The same strong temperature-reactivity effects will assure a very low equilibrium power even with failure of reactivity control and shutdown systems.

(2) At these low or decay power levels, if normal heat transfer systems fail, all heat can be removed from the reactor by a passive heat transfer system that permits atmospheric air to flow by natural

convection through a cavity surrounding the reactor vessel. Under these conditions, the reactor core and the vessel will attain temperatures only slightly above their normal operating values.

(3) If this passive heat removal system should become unavailable (e.g., by blockage of air flow), heat at low power or at decay heat levels would be transferred from the reactor cavity by conduction directly to the earth surrounding the reactor building. Under these conditions, fuel would remain below 1600°C, but the reactor vessel would eventually heat to well beyond its normal operating temperature. Whether the reactor could be returned to normal operation after exposure of the vessel to such overtemperature is problematic at the present time. But, the vessel would remain sufficiently intact for the safe removal of decay heat.

The passive heat transfer functions in items (2) and (3) above require that the reactor core and vessel be small enough so that heat transfer can be accomplished without core temperatures becoming excessive. This dictates the reactor size and leads to the modular design and the long, small-diameter core.

The reactor core is normally cooled by inert helium gas circulated through the core at high pressure. Certain improbable failures of the reactor vessel could permit air to enter the core. However, air flow through the core by natural convection would be at a very low rate. With this restricted supply of oxygen, oxidation of graphite would be so slow that after many hours only a small fraction of the graphite would be consumed and the core would remain structurally intact. Even if the graphite should burn, through some undetermined mechanism, the indications are that the graphite temperature would be well below the 1600°C critical temperature for the fuel particles. The combination of nuclear decay and combustion heat would not be expected to increase core temperature to greater than 1600°C.

The Safety Issues

The challenge in assuring that the key safety characteristics claimed for the MHTGR design are realized in an actual plant is, in simplest terms, in assuring that the following issues are adequately addressed:

(1) Fuel particles must have the retention capabilities attributed to them and this must be assured with recognition of inevitable variability and imperfection in the fuel particles and their compaction process. This will require a higher level of quality in manufacture than has been achieved and must be experimentally verified.

(2) The reactivity and temperature-reactivity characteristics used in safety analyses are based on limited data. Further verification of these characteristics as a function of fuel burnup, core shuffling, and a variety of operational transients is needed.

(3) Inadvertent ingress of water or steam into the core must be precluded with high reliability. Water or steam could cause corrosion and mechanical damage to the graphite and would also add a positive reactivity contribution. This seems to be a possible complication of, for example, steam generator tube failures that is not present in LWRs. Internal flooding of the underground reactor cavity could lead to similar problems.

(4) There must be assurance that decay and low-power heat transfer can be accomplished without causing excessively high core temperatures. Performance of the passive atmospheric cooling system and the ability to conduct heat to the surrounding earth must be demonstrated.

(5) The structural properties of the graphite must be demonstrated and assured.

(6) Some of the important safety benefits of the design (e.g., passive decay heat removal and resistance to graphite burning) depend upon the core geometry remaining unperturbed. Questions of seismic resistance, effects of aging, and the possible cascading effects of certain reactor accidents remain to be fully answered.

A major issue is whether a conventional containment structure or some other mitigation system or process should be required. Neither the designers, the NRC staff, nor the members of the ACRS have been able to postulate accident scenarios of reasonable credibility, for which an additional physical barrier to release of fission products is required in order to provide adequate protection to the public. This does not mean that a conventional containment should not be provided or required as further defense in depth against unforeseen and unforeseeable events. However, it does mean that the design basis for a containment would have to be arbitrary, not altogether unlike what was done in the early days for LWRs. We believe that the decision to require a containment will have to be made on the basis of technical judgment, with appropriate consideration of the effects on other technically based safety features now a part of the design. In addition, there may be safety and economic tradeoffs between provision for containment and provision for passive decay heat removal.

Recommendations

A substantial program of research and development must be continued to support the final design for the MHTGR. This program should concentrate on providing assurances relative to the safety issues we have discussed above.

General Atomics has generated extensive data on fuel performance, but a comprehensive program on the reference fuel appears to be needed. This would include testing of irradiated fuel, fuel from large-scale manufacturing, and fuel exposed to a variety of environmental conditions and temperatures such as might be encountered in possible accidents.

A hot critical experiment may be necessary. The core is of an unusual geometry and has nuclear characteristics different from those in previous HTGRs. Assuring that the safety response of the plant is as predicted will require comprehensive information on the reactivity characteristics of the core over a broad range of normal and accident conditions.

More extensive analysis is needed of the response of the plant to accidents that might change the core geometry. Certain accident scenarios can be hypothesized that would affect core geometry and influence coolant distribution and reactivity characteristics.

A prototype should be built and appropriately tested before design certification.

Concepts for a containment or another sort of physical mitigation system require further study.

Finally, there are two issues identified in our letter to you dated July 20, 1988, "Report on Key Licensing Issues Associated With DOE Sponsored Reactor Designs," that we believe should be given early consideration as the design of this plant progresses. These issues are related to design for (1) resistance to sabotage and (2) operation and staffing. The appropriate excerpts from that letter are attached.

Additional comments by ACRS Members Forrest J. Remick and Charles J. Wylie, and William Kerr are presented below.

Sincerely,

WKerr

William Kerr
Chairman

Additional Comments by ACRS Members Forrest J. Remick and Charles J. Wylie

In general, we agree with our colleagues in the above letter. However, we cannot in good conscience recommend a design of a nuclear power plant for design certification which does not have a conventional containment or other mitigation system which would serve as a more robust external barrier than is currently proposed to protect the public from radiological releases.

The designers of the MHTGR deserve much credit for their effort to incorporate inherent and passive safety features in the design concept. However, even though we believe that the proposed design has a good potential for providing enhanced safety, experience has shown that new reactor designs have technical unknowns. Because of the possible technical unknowns, the known uncertainties associated with the postulated inherent and passive safety features and the lack of experience with operation of a reactor of this new design, we do not recommend these reactors for design certification without a more extensive external barrier consisting either of a conventional containment structure or other appropriate mitigation system.

We think it important that the ACRS and the Commission make this technical judgment at this time in order that the designers of this promising reactor concept have ample opportunity to thoroughly consider alternate designs.

Additional Comments by ACRS Member William Kerr

I remind the Commission of the comments on containment included in the Committee's letter of July 20, 1988, namely:

> "We are not prepared at the present time to accept these approaches to defense in depth as being completely adequate. Further, we are not prepared at this time to accept the arguments that increased prevention of core melt or increased retention capacity of the fuel provide adequate defense in depth to justify the elimination of the need for conventional containment structures. This is not to say that we could not decide otherwise in the future, in response to an unusually persuasive argument."

That is still my position on the containment issue. I would add only that I have not yet heard the "persuasive argument."

References:

1. Office of Nuclear Regulatory Research, "Pre-Application Safety Evaluation Report for the Modular High Temperature Gas Cooled Reactor," dated August 1988 (Predecisional Draft)
2. Stone & Webster Engineering Corporation (DOE Contract), HTGR-86-024, "HTGR Preliminary Safety Information Document for the Standard MHTGR," Volumes 1-5, 1986
3. GA Technologies, Inc. (DOE Contract), DOE-HTGR-86-011, "HTGR Probabilistic Risk Assessment for the Standard Modular High Temperature Gas-Cooled Reactor," Volumes 1-2, January 1987

Attachment:
Excerpts from July 20, 1988 ACRS Letter, "Report on Key Licensing Issues Associated With DOE Sponsored Reactor Designs"

ATTACHMENT TO ACRS LETTER ON MODULAR HIGH TEMPERATURE
GAS COOLED REACTOR

Excerpt from July 20, 1988 ACRS Letter, "Report on Key Licensing Issues Associated With DOE Sponsored Reactor Designs"

Design for resistance to sabotage

It is often stated that significant protection against sabotage can be inexpensively incorporated into a plant if it is done early in the design process. Unfortunately, this has not been done consistently because the NRC has developed no guidance or requirements specific for plant design features, and there seems to have been no systematic attempt by the industry to fill the resulting vacuum. We believe the NRC can and should develop some guidance for designers of advanced reactors. It is probably unwise and counterproductive to specify highly detailed requirements, as those for present physical security systems, but an attempt should be made to develop some general guidance.

Operation and staffing

Little is said in the staff paper about requirements for operation and staffing of advanced reactors. We find this to be a serious oversight. Experience with LWRs has shown that issues of operation and staffing are probably more important in protecting public health and safety than are issues of design and construction. The designers of the three reactor proposals seem to be claiming that the designs are so inherently stable and error-resistant that the questions of operation and staffing, so important for LWRs, are unimportant for the advanced reactors. And that, in fact, the advanced plants can be operated with only a very small staff. We believe these claims are unproven and that more evidence is required before they can be accepted.

The two major accidents that have been experienced in nuclear power, those at TMI-2 and Chernobyl 4, were caused, in large measure, by human error. These were not simple "operator errors" but instead were caused by deliberate, but wrong, actions. There are some indications that the advanced reactor designs being considered have certain characteristics tending to make them less vulnerable to such maloperation. But, this has not been demonstrated in any systematic way. The traditional methods of PRA are not capable of such analyses; but, we believe a systematic evaluation should be made. There seems little merit in making claims for the improved safety of new reactor designs if they have not been evaluated against the actual causes of the most important reactor accidents in our experience.

APPENDIX D

EXPECTATION OF SAFETY ENHANCEMENT

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EXPECTATION OF SAFETY ENHANCEMENT

The Advanced Reactor Policy Statement (51 FR 24643) states:

> ...the Commission expects, as a minimum, at least the same degree of protection of the public and the environment that is required for current-generation LWRs. Furthermore, the Commission expects that advanced reactors will provide enhanced margins of safety and/or utilize simplified, inherent, passive, or other innovative means to accomplish their safety functions.

The staff interprets this statement in the sense that it is the staff's responsibility to ensure that the MHTGR concept has the potential to provide the public and the environment with the same level of protection as that of the current-generation light-water reactors (LWRs), either by satisfying itself as to the safety adequacy of the design features and the overall plant as proposed or by requiring selected changes, additions, or quality upgrades with respect to inadequate or needed equipment.

Although the role of both DOE and the staff is thus clear with respect to safety adequacy, the policy statement provides no direct guidance on how to achieve and demonstrate enhanced safety. However, the staff has considered the following four topics as indicators of the existence of enhanced safety: (1) DOE's response to staff questions in this regard, (2) the approach and degree of conformance with the nine "attributes" for advanced reactors given in the policy statement, (3) the conclusiveness of the research program with respect to the analysis of severe accidents, and (4) the Probabilistic Risk Assessment (PRA) and its staff review. Each of these indicators is discussed separately below followed by the staff's conclusions.

DOE Response to Staff Questions

In Comment 15-11, the staff requested that DOE describe the MHTGR enhanced safety with respect to (1) its design features, (2) potential improvements including consideration of a conventional containment building, and (3) demonstration of its safety characteristics by prototype-plant testing. The staff requested that the description be made by comparison to current-generation LWRs, such as the advanced boiling-water reactor (ABWR) of General Electric.

DOE responded by summarizing its previous descriptions of MHTGR safety but declined to make a comparison to the ABWR on the grounds that suitable documentation was not available. DOE provided a discussion of enhanced safety characteristics considering the MHTGR's predictable slow thermal transients; its insensitivity to operator errors; the fuel's capability to retain fission products under extremely adverse postulated events; the highly reliable, passive, and inherent safety features for decay-heat removal and reactor shutdown; and the simplification of the design, particularly through the use of the single phase, nonchemically reacting helium coolant. With regard to potential improvements including a conventional containment building, DOE stated: "Because of its enhanced safety characteristics, the MHTGR has such a high level of safety that no further meaningful improvement in public risk can be obtained at reasonable cost." Finally, DOE reiterated its position that the research and development needs are so modest that startup plant testing is needed only to confirm the integrated system performance and that no demonstration plant will be necessary for special or licensing tests. The staff has found this later point unacceptable as described in Chapter 14.

Conformance With Advanced-Reactor Attributes

In Table D.1, each attribute is listed together with a statement identifying the staff's view of the MHTGR's approach and conformance. In a broad sense the MHTGR conforms well by means of (1) the existing design, (2) the staff's required design selections, (3) expected favorable resolutions of certain safety issues at a later design stage, or (4) anticipated favorable results of the research, development, or testing programs. Although the policy statement does not require conformance with any single attribute, the degree of overall conformance is accepted tentatively by the staff as indicating the degree of enhanced safety.

Research Program Conclusiveness Pertaining to Severe-Accident Analysis

The staff believes that research programs pertaining to fuel integrity and fission-product transport are likely to be straightforward and will become well defined in the Reactor Technology Development Plan. Accordingly, they will lead to conclusive results pertaining to severe accidents if successful. Furthermore, although these programs will require considerable effort, the staff believes they can be termed "modest," in comparison to severe-accident research for LWRs, while being capable of achieving a greater degree of conclusiveness. Such conclusiveness in itself does not ensure enhanced safety, but rather demonstrates that the MHTGR concept is capable of a mechanistic analysis for severe accidents, clearly a safety enhancement over the need to postulate a nonmechanistic siting source term for LWRs to account for uncertainties in phenomenological representations.

PRA and Staff Review

The PRA performed by DOE was encouraging regarding safety enhancement, since no sequences down to frequencies of 10^{-7} and below per plant-year could be identified that would lead to a large radionuclide release. However, the staff was concerned about the validity of this conclusion because of potential vulnerabilities from very large seismic events, lack of an empirical data base for many important components, and the overall uniqueness of the design that could be caused by failure mechanisms now hidden and that might not become apparent even at later stages of review. Furthermore, since the essential safety features are mainly structures of passive function, PRA is not as revealing of their failure modes in the same sense as research, engineering analysis, and test data. Consequently, although the PRA results are encouraging with respect to safety enhancement, the PRA results at this stage of MHTGR development can only be accepted as a favorable indicator but not a conclusive finding for the existence of safety enhancement.

Conclusions

Taken as a whole and individually, the four topics chosen as indicators of safety enhancement support the preliminary conclusion that the MHTGR can provide, qualitatively, a greater level of protection to the public than current-generation LWRs. Since much of this support depends on development of information that will be reviewed at a later design stage, the staff can only conclude at this time that the MHTGR has the potential to provide an enhanced level of safety, and thus has the potential to meet the objective of the Advanced Reactor Policy Statement with regard to the Commission's expectation of enhanced safety for advanced HTGRs.

The staff recommends, as the design advances, that DOE consider developing a means to quantify the MHTGR's safety enhancements and provide at the preliminary standard safety analysis report stage a demonstration of how these quantitative enhancements are in fact achieved. Otherwise, the MHTGR safety features could be viewed simply as alternative means of providing an adequate safety level rather than an enhanced safety level. Although the present qualitative approach is believed to be adequate at this review stage to satisfy the policy statement's objectives, quantitative information would clearly demonstrate that the Commission's expectation of enhanced safety has been met by the MHTGR.

Table D.1 Approach and conformance with advanced-reactor attributes

Attributes from Advanced Reactor Policy Statement	MHTGR approach and conformance
(1) Highly reliable and less complex shutdown and decay heat removal systems. The use of inherent or passive means to meet this objective is encouraged (negative temperature coefficient, natural circulation).	Insertion of movable reactivity poisons is not needed to achieve hot reactor shutdown* due to a large reactivity feedback from increased core temperature. Decay heat can be removed by a passive system that depends on heat transmission from the reactor-vessel surface to a passive heat sink in the reactor cavity.
(2) Longer time constants and sufficient instrumentation to allow for more diagnosis and management before reaching safety systems challenge and/or exposure of vital equipment to adverse conditions.	The MHTGR has the same slow response to core-heatup transients characteristic of other HTGRs because of its low power density and high heat capacity and the high-temperature capability of the fuel. It is the staff's position that additional and/or improved quality of the instrumentation proposed is needed. These requirements are identified in Table 1.5.
(3) Simplified safety systems that, where possible, reduce required operator actions, equipment subjected to severe environmental conditions, and components needed for maintaining safe-shutdown conditions. Such simplified systems should facilitate operator comprehension, reliable systems function, and more straightforward engineering analysis.	The staff's safety analysis confirms that no operator actions should be needed to shut down the reactor and to remove decay heat. However, the staff requires that licensed and trained operators have the capability to manually trip the reactor from an accessible and habitable location and be otherwise available to perform mitigative and recovery actions, if necessary. Transients may expose the reactor vessel to higher temperatures than currently allowed by the ASME Code, a concern expected to be resolved at a later review stage. The staff's review to date has indicated that engineering analyses pertaining to severe-accident phenomena are simplified and more straightforward than for light-water reactors (LWRs).

*See Section 4.3.1 for discussion of hot reactor shutdown limitations.

Table D.1 (Continued)

Attributes from Advanced Reactor Policy Statement	MHTGR approach and conformance
(4) Designs that minimize the potential for severe accidents and their consequences by providing sufficient inherent safety, reliability, redundancy, diversity, and independence in safety systems.	The integrity of the fuel and the passive and inherent design features that are proposed to maintain fuel integrity under severe-accident-type service conditions are the keys to minimizing the potential for and consequence of severe accidents in the MHTGR. On the basis that successful completion of research, development, and testing programs needed to ensure fuel integrity can be achieved, the staff concludes that safety-grade equipment that provides the full redundancy, diversity, and independence customary for LWR decay heat removal systems will not be needed for the MHTGR.
(5) Designs that provide reliable equipment in the balance of plant (or safety-system independence from balance of plant) to reduce the number of challenges to safety systems.	A stated MHTGR design objective is to provide safety-system independence from the balance of plant. The staff will confirm that this objective has been met or an acceptable alternative has been provided at a later review stage.
(6) Designs that provide easily maintainable equipment and components.	The components of the helium transport systems, that is, the main circulator, steam generator, and main loop shutoff valve, are located in a steel vessel separate from the reactor vessel, which should provide ease of maintenance for these components. Also, experience with Fort St. Vrain provides guidance with respect to fuel handling, maintenance of reactor systems, and maintenance of many other components. However, the MHTGR contains features and equipment unique to its design (e.g., the hot duct within the crossduct vessel) for which the staff has deferred considerations of inspectability and maintainability to a later review stage.

Table D.1 (Continued)

Attributes from Advanced Reactor Policy Statement	MHTGR approach and conformance
(7) Designs that reduce potential radiation exposures to plant personnel.	DOE's estimate for total occupational exposure for the MHTGR is between 4 and 7 times lower than for LWRs. This estimate is consistent with the operational experiences of Fort St. Vrain and Peach Bottom 1. Although the MHTGR differs in system design and plant layout from these earlier HTGRs, the staff believes low MHTGR occupational exposure is achievable because of the existing and proven technology available.
(8) Designs that incorporate defense-in-depth philosophy by maintaining multiple barriers against radiation release and by reducing the potential for the consequences of severe accidents.	As originally proposed, MHTGR fuel integrity was designated as the only safety-grade barrier against fission-product release. The staff has required that all portions of the primary coolant pressure boundary meet safety-grade classification standards in order to incorporate additional defense-in-depth. The staff has also found that the proposed frequency of elevated-temperature challenges to the reactor vessel is unacceptable with respect to defense-in-depth and requires resolution at a later review stage. Based on future demonstration of acceptable fuel integrity by research and testing and an acceptable means to reduce the frequency of elevated-temperature challenges to the reactor vessel, the staff continues to review the DOE proposal that an additional barrier against radionuclide release that would be provided by a conventional, leak-tight containment building would not be necessary to mitigate the consequences of severe accidents.

Table D.1 (Continued)

Attributes from Advanced Reactor Policy Statement	MHTGR approach and conformance
(9) Design features that can be proven by citation of existing technology or that can be satisfactorily established by commitment to a suitable technology development program.	Fort St. Vrain, Peach Bottom 1, and HTGR operating experience in the Federal Republic of Germany provide an existing technology base that supports the successful development of MHTGR safety features. DOE has committed to a technology development program that is expected to be suitable when modified in accordance with staff positions.

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13. ABSTRACT (200 words or less)

This draft safety evaluation report (SER) presents the preliminary results of a preapplication design review for the standard modular high-temperature gas-cooled reactor (MHTGR) (Project 672). The MHTGR conceptual design was submitted by the U.S. Department of Energy (DOE) in accordance with the U.S. Nuclear Regulatory Commission (NRC) "Statement of Policy for the Regulation of Advanced Nuclear Power Plants" (51 FR 24643), which provides for early Commission review and interaction. The standard MHTGR consists of four identical reactor modules, each with a thermal output of 350 MWt, coupled with two steam turbine-generator sets to produce a total plant electrical output of 540 MWe. The reactors are helium cooled and graphite moderated and utilize ceramically coated particle-type nuclear fuel. The design includes passive reactor-shutdown and decay-heat-removal features. The staff and its contractors at the Oak Ridge National Laboratory and the Brookhaven National Laboratory have reviewed this design with emphasis on those unique provisions in the design that accomplish the key safety functions of reactor shutdown, decay-heat removal, and containment of radioactive material. Final guidance on the acceptability of the MHTGR standard design is contingent on receipt and evaluation of additional information requested from DOE pertaining to the adequacy of the containment design.

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