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450 MW(T) MHTGR SOURCE TERM AND CONTAINMENT REPORT

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450 MW(T) MHTGR

SOURCE TERM AND CONTAINMENT REPORT

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ACRONYMS

ABB-CE	ABB-Combustion Engineering
AOO	Anticipated Operational Occurrence
ASME	American Society of Mechanical Engineers
AVR	Arbeitsgemeinshaft Versuch Reaktor
B&PV Code	Boiler and Pressure Vessel Code
BNI	Bechtel National, Inc.
DBE	Design Basis Event
DCC	Depressurized Conduction Cooldown
DF	Decontamination Factor
DOE	Department of Energy
EAB	Exclusion Area Boundary
ECA	Energy Conversion Area
EPBE	Emergency Planning Basis Event
EPZ	Emergency Planning Zone
FRG	Federal Republic of Germany
FSAR	Final Safety Analysis Report
FSV	Fort St. Vrain
GA	General Atomics
GCRA	Gas Cooled Reactor Associates
HPS	Helium Purification System
HTGR	High Temperature Gas-Cooled Reactor
HTS	Heat Transport System
HVAC	Heating, Ventilating, and Air Conditioning
LPVC	Low Pressure Vented Containment
LWR	Light Water Reactor

MHTGR	Modular High Temperature Gas-Cooled Reactor
NI	Nuclear Island
NRC	Nuclear Regulatory Commission
OPDS	Overall Plant Design Specification
OPyC	Outer Pyrolytic Carbon
ORNL	Oak Ridge National Laboratory
PAG	Protective Action Guidelines
PCC	Pressurized Conduction Cooldown
PSAR	Preliminary Safety Analysis Report
PSER	Pre-application Safety Evaluation Report
PSID	Preliminary Safety Information Document
PyC	Pyrolytic Carbon
RCCS	Reactor Cavity Cooling System
RCPB	Reactor Coolant Pressure Boundary
SCS	Shutdown Cooling System
SiC	Silicon Carbide
SRDC	Safety Related Design Condition
SWEC	Stone and Webster Engineering Corporation
THTR	Thorium Hochtemperatur Reaktor
VLPC	Vented Low Pressure Containment

MHTGR SOURCE TERM AND CONTAINMENT STUDY

EXECUTIVE SUMMARY

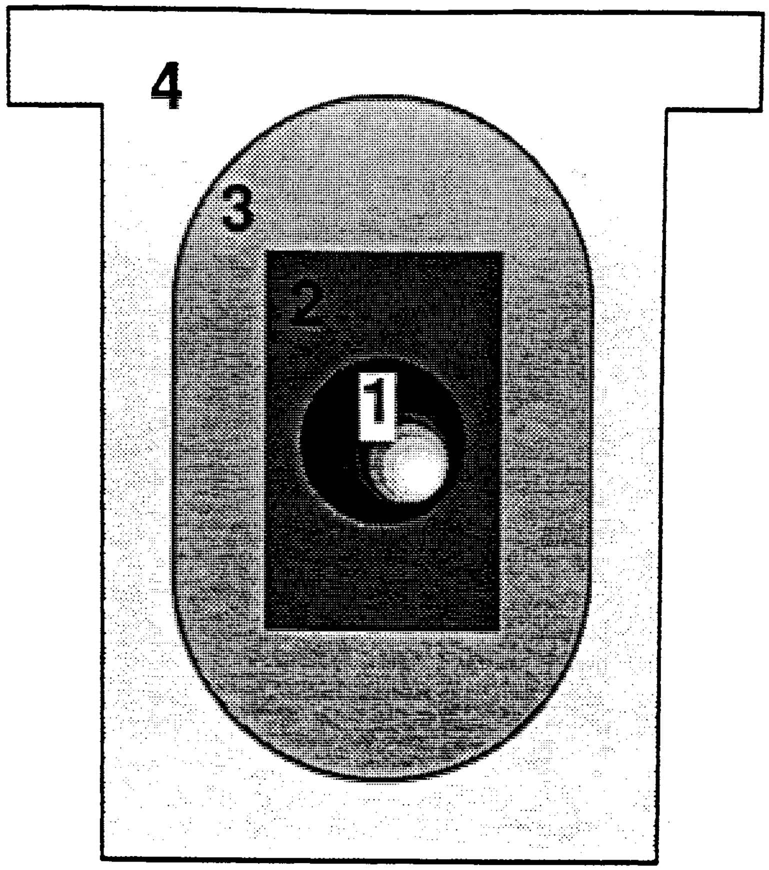
PURPOSE

The purpose of this report is to evaluate the MHTGR response to alternative source term assumptions for a range of vented low pressure containment designs. Specifically, the program has evaluated alternative passive containment features assuming 1) the possibility of higher fuel defect fraction, 2) very rapid hydrolysis of defective fuel, 3) the possibility that the fuel could suffer an undetected weakness, and 4) a higher than expected release of plateout activity. This report directly responds to an NRC letter requesting additional studies on these topics (Morris, B. M., Letter to Dr. Sol Rosen, May 9, 1990).

INTRODUCTION

The MHTGR design emphasizes retention of fission products at the source, within the ceramic coated fuel particles. The core is configured to preclude temperature transients which could cause failure of the ceramic coated fuel. Figure E-1 illustrates conceptually the multiple barriers provided by the MHTGR against the release of radiation. In contrast to light water reactor technology, the most important barrier is that provided by the fuel particle and coating, barrier 1. Barriers 2 and 3, i.e., the graphite core and the reactor coolant pressure boundary, are designed to complement and supplement the fuel barrier. The fourth barrier, the vented low pressure containment, is designed as an independent barrier to aid in meeting requirements, providing additional margins, and providing defense-in-depth.

The overall MHTGR source term is a function of fuel quality from the manufacturer, fuel performance during operation, and the extent of fuel heatup during a loss of forced cooling. There are generally two distinct components to the MHTGR accident source term: **a prompt term** which can be released immediately and a **delayed term** whose timing is dictated by the slow heatup of the core. The MHTGR **prompt source term** is comprised of both circulating and plateout inventories. Circulating inventories are characterized as fission products that are released from the fuel and core during normal operation entrained in the flowing helium coolant and could be released from the primary system in the event helium coolant escapes. The plateout inventories are fission products released from the fuel and core during normal operation that are located on the inside surfaces of the reactor coolant pressure boundary (RCPB). A break in the RCPB large enough to allow a rapid blowdown of helium could result in the re-entrainment of a fraction of these fission products and their release from the RCPB. Plateout activity could also be washed off and made available for release from the



Barrier 1: Fuel Kernel and Particle Coatings - The kernel of a failed fuel particle will retain more than 95% of the radiologically important short-lived fission gases such as Kr-88 and I-131. However, the effectiveness of a kernel for retaining gases can be reduced if the exposed kernel is hydrolyzed by reaction with water vapor. Barrier 1 relies primarily on the silicon carbide layer, which is impervious to most radionuclides over the full range of normal and accident temperatures. For a very small portion of the approximately ten billion coated fuel particles in a MHTGR core, this barrier will not be fully intact following fabrication. The silicon carbide layer is basic to the fuel performance model and the source term calculation.

Barrier 2: Fuel Rod Matrix and Fuel Element Graphite - The graphite binder, which is used when coated particles are formed into fuel compacts, is highly sorptive of metallic fission products but provides little holdup of fission product gases. The graphite of the fuel element blocks into which the fuel compacts are inserted is less sorptive of fission metals but is more effective as a barrier to diffusion.

Barrier 3: Reactor Coolant Pressure Boundary - The reactor vessel, steam generator vessel, cross vessel and their connections and appurtenances act to retain the helium reactor coolant and any entrained radioactive material. When the release of coolant is slow the vessels provide a volume which holds up fission products released from the core during the latter stages of the event.

Barrier 4: Vented, Low Pressure Containment - The containment is a normally closed space, located below grade, which will act to retain fission products. It is equipped with a vent which will open if the pressure inside the containment exceeds its design setpoint, releasing the mass and energy associated with a blowdown and protecting the integrity of the building and the Reactor Cavity Cooling System. The reactor building inhibits the release of fission products to the environment by plateout, deposition, and decay even if the vent opens. While the vent allows the release of fission products released promptly, the release of associated gases early in the event eliminates the driving pressure which could transport the delayed source term out of the building.

Figure E-1 MHTGR BARRIERS AGAINST FISSION PRODUCT RELEASE

RCPB by the action of steam or water during an event involving a steam generator tube failure and subsequent lifting of the primary coolant relief valve(s).

The magnitude of the **prompt source term** is a result of power operation which generates circulating and plateout radionuclide levels. The circulating and plateout inventories released from barriers 1 and 2 and available for release in a depressurization event will be limited by Technical Specifications imposed on reactor operations. The amount of radionuclides actually released from the RCPB will depend on the inventories and liftoff/washoff parameters such as shear force, species adherence characteristics and amount of water ingress.

The **delayed source term** is a larger fraction of the total fission products and consists of the release from fuel particles which have manufacturing defects, including heavy metal contamination. These fission products are released from the fuel and core (barriers 1 and 2) into the coolant by temperature transients or chemical attack from water ingress events. The differences between the MHTGR and other current reactor concepts are based in part on the use of a solid (graphite) moderator in a low power density core. In the event of an accident, the graphite provides a heat sink and heat rejection path for residual decay energy. As a result, the fuel temperature transients are limited and occur slowly (over several days) resulting in the delayed timing of this second component of the MHTGR source term. Similarly, if a water ingress event occurs, the reaction between water and the small fraction of defective fuel proceeds at a rate that requires hours to become significant. The delayed source term released from the fuel and core by temperature excursions or by water-fuel reactions, comprise the majority of the fission products associated with the most severe MHTGR events. It contributes to the offsite radiation doses only when transported out of the RCPB, barrier 3, by some mechanism such as leaking reactor coolant or the open failure of a relief valve, hours to days after the start of the event. The delayed source term is also controlled by Technical Specifications on the as-manufactured fuel quality, core materials and geometry. The amount of radionuclides actually released from the RCPB will depend on the inventories and accident parameters such as temperature and oxidants.

The MHTGR is located in a reinforced concrete building which is embedded so that the operating floor, above the vessel system, is at grade. The building (Barrier 4) incorporates a pressure relief vent which is designed to open at 1 psid and allow the escape of gases blowing down from the primary or secondary coolant boundaries during a boundary failure. The vent is designed to reclose following a transient. With a leak rate specified at 1 volume per day, this vented, low pressure containment (VLPC) building is a redundant and independent fission product retention barrier, providing defense in depth for low probability accidents.

During the development of the design, a Preliminary Safety Information Document (PSID, Ref. 1) was written for the 350 MW(t) reference MHTGR design. The PSID and an accompanying Probabilistic Risk

Assessment (PRA, Ref. 2) were written in the context that the fuel would meet the quality and performance specified, and the NRC was requested to provide a conditional judgement on the design, assuming a successful outcome of the fuel and fission product technology program. The NRC reviewed those documents and provided DOE with comments in a draft Pre-application Safety Evaluation Report (PSER, Ref. 3). Following issuance of the draft PSER, the program prepared a trade study which evaluated the costs and benefits of additional reactor building features and alternative containment strategies. That Containment Report (Ref. 4) evaluated the performance of alternative features over a statistically wide range of parameters, including weather models and the variation in the fuel quality within the specification confidence intervals. The NRC requested that additional studies be undertaken which addressed lower levels of fuel integrity and reliability parameters (Ref. 5, included as an Appendix).

In this document the information presented in the prior assessments is refined and the concerns raised regarding the accident source terms are addressed. This report is based on the design and features of the 450 MW(t) MHTGR, and presents calculated offsite doses for that design; however the issues, criteria, phenomena, and conclusions would apply equally well to the 350 MW(t) design.

SAFETY REQUIREMENTS AND LICENSING BASIS EVENTS

The MHTGR is designed to achieve higher levels of radiological safety and public acceptance, consistent with the advanced reactor policy statement (Federal Register, Vol. 51, No. 130, page 24643). In addition to all statutory requirements derived from 10CFR100, 10CFR 50 Appendix I, 10CFR20, and 40CFR190, the MHTGR is designed to meet a utility/user requirement to avoid the need to hold offsite public evacuation drills. To meet this objective, site boundary dose goals for individual events were set by the lower threshold EPA Protective Action Guidelines (PAG) (Ref. 6): 5 Rem thyroid, 1 Rem whole body. To justify setting the Emergency Planning Zone boundary at the same distance as the Exclusion Area Boundary, a wide spectrum of events is evaluated down to a frequency of 5×10^{-7} per plant year to show that on a cumulative basis that the PAGs are not exceeded at the plant boundary of 425m.

The following definitions are used in this report:

- Licensing Basis Events (LBE) - Full spectrum of events from anticipated operational occurrences to events beyond the design basis with frequencies as low as 5×10^{-7} per plant year.
- Design Basis Events (DBE) - Events not expected in the plant lifetime but which might occur in the lifetime of a population of plants for which the plant is designed and conservatively assessed against 10CFR100.

- Safety Related Design Conditions (SRDC) - Limiting conditions for the safety related equipment which are derived from the DBEs by assuming that only safety related equipment is available to mitigate the consequences. SRDCs are conservative extensions of the DBEs which are evaluated using conservative parameters in the Safety Analysis Reports against 10CFR100.

A study of the limiting licensing basis events which could result in the release of fission products has been performed for various VLPC concepts considering three classes of events. For purposes of this study three of the SRDCs, namely SRDC-6, SRDC-11 and SRDC-10, have been used to represent the three classes of events important to MHTGR radiological consequences. In this context they have been evaluated realistically on an individual event basis and on a cumulative basis against the PAGs.

The first class of accidents considered are steam generator tube failures with depressurized conduction cooldown (SRDC-6). These events encompass families of accidents that can potentially display rapid combined releases of prompt and delayed source terms. Such events assume a large failure of the steam generator, moisture ingress, opening of the RCPB relief valve and subsequent loss of core cooling. Moisture ingress events without forced cooling can result in RCPB relief valve opening after about a day. These events result in releases of circulating activity, washoff of a portion of the plateout, and hydrolysis of defective fuel with a release of a fraction of the fuel inventory. If the relief valve fails open, additional releases from the RCPB of the fuel inventory fission products occur due to the core heatup. These fission products are transported from the RCPB by thermal expansion of the coolant as the core heats up. SRDC-6 is an extension of DBE-6. It assumes that all non safety-related systems and components are assumed to fail and that only the safety related equipment is available. The expected frequency for SRDC-6 is below 10^{-7} per plant year.

The second class of events considered are small helium leaks with depressurized conduction cooldown (SRDC-11). These events are considered since they provide a means to transport a portion of both the prompt and delayed source term. These events are limited to small RCPB leaks of less than .05 sq. in. combined with subsequent core conduction cooldown events that provide a driving force from the vessel due to the slow depressurization of helium coolant (e.g., over several days). During the depressurization the circulating activity, a portion of the plateout activity, and a fraction of the fuel inventory which might have failed during the conduction cooldown are released. The expected frequency for SRDC-11, which is an extension of DBE-11, is approximately 3×10^{-5} per plant year.

The final class, rapid helium depressurizations, are characterized by openings up to the design basis of 13 sq. in. which result in depressurized core conduction cooldown (SRDC-10). In these events the helium leak size results in a rapid blowdown (eg., blowdown complete within minutes) followed by a relatively slow transport of gases from the vessel due to thermal expansion of the coolant as the core heats up and thermal contraction as the core cools down. This class of events typically has the most rapid release of the prompt source term. A much smaller portion of the delayed source term as compared to small leaks is released because the helium coolant, which transports the fission products out of the RCPB, is essentially all released before the majority of the delayed source term is released from the core. SRDC-10 is an extension of DBE-10, and includes the failure of both main and shutdown cooling systems, which could remove heat from the core even at a coolant pressure of only one atmosphere. The frequency of SRDC-10 is estimated to be approximately 10^{-5} per plant year.

These events have been selected since past studies have shown that SRDC-6 releases are the largest; SRDC-11 like events are risk dominant and SRDC-10 accident releases are most prompt.

EXPECTED MHTGR PERFORMANCE

Table E-1 shows the reference I-131 source terms for the limiting accidents, which are summarized and described in greater detail in the body of the report. I-131 is the major contributor to the thyroid dose which is the limiting offsite dose for the MHTGR. As shown the as-manufactured fuel quality which is controlled by the Technical Specifications limits the potential fraction of fission products outside the standard fuel particles to less than 0.01%. Furthermore, the potential prompt source terms are smaller than the delayed terms, namely, the circulating and plateout activity inventories are significantly smaller than those in the core.

With the specified fuel quality and anticipated radionuclide retention performance of the fuel and the VLPC, the MHTGR will meet 10CFR100 dose guidelines at a site boundary of 425 m (1400 ft). These doses are for an individual at the plant boundary for an indefinite period. Also as shown in Table E-2, the MHTGR is predicted to meet goals related to the PAGs for individual events and for the cumulative risk PAG goal for the whole body dose, 1 Rem at a frequency of 5×10^{-7} per plant year. However, the cumulative thyroid dose from all events taken at event frequencies of 5×10^{-7} per plant year and higher is approximately 12 to 14 Rem compared to the 5 Rem requirement. The cumulative risk is dominated by events like SRDC-11, but includes events represented by SRDC-6 and SRDC-10. This expected dose results from an unfavorable and very unlikely combination of uncertainties that are well beyond 95 percent confidence. Additional detail in modeling and analysis, improved understanding through the fuel and fission product technology development program, or addition fission product retention design strategies are required in order to meet the cumulative thyroid PAG risk goal. It may also be appropriate to review the basis for the frequency at which this cumulative goal should be met.

Table E-1
I-131 INVENTORY RELEASED BY LIMITING EVENTS - 450 MW(t) MHTGR
(Best Estimate Fuel Temperatures)

Location	Inventory (Ci)	Timing of Release	Release Events					
			Water Ingress with Loss of Forced Cooling (SRDC-6)		Slow Depressurization and Loss of Forced Cooling (SRDC-11)		Rapid Depressurization and Loss of Forced Cooling (SRDC-10)	
			Release Mech.	I-131 Rel. (Ci)	Release Mech.	I-131 Rel. (Ci)	Release Mech.	I-131 Rel. (Ci)
Potential Prompt Source Term								
- Circulating	0.03	Minutes	Helium Blowdown	0.03	Helium Blowdown	0.03	Helium Blowdown	0.03
- Plateout	26	Minutes	Washoff	15.4	Liftoff	0.04	Liftoff	0.05
Potential Delayed Source Term								
Defective Fuel								
Missing Coatings	600	min-hr	Hydrolysis	219				
Heavy Metal Contamination	120	hr-days	Heatup	72	Heatup	87	Heatup	87
Standard Particles	1.2×10^7			0		0		0
	Total from the Core			291		87		87
	Total from the RCPB			266		48		12

Table E-2
COMPARISON OF SITE BOUNDARY DOSES WITH UTILITY/USER REQUIREMENT TO MEET PAGs
(Best Estimate, Reference Quality Fuel and Behavior Models)

		SRDC-6	SRDC-11	SRDC-10	CUMULATIVE EVENTS > 5×10^{-7} per plant year
Reference Assessment	Thy[1] Rem	1.51	.142	1.48×10^{-2}	~12 - 14
PAG Requirement	Thy Rem	5	5	5	5
Reference Assessment	WB[2] Rem	1.08×10^{-2}	4.91×10^{-4}	2.81×10^{-4}	less than 1
PAG Requirement	WB Rem	1	1	1	1

Note: Shaded blocks represent cases that meet the EPA PAG dose limits of 5 Rem Thyroid and 1 Rem Whole Body.
(1) Thy = Thyroid dose
(2) WB = Whole body dose

ALTERNATIVE ACCIDENT SOURCE TERMS

In addition to the performance of the plant with the as-specified fuel quality and behavior, calculations have been made to determine the performance of the plant under fuel quality and behavior assumptions based on the concerns raised in the NRC review of the design and of the Containment Report. The assessment of MHTGR accidents including alternative, hypothetical source terms are treated in a manner similar to hypothetical events previously postulated by the NRC staff during review of the PSID. Those events were assessed on a best estimate basis and discussed in order to assure low residual risk (i.e., no significant increase in risk due to the inclusion of lower probability events).

Four alternative fuel quality and behavior models as described below have been assessed. These alternate source term events are assessed in conjunction with alternate reactor building options to explore the capability to mitigate releases from the limiting hypothetical events.

1. Lower Limit Fuel Quality

Accident specific source terms assuming a fuel defect fraction which is approximately a factor of ten higher than specified fuel have been developed and evaluated in detail. This defect fraction is similar to the fuel quality manufactured for the Fort St. Vrain HTGR.

2. Rapid hydrolysis

A bounding, conservative assumption regarding the hydrolysis of defective fuel following a water ingress event has been developed and the dose consequences which would result have been estimated. Defective fuel particles designated as bare kernels are susceptible to kernel hydrolysis. This reaction, which enhances fission product release, is assumed to occur instantaneously, resulting in source terms which include the hydrolysis of all exposed fuel.

3. Weak fuel

A set of arbitrary assumptions relating to the concern that weak fuel has been used. Weak fuel is defined as fuel which performs as predicted under normal operating conditions, but for some unknown reason fails during core heatup even though extensive fuel capsule tests has not shown this phenomena. Assessments have been made assuming half of a core reload contains weak fuel at the time an SRDC-11 like event occurs. Weak fuel is postulated to fail when temperatures exceed 1200°C. Since this temperature is experienced by only a small fraction of the fuel during normal operation, it is conservatively assumed that "weak fuel" would not be detected until several hours into an SRDC-11 event. When portions of the core reach 1200°C, the weak fuel is assumed to behave like exposed kernels, i.e., fuel with all coatings failed. The inventory in this source was conservatively assumed to be released rapidly, similar to heavy metal contamination. The resulting source term is incremental to the source term associated with the defective fuel allowed within the fuel specification. While these assumptions are severe in terms of source term magnitude, it should be noted that the MHTGR uses a completely inert coolant, and that all materials in the active core are ceramic, with melting temperatures far above any temperatures identified in postulated accidents. Therefore, this "weak fuel" scenario cannot lead to energetic or fission product consequences like the severe accident scenarios used in LWR residual risk evaluation.

4. Increased prompt release

A bounding, conservative assumption regarding the re-entrainment of plated-out activity has been developed and evaluated. Plateout is expected to adhere to the inside surfaces of the RCPB, and only contributes to the offsite dose consequences if it is liberated by mechanical forces associated with the blowdown of helium coolant, or the mechanical and chemical forces associated with a water ingress. Much less than 1% is predicted to be released. However, it is assumed for purposes of evaluating the VLPC alternatives that 100% of the plateout activity could be liberated. Two levels of fuel quality are considered.

The release of I-131 from the RCPB for the reference fuel specification and performance and for each of the alternative source term scenarios are summarized for SRDC-6, SRDC-10 and SRDC-11 in Table E-3.

Table E-3
REFERENCE AND ALTERNATIVE SOURCE TERMS
Curies of I-131 Released from the Reactor Coolant Pressure Boundary

Alternative Source Term	SRDC-6	SRDC-11	SRDC-10
0. Reference Fuel Quality	266	48	12
1. Lower Limit Fuel Quality	2863	471	--
2. Rapid Hydrolysis with Lower Limit Fuel Quality	6305	--	--
3. Weak Fuel	--	15,000	--
4. Increased Prompt Release a. with Reference Fuel Quality	--	--	37
b. with Lower Limit Fuel Quality	--	--	370

ALTERNATIVE FISSION PRODUCT RETENTION DESIGN OPTIONS

The following design options have been evaluated as additions or modifications to the VLPC.

1. Filters on the RCPB Relief Line

This modification is the addition of a piping network to collect the discharge from main helium relief valves and to discharge it to the environment via a filter. This subsystem would be designed to operate for all events which result in the opening of helium relief valves, including water ingress events.

2. Filters on the VLPC

This mitigation measure adds a filtered release pathway from the VLPC, independent from the containment vent. This filter would be designed to be effective during radiologically important RCPB leaks which result in pressures below the containment vent opening pressure. This design option would filter gases escaping from RCPB leaks (less than 1 in^2) on the way to the environment.

3. Changes in the Site Boundary Distance and VLPC Release Elevation

Increasing the EAB from 425 m to 805 m, and the addition of a 90 m (300 ft) tall VLPC stack are evaluated. These design options would mitigate the doses for all classes of accidents.

Table E-4 defines the specific VLPC cases considered. Cases B and C add filters of two different efficiencies to the relief line and containment; Case D additionally increases the site boundary; and Case E still further adds stacks. Those VLPC cases are compared to two low leakage high pressure containments previously evaluated in Ref. 4. As defined in Table E-3 Cases F and G vary in the leakage rate.

Table E-4
MITIGATION MEASURES CONSIDERED

Vented, Low Pressure Containment

Case A is the current, reference reactor building. Its vent path is designed to open at 1 psid and it can withstand a 10 psi internal pressure transient load. Fission products which leak into the building are reduced by plateout and deposition before release to the environment via the vent path.

The building has a leak rate of 1 volume per day. Helium relief valves discharge into the building. Building leakage and the vent path discharge are considered ground level releases, and the site boundary distance is 425 m.

Case B adds a system to collect relief discharges and conduct them to a simple filter located outside the building. It also adds a second building vent path which goes to another simple filter. This filtered path will relieve the pressure from small leaks, which are radiologically important, so that the larger building vent will not open.

Case C is the same as Case B, except that it uses more efficient filters at both locations. Increasing the area of the reactor building filter improves the decontamination factor and also affects the percent of release to the environment through building leaks which bypass the filter.

Case D is the same as Case C, except that the site boundary distance is increased from 425 m (1400 ft) to 805 m (2640 ft)

Case E modifies Case D by adding a tall stack, so that the releases can be considered as elevated releases. The stack, which must be three times as tall as the tallest building, will be about 90 m (300 ft). The discharge from both filters and the main building vent is routed to the stack. Discharges from building leaks are still considered as ground level releases.

Low-leakage, High Pressure Containment

Case F is a conventional, high pressure containment structure based on the design included in the previous Containment Report. It has a 5 vol%/day leak rate. The internal design pressure would need to be on the order of 55 psig, and alternative possibly active decay heat removal and containment cooling systems, would be required.

Case G is identical to Case F except that a containment leak rate of 0.5 vol%/day has been evaluated at a site boundary distance of 805 m.

RESULTS

The calculated radiation doses at the site boundary are shown for various MHTGR source term and containment alternatives in three tables. Table E-5 gives the thyroid and whole body doses resulting from a water ingress event (SRDC-6) for the reference and alternative source terms versus the selected range of alternative containment features. The top half of Table E-5 specifies the containment characteristics. Cost estimates of various features have not been made in detail, but have been drawn from the Containment Report (Ref. 4), or the Cost Reduction Study (Ref. 7) and adjusted. The bottom half of the table summarizes the SRDC-6 offsite doses. As shown by the shading, nominal doses for all containment cases meet the PAG lower threshold doses when the reference fuel quality and behavior are assumed. The margin to the requirement increases with the degree of containment increasing left to right from Case A to Case G. Under assumptions of lower limit fuel quality and rapid hydrolysis with lower limit fuel quality, filters on the relief line are needed to meet the PAGs for the thyroid dose. Because the alternative accident source terms coupled with the SRDC accident frequencies are beyond the licensing basis, the consequences of these events are best compared to values which would lead to prompt fatalities. This is about 600 Rem to the whole body. This table shows how filters can effectively reduce the consequences of the MHTGR events with the largest doses.

Table E-6 gives the thyroid and whole body doses resulting from a small break in the helium pressure boundary (SRDC-11) for the same range of fuel assumptions and features. In this case again the nominal doses for all containment cases meet the PAGs when the reference fuel quality and behavior are assumed.
Similar to the SRDC-6 results, SRDC-11 doses for the alternative source terms require filters to meet the PAGs but all cases are well below the prompt fatality doses. However, the cumulative risk is controlled by SRDC-11 type events and as shown by the shading this requirement is not met for the thyroid dose at 5×10^{-7} per plant year for the reference VLPC, Case A. Furthermore, as discussed in Ref. 4 the cumulative risk includes events in which the containment operates as expected and also events in which it fails. This is important for those containment options that require isolation and that are susceptible to failure due to the helium blowdown and related energetics. For this reason, there is no significant improvement in the cumulative risk after filters are added and the plant boundary increased in Case D. This table and specifically the cumulative risk is the basis for the belief that a vented low pressure containment is superior to an essentially leak tight containment.

If it is assumed that <u>all</u> the plateout activity is released by liftoff during helium leaks, estimates provided in Table E-7 show that PAGs are met for reference fuel quality and no filters - Case A. Accident consequences for the large leak (SRDC-10) are essentially unchanged for Cases B and C since the blowdown bypasses the building filters. On-line measurement of circulating gases will provide early indication of abnormally high gaseous releases from heavy metal contamination and/or defective fuel. In addition periodic plateout activity measurements made directly or indirectly can also be used to establish plateout levels. The circulating and plateout inventories released in a depressurization event will be limited by Technical Specifications imposed on reactor operations. If lower than reference fuel quality was inadvertently used for the MHTGR, circulating activity levels could reach the specified Technical Specification limits. In an event such that the Technical Specifications are reached, the reactor would be shutdown and the problem corrected.

If it is further assumed that somehow a lower fuel quality (10x worse than reference) is inserted in the core and that the technical specification which controls circulating and plateout activities is ignored and a depressurization event were to occur the estimated doses will still meet 10CFR100 with margin with all plateout released. The results of Table E-7 confirm that the vented containments have large margins. Similar results are found with lower limit fuel quality and 100% washoff during SRDC-6 as discussed in the body of the report.

Table E-5

MHTGR RESPONSE TO LIMITING WATER INGRESS EVENTS (SRDC-6)[a] WITH ALTERNATIVE CONTAINMENT FEATURES AND SOURCE TERMS

Case		A	B	C	D	E	F	G
Relief line filter DF[b] for iodine		No 1	Yes[c] 10	Yes 100	Yes 100	Yes 100	No N/A	No N/A
Increase EAB 425 m to 805 m		No	No	No	Yes	Yes	No	Yes
Elevate release with stacks		No	No	No	No	Yes	No	No
Leak rate (%/day)		100	100	100	100	100	5	0.5
Est. incremental Cost, $ millions (Base is approx. $1.4 B)		Base	8	16	28	50	132	144
Nominal Dose, 0 - 30 days								
0. Reference fuel quality and behavior	Thy[d] Rem	1.51	0.308	3.08×10^{-2}	2.24×10^{-2}	1.45×10^{-3}	1.80×10^{-3}	9.85×10^{-5}
	WB[e] Rem	1.08×10^{-2}	3.49×10^{-3}	2.01×10^{-3}	1.49×10^{-3}	9.88×10^{-5}	7.58×10^{-5}	2.81×10^{-6}
1. Lower limit quality with reference behavior	Thy Rem	19.5	4.07	0.407	0.276	1.60×10^{-2}	3.36×10^{-2}	1.80×10^{-3}
	WB Rem	0.101	3.53×10^{-2}	2.08×10^{-2}	1.48×10^{-2}	9.33×10^{-4}	7.54×10^{-4}	2.66×10^{-5}
2. Rapid hydrolysis with lower limit fuel quality	Thy Rem	35.7	7.30	0.730	0.530	3.45×10^{-2}	4.26×10^{-2}	2.33×10^{-3}
	WB Rem	0.256	8.26×10^{-2}	4.76×10^{-2}	3.53×10^{-2}	2.34×10^{-3}	1.80×10^{-3}	6.66×10^{-5}

Notes: Shaded blocks represent cases that meet the EPA PAG dose limits of 5 Rem thyroid and 1 Rem whole body.

(a) SRDC-6 is initiated by a water ingress event as might result from a steam generator tube rupture.

(b) DF is decontamination factor

(c) Relief line filter acts on SRDC-6.

(d) Thyroid dose

(e) Whole body dose

Table E-6
MHTGR RESPONSE TO SMALL HELIUM LEAK EVENTS (SRDC-11) [(a)], WITH ALTERNATIVE CONTAINMENT FEATURES AND SOURCE TERMS

Case		A	B	C	D	E	F	G
Reactor Building filter DF[(b)] for iodine DF for particulates Filter bypass		No 1 1 N/A	Yes[(c)] 3 3 10%	Yes 30 30 1%	Yes 30 30 1%	Yes 30 30 1%	No N/A N/A N/A	No N/A N/A N/A
Increase EAB 425 m to 805 m		No	No	No	Yes	Yes	No	Yes
Elevate release with stacks		No	No	No	No	Yes	No	No
Leak rate (%/day)		100	100	100	100	100	5	0.5
Est. incremental cost, $ millions (base is approx. $1.4 B)		Base	8	16	28	50	132	144
Nominal Dose, 0 - 30 days								
0. Reference fuel quality and behavior	Thy[(d)] Rem	0.142	$7.41x10^{-2}$	$3.39x10^{-2}$	$1.73x10^{-2}$	$7.51x10^{-4}$	$1.77x10^{-3}$	$9.08x10^{-5}$
	WB[(e)] Rem	$4.91x10^{-4}$	$3.52x10^{-4}$	$2.69x10^{-4}$	$1.42x10^{-4}$	$5.47x10^{-6}$	$1.36x10^{-5}$	$4.87x10^{-7}$
0. Reference fuel quality and behavior	Thy Rem	~12-14	~4-7	~3	~1	~1	~1.5	~1.5
Cumulative[(f)] for events $>5x10^{-7}$/pl year	WB Rem	less than 1	less than 1	less than 0.01	less than 0.01	less than 0.01	less than 0.01	less than 0.01
1. Lower limit fuel quality	Thy Rem	1.35	0.708	0.327	0.167	$7.29x10^{-3}$	$1.71x10^{-2}$	$8.77x10^{-4}$
	WB Rem	$3.88x10^{-3}$	$2.62x10^{-3}$	$1.87x10^{-3}$	$9.95x10^{-4}$	$4.26x10^{-5}$	$1.24x10^{-4}$	$4.25x10^{-6}$
3. Weak fuel (Incremental Dose)	Thy Rem	24.4	12.8	5.84	2.99	0.129	0.305	$1.57x10^{-2}$
	WB Rem	$8.47x10^{-2}$	$6.07x10^{-2}$	$4.64x10^{-2}$	$2.44x10^{-2}$	$9.43x10^{-4}$	$2.35x10^{-3}$	$8.40x10^{-6}$

Notes: Shaded blocks represent cases that meet the EPA PAG dose limits of 5 Rem thyroid and 1 Rem whole body.

(a) SRDC-11 is initiated by a small break in the reactor coolant pressure boundary.
(b) DF is Decontamination Factor.
(c) Reactor Building filter acts on SRDC-11.
(d) Thyroid dose
(e) Whole body dose
(f) Cumulative risk is dominated by SRDC-11 like events

Table E-7
MHTGR RESPONSE TO MODERATE HELIUM LEAK EVENTS (SRDC-10)[a] WITH ALTERNATIVE VENTED LOW PRESSURE CONTAINMENT FEATURES AND SOURCE TERMS

Case		A	B	C	D	E	F	G
Relief line filter DF[b] for Iodine		No 1	Yes[c] 10	Yes 100	Yes 100	Yes 100	No N/A	No N/A
Reactor Building filter DF for iodine DF for particulates Filter bypass		No 1 1 N/A	Yes[d] 3 3 10%	Yes 30 30 1%	Yes 30 30 1%	Yes 30 30 1%	No N/A N/A N/A	No N/A N/A N/A
Increase EAB 425 m to 805 m		No	No	No	Yes	Yes	No	Yes
Elevate release with stacks		No	No	No	No	Yes	No	No
Leak rate (%/day)		100	100	100	100	100	5	0.5
Est. incremental cost, $ millions (base is approx. $1.4 B)		Base	8	16	28	50	132	144
Nominal Dose, 0 - 30 days								
0. Reference fuel quality and behavior	Thy[e] Rem	1.48×10^{-2}	1.34×10^{-2}	1.26×10^{-2}	5.90×10^{-3}	2.70×10^{-4}	4.83×10^{-4}	2.20×10^{-5}
	WB[f] Rem	2.81×10^{-4}	2.75×10^{-4}	2.72×10^{-4}	1.37×10^{-4}	3.81×10^{-6}	6.59×10^{-6}	2.81×10^{-7}
4.a Increased prompt source with reference fuel quality	Thy Rem	0.660	0.635	0.620	0.313	8.18×10^{-3}	5.59×10^{-3}	2.81×10^{-4}
	WB Rem	2.48×10^{-3}	2.40×10^{-3}	2.35×10^{-3}	1.19×10^{-3}	3.12×10^{-5}	2.56×10^{-5}	1.25×10^{-6}
4.b Increased prompt source with lower limit fuel quality	Thy Rem	6.60	6.35	6.20	3.13	8.18×10^{-2}	5.59×10^{-2}	2.81×10^{-3}
	WB Rem	2.48×10^{-2}	2.40×10^{-2}	2.35×10^{-2}	1.19×10^{-2}	3.12×10^{-4}	2.56×10^{-4}	1.25×10^{-5}

Notes: Shaded blocks represent cases that meet the EPA PAG dose limits of 5 Rem thyroid and 1 Rem whole body.

(a) SRDC-10 is initiated by a moderate break in the reactor coolant pressure boundary.
(b) DF is Decontamination Factor.
(c) RCPB relief line filter does not act on SRDC-10.
(d) VLPC filter acts on SRDC-10 only after the containment vent closes.
(e) Thy = Thyroid dose
(f) WB = Whole body dose

CONCLUSIONS AND RECOMMENDATIONS

This analysis shows that the MHTGR can meet its design goals using a vented, low pressure containment building. Based on U.S. and German experience to date, it is expected that the fuel and fission product behavior technology development program will confirm the specified fuel quality and behavior models. Furthermore, even under arbitrary assumptions regarding significantly lower than expected fuel performance, low pressure vented containment options can effectively reduce doses. The reference design with a VLPC meets the 10CFR100 and prompt fatality doses even if

a) a lower fuel quality corresponding to FSV's fuel is assumed,
b) a rapid hydrolysis source term is assumed,
c) a weak fuel source term is assumed, or
d) a lower limit fuel quality is assumed plus 100% liftoff/washoff.

However, to meet the advanced reactor policy guidance on defense-in-depth and in acknowledgement of the desire for conservatism in developing a new design concept, a number of alternative strategies are available to augment the current vented, low pressure containment design and to enhance fission product retention. A filter system for the RCPB relief line can be designed which will further reduce offsite doses, reduce residual risk, and has the additional benefits of improving operator safety and radioactive contamination control. A filter for the VLPC can be developed if required for further margins. The study shows that the MHTGR can meet site boundary PAGs for both an individual event and cumulative basis by adding these passive features to the VLPC. Even with additional filters, the VLPC is significantly more cost effective, is much more passive and compatible with passive decay heat removal systems, and requires less supervision and vigilance from plant operators than a conventional, high pressure, low leakage rate containment. When evaluated for low probability events, the vented low pressure containment is superior to the conventional containment because of its greater passivity, higher reliability, and fewer fault modes.

The vented low pressure containment strategy provides effective radionuclide retention for the performance specific characteristics of the MHTGR, namely a small prompt source term released with the helium blowdown decoupled from a delayed source term released with the gradual heatup of the core.

1. INTRODUCTION

1.1 BACKGROUND

Over the past eight years, the U. S. Department of Energy (DOE) has supported the development of a standard Modular High Temperature Gas-Cooled Reactor (MHTGR). Four Prime Contractors are currently engaged in this work. General Atomics (GA) and ABB-Combustion Engineering (ABB-CE) are responsible for the design of the reactor, vessel system, and heat transport loop while Bechtel National, Inc. (BNI) and Stone and Webster Engineering Corp. (SWEC) are responsible for the auxiliary systems, power conversion systems, and plant structures. DOE is also supporting related research and development work at the Oak Ridge National Laboratory (ORNL). Utility/user support for the program and guidance for the design is primarily through Gas Cooled Reactor Associates (GCRA).

The MHTGR is designed to meet top level regulatory requirements (Ref. 8). These include 10CFR100 guidelines which limit the doses to the public resulting from off-normal events that are not expected within the plant lifetime. Conservatively calculated doses to a member of the public at the site boundary are limited to 300 Rem to the thyroid or 25 Rem to the whole body. In addition, two key utility/user requirements expressed through GCRA are important to the design of the MHTGR (Ref. 9):

- The MHTGR shall have no need for off-site emergency planning or preparedness drills. To meet this requirement, doses to the public at the exclusion area boundary (EAB) must not exceed the Environmental Protection Agency (EPA) Protective Action Guidelines (PAGs, Ref. 6) lower threshold values for evacuation and sheltering (5 Rem thyroid, 1 Rem whole body). These dose limits apply to individual events of the following types:

  - Design Basis Events (DBEs) with frequencies down to 10^{-4} per plant year,
  - Emergency Planning Basis Events (EPBEs) with frequencies down to 5×10^{-7} per plant year.

  This requirement also means that the minimum Emergency Planning Zone (EPZ) radius shall be equal to the minimum exclusion area radius. To achieve this goal, the frequency with which the PAG threshold doses are exceeded due to any and all events shall be less than 5×10^{-7} per plant year. The process for setting the EPZ radius for Light Water Reactors is used to guide the interpretation of this requirement.

- MHTGR shall meet its safety goals without reliance on emergency AC electrical power or operator intervention and must be tolerant of inappropriate operator action.

To meet these objectives, specific design features of the MHTGR, including the size, configuration and power density of the core, key materials of construction, and the configuration of important systems, were selected to provide inherent or passive means for ultimately accomplishing key safety functions. In addition to active systems to insert negative reactivity and remove decay heat, the MHTGR has the following capabilities:

- Containment of fission products by multiple barriers, the most significant of which are the fuel kernel and coatings: Subsequent barriers to fission product release are the fuel rod matrix and fuel element graphite, the reactor coolant pressure boundary, and the reactor building. Because the fuel coating integrity is very high and there are no fuel coating failure modes which depend on the integrity of subsequent barriers, the demands placed on the reactor coolant pressure boundary and on the reactor building are very modest. The multilayered fuel coatings (TRISO coatings) include a layer of porous carbon to allow space for the evolution of fission product gasses, multiple layers of pyrocarbon, plus a layer of silicon carbide. The entire coating system is made of ceramic materials and retains fission products over the full range of licensing basis events (LBEs).

- Control of heat generation by negative temperature coefficient: The fuel composition and graphite moderator result in a highly negative temperature reactivity coefficient which limits power excursions.

- Removal of decay heat by passive means: If both the normal heat transport system (HTS) and the shutdown cooling system (SCS) are unavailable, heat is rejected by convection and radiation to the air-cooled reactor cavity cooling system (RCCS), which rejects heat to the atmosphere by natural circulation. This heat rejection method is not dependent on the presence of the helium coolant in the reactor.

By selecting reactor parameters and support systems carefully, the MHTGR avoids any event which could result in temperatures in the core high enough to damage the fuel coatings. As a result, the primary source of mobile fission products in the spectrum of licensing basis events (LBEs) is the very small fraction of fuel which is defective when manufactured, the only fuel expected to fail during operation. Defective fuel can occur when particles fail to receive one or more complete layers during the TRISO coating application process or when uranium contaminates the exterior of otherwise good particles. When fission occurs in defective particles or in heavy metal contamination, the resulting fission products become distributed within the MHTGR system by a slow process of migration. A small amount of these fission products escape from the fuel matrix and enter into the reactor coolant, where they either remain in circulation with the helium coolant or deposit on the surfaces of the reactor and heat transport system. These circulating and plateout constituents can be released by some events in a matter of minutes. The remainder of the fission products which are formed in defective fuel or heavy metal contamination remain within the fuel matrix, and can only be released during some events over a period of hours or days. These "prompt" and "delayed" source terms are discussed in greater detail in Section 2.

Thus, a fundamental parameter in the development of the MHTGR design is the fuel manufacturing defect fraction. Based on extensive experience in the U. S. and in Germany, the MHTGR designers have developed a fuel quality specification which limits the defect fraction to the point where no more than 5 in 100,000 fuel particles are expected to fail during normal operation (Ref. 4). This quality level is central to the source terms formulated for use in evaluating the adequacy of the MHTGR safety design. To provide sufficient assurance that the specified quality can be achieved, a test program encompassing manufacture, inspection, irradiation, and fission product transport characterization has been established.

The MHTGR has been designed to be housed in a below grade reactor building which is capable of venting to the atmosphere following a discharge of primary or secondary coolant inside the building. To provide assurance that this design meets all of its objectives, a broad range of potential events has been identified and evaluated and a set of DBEs selected to represent various event families. These events, the plant response, including that of the vented reactor building, and the resultant offsite dose consequences are discussed in the Preliminary Safety Information Document (PSID, Ref. 1), prepared by the MHTGR Program in 1986 and submitted to the U. S. Nuclear Regulatory Commission (NRC) for review. The NRC prepared a Pre-application Safety Evaluation Report (PSER, Ref. 3), which was issued as a draft in 1989. A continuing series of discussions, presentations, responses to questions, and PSID amendments have occurred in support of the effort to issue the PSER in final form. In response to an NRC request, the program prepared and submitted a Containment Study for the MHTGR (Ref. 4) which surveyed alternative reactor building features. In their review of that document, the NRC requested a further evaluation of the fuel quality and possible additional fuel failure mechanisms (Ref. 5, included as an Appendix).

This document re-evaluates key accidents and the response of the MHTGR reactor building and alternative building features for fuels of varying quality and for additional fuel failure mechanisms. This evaluation also considers design aspects of the MHTGR that have been further developed since the preliminary safety evaluations. One important feature that has changed is the size of the reactor. The reactor design described in the PSID has 66 columns of fuel and produces 350 MW(t). In order to meet the specified utility industry economic goals, a larger reactor with 84 columns of fuel along with one additional ring of unfueled graphite reflector blocks in the center region and rated at 450 MW(t) has been proposed (Ref. 7). As a result of this change in the reactor size about 30% more power can be generated. In addition, the helium temperatures have been changed, increasing the temperature differentials in the steam generator and resulting in a more optimum steam generator which is larger in diameter but about the same volume as that required for the 350 MW(t) design. Fertile particles of thorium have been replaced with particles of natural uranium, reducing the decay heat load. Because the unfueled mass in the center of the core is increased, the core power density is approximately the same, and the heat transfer path from the innermost fuel to the reactor vessel is unchanged, peak fuel temperatures during events which involve a loss of all forced cooling are approximately the same in the 450 MW(t) design as they were with the 350 MW(t) core. Several other aspects of the plant design have also evolved. This report describes the 450 MW(t) MHTGR design, but the issues, criteria, phenomena, and conclusions would apply equally well to the 350 MW(t) design.

1.2 DESCRIPTION OF THE MHTGR CONTAINMENT SYSTEM

The MHTGR Containment System consists of four barriers to fission product transport. These barriers, depicted schematically in Fig. 1-1, act to ensure that, for the full range of licensing basis events, there will be no release of radionuclides into the environment of sufficient magnitude to adversely affect public health and safety. The fissile kernel, consisting of blended uranium oxide and uranium carbide, is itself a significant impediment to fission product transport. The Containment System barriers which surround the fissile material are as follows:

- Barrier 1: Fuel Kernel and Particle Coatings - The kernel of a failed fuel particle will retain more than 95% of the radiologically important, short-lived fission gases such as Kr-88 and I-131. However, the effectiveness of a kernel for retaining gases is reduced if the exposed kernel is hydrolyzed by reaction with water vapor. Barrier 1 relies primarily on the silicon carbide layer, which is impervious to most radionuclides over the full range of normal and accident temperatures. For a small portion of the approximately 10 billion coated fuel particles in an MHTGR core, this barrier will not be fully intact following fabrication. The silicon carbide layer is basic to the fuel performance model and the source term calculation.

- Barrier 2, the Fuel Rod Matrix and Fuel Element Graphite: The graphite binder which is used when coated particles are formed into cylindrical fuel compacts is low in density and highly sorptive of metallic fission products but provides little holdup of fission product gases released from a failed particle. In contrast, the graphite of the fuel element blocks into which the fuel

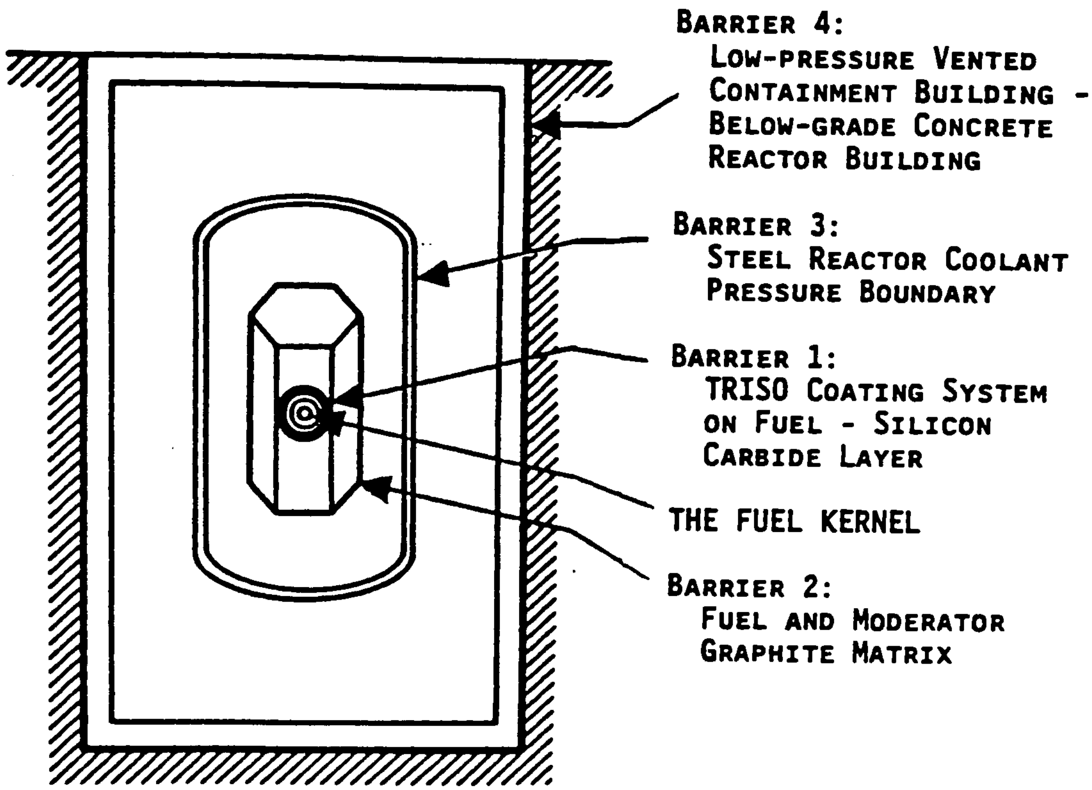


Figure 1-1 THE MHTGR CONTAINMENT SYSTEM

compacts are inserted is denser and somewhat less sorptive of fission metals but is more effective as a barrier to diffusion.

- Barrier 3, the Reactor Coolant Pressure Boundary (RCPB): The reactor vessel, steam generator vessel, and cross vessel and their connections and appurtenances act to retain the helium reactor coolant and any radioactive material entrained in it. Plateout on the relatively cool inside surfaces of the RCPB also serves to reduce the level of these fission products circulating in the helium coolant during normal operation. Even if a breach of the RCPB should occur, it still serves as a fission product retention mechanism. While almost all of the circulating activity in the escaping coolant is released during a rapid depressurization event, only a small fraction of the plateout activity will be re-entrained, with the bulk of it retained within the RCPB. When the release of coolant is slow, the vessels provide a volume in which to hold up those fission products which are released from the core during the latter stages of the event. The MHTGR release model only takes into account radioactive decay and conservatively neglects gravitational settling and plateout inside the reactor and steam generator vessels under these conditions. Since decay heat will be rejected by conduction and radiation, the loss of helium coolant inventory does not pose a threat to the radionuclide retention capabilities of either barrier 1 or barrier 2.

- Barrier 4, the Vented Low Pressure Containment (VLPC): The VLPC is a normally closed space which will act to retain fission products. It is equipped with a vent which will open if the pressure inside the VLPC exceeds the vent's design setpoint, releasing the mass and energy associated with a blowdown, thereby protecting the integrity of the building and the RCCS. The VLPC inhibits the release of fission products to the environment by plateout, deposition, and decay even if the vent opens. While the vent allows the release of fission products released promptly, the release of associated gasses early in the event eliminates the driving pressure which could transport the delayed source term out of the building.

The arrangement of the MHTGR Containment System barriers is sequential; that is, Barrier 1 is completely enclosed by Barrier 2, which is completely enclosed by Barrier 3, which is completely enclosed by Barrier 4. However, as will be described further in Section 3, the inventory of radionuclides is not completely within Barrier 1. That very small fraction of fuel which was defective at manufacture (missing or failed coating layers) and the fissile material which was unintentionally transferred to the outside of some fuel particles during fabrication (heavy metal contamination) will lead to a fission product inventory which can eventually contaminate the reactor coolant and plate out on the inside surfaces of the RCPB.

1.3 PURPOSE OF THIS REPORT

The purpose of this report is to evaluate the MHTGR response to alternative source term assumptions for a range of vented low pressure containment designs. Specifically, the program has evaluated alternative passive containment features assuming 1) the possibility of higher fuel defect fraction, 2) very rapid hydrolysis of defective fuel, 3) the possibility that the fuel could suffer an undetected weakness, and 4) a higher than expected release of plateout activity. This report directly responds to an NRC letter requesting additional studies on these topics (Morris, B. M., Letter to Dr. Sol Rosen, May 9, 1990).

The report begins by discussing the design of the fuel and the source, location, and mobility of radionuclides within the MHTGR. The processes used to identify and evaluate design basis events are described, including the method of evaluating the estimated offsite dose consequence relative to the PAG threshold doses. The events which are limiting are described in more detail. The proposed 450 MW(t) MHTGR design is described, with a focus on the barriers to fission product transport. The range of source terms as determined by fuel quality and performance is established. The report then evaluates the spectrum of additional mitigative plant features as they pertain to safety consequences and describes the plant response to challenging events. Then the capability of the containment system including various additional mitigative features is evaluated assuming higher source terms corresponding to variations in the fuel behavior model. Finally, conclusions and recommendations are discussed.

2. DESCRIPTION OF REFERENCE PLANT DESIGN FEATURES

The MHTGR design is based on a synthesis of US and German experience and has evolved over the past 10 years. The reactor fuel is based on ceramic coated fuel particles embedded in hexagonal prism graphite moderator blocks. The core is assembled with fueled blocks arrayed in an annular cylinder, with unfueled reflector blocks located in the center of the cylinder and around the outside of the fueled region for neutron economy.

The core is contained inside an uninsulated steel pressure vessel. Heat is transferred from the reactor to the steam turbine system by a single heat transport loop using a submerged electric motor driven circulator and a helical-tube, once-through steam generator located in an adjacent vessel. The vessels are connected by a concentric cross vessel. High pressure helium is the primary reactor coolant. Hot helium (1300°F) from the core outlet at the bottom of the reactor flows through the inner hot duct in the cross vessel to the top of the steam generator. Helium flows downward over the steam generator, where heat is transferred to the secondary coolant. Cool helium (550°F) flows upward to the top of the steam generator vessel, where it enters a single-stage axial compressor, driven by an electric motor which is located inside the helium pressure boundary. Helium then flows through the outer part of the cross vessel to the reactor vessel and upward between the vessel wall and the core barrel to the top of the core.

When the reactor is shutdown, decay heat can be removed by the Heat Transport System (HTS). If the HTS is unavailable, the Shutdown Cooling System (SCS), with its own helium circulator and heat exchanger, located in the bottom of the reactor vessel, and its own heat sink, can control the temperatures in the core. A natural draft air-cooled system of ducts and panels (the Reactor Cavity Cooling System or RCCS) surrounds the reactor vessel and is the third means of removing decay heat from the reactor.

The reactor is refueled by replacing fuel blocks in alternating columns. Half of the core is replaced during each refueling outage, scheduled to occur at approximately eighteen month intervals. Access to the core is obtained through the control rod drive penetrations. Shielded machines are used to unstack and then restack the core in sixty-degree sectors. To facilitate fuel handling operations and to provide a structure which serves all the needed functions at the lowest cost, the reactor is located in a fully embedded building. The lower portion of the structure is cylindrical to resist external soil and groundwater pressure. The standard MHTGR power plant is composed of four MHTGR reactor modules arranged in a row to facilitate sharing of fuel handling equipment. Sequential deployment may be accommodated to comply with changing load growth and/or financial constraints. Common facilities for fuel handling, radioactive waste management, and personnel services are located at one end of the reactor row. The reactor modules and auxiliary support systems which require a nuclear grade of construction or which support the reactor systems are physically separated from the turbine generators and their related systems. The two parts of the plant are termed the Nuclear Island (NI) and the Energy Conversion Area (ECA). Each of the four reactors is connected to one of four power conversion systems consisting of a non-reheat steam turbine and associated feedwater system. There is no cross connection or mixing of steam cycle systems, although the MHTGR concept allows several different owner-specified ECA arrangements. Common power conversion support systems, including mechanical draft cooling towers, maintenance facilities, and switchyard, are provided. A single plant control building housing a common control room is included in the ECA.

Figure 2-1 shows the reactor horizontal cross section and illustrates the annular core concept. The core, vessel system, and heat transport loop are shown in Figure 2-2. Control rods enter the core by gravity from the top. The outer ring of control rods are used for power control. The inner rods are used for startup. Reserve shutdown control material (boronated pellets) can be inserted into channels adjacent to the inner control rod locations. The fueled region of the core is approximately 15 ft in outer diameter and 25 ft high. The unfueled center of the core has a diameter of approximately 8 ft. Table 2-1 lists important reactor core parameters.

Table 2-1
KEY PARAMETERS - 450 MW(t) REACTOR CORE

Fuel columns	84
Core layers	10
Power density (w/cc)	6.0
Vessel ID (ft)	23.7
Coolant pressure (psia)	1025.0
Core inlet temp. (°F)	550.0
Core outlet temp. (°F)	1300.0
Refueling interval (months)	18.0
Operating control rods	24
Startup control rods	12
Reserve shutdown channels	12

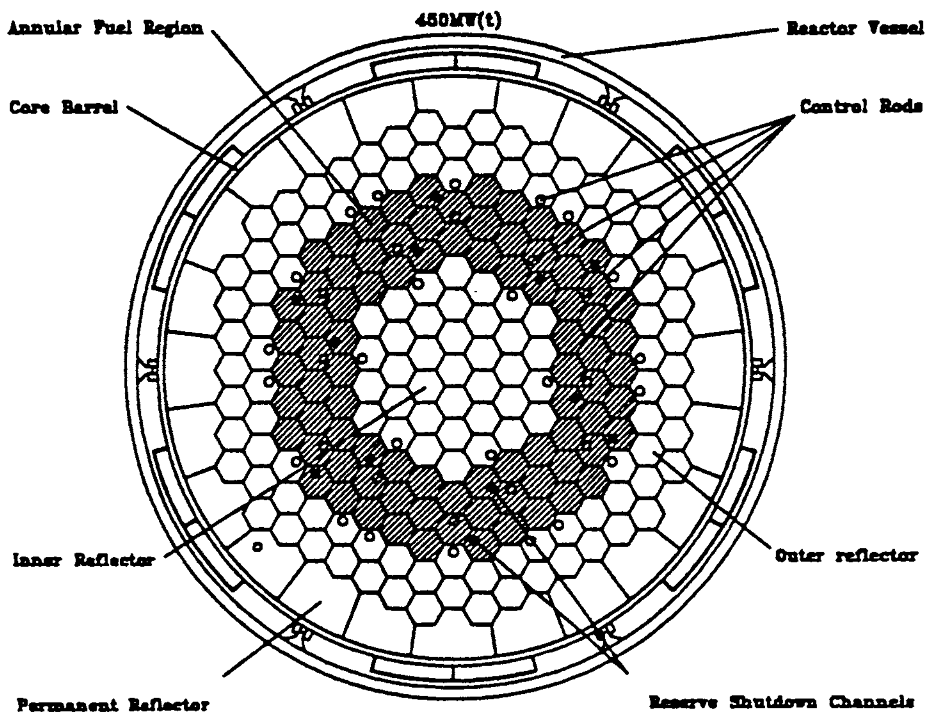


Figure 2-1 CORE CROSS-SECTION

The reactor coolant pressure boundary (RCPB) consists of three carbon steel vessels and their appurtenances. The entire system is constructed of SA-533 Grade B steel plates and SA-508 Grade B forgings, commonly used in the construction of light water reactor vessels. The key dimensions and parameters for each of the three vessels are shown on Table 2-2. The vessels are fabricated in a factory and shipped to the construction site. Present expectations are that the steam generator will be shipped in the steam generator vessel. The cross vessel will be field welded to the reactor vessel and steam generator vessel but the closures on both of the large vessels will be bolted. The reactor vessel is required to maintain the geometry of the core

Table 2-2
CHARACTERISTICS OF RCPB VESSELS

Parameter	Reactor Vessel	Cross Vessel	Steam Generator Vessel
Inside diameter	23 ft. 8.5 in.	6 ft. 8 in.	17 ft. 3 in.
Maximum wall thickness	7.5 in.	2.25 in	6.25 in.
Overall length	89 ft. 3.9 in.	14 ft 0 in.	82 ft. 7 in.
Largest piping connection open inside to reactor coolant	2 in. - Helium Purification System	None	3.5 in. - Main helium relief valves
Largest piping connection which is closed inside	6 in. - Shutdown Cooling Water System	None	36 in. - main steam tubesheet access (14 in. Main steam line)
Design code	ASME B&PV Code Section III	ASME B&PV Code Section III	ASME B&PV Code Section III
ISI code	ASME B&PV Code Section XI	ASME B&PV Code Section XI	ASME B&PV Code Section XI

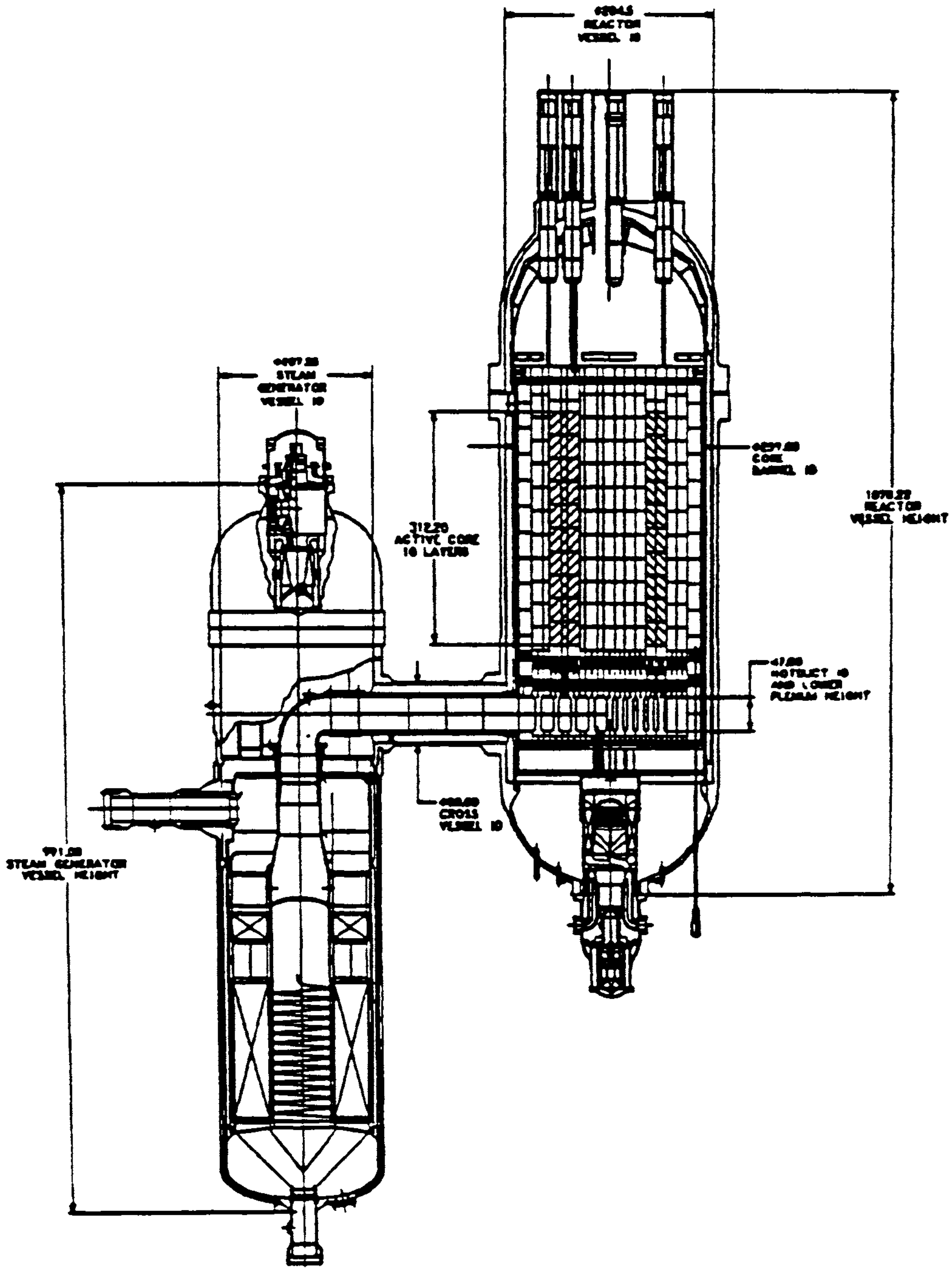


Figure 2-2 REACTOR ELEVATION VIEW

for decay heat rejection. However, the MHTGR can reject decay heat passively through the graphite and uninsulated reactor vessel to the RCCS. As a result, it is not essential to maintain the presence of the helium coolant in the reactor in order to accomplish the decay heat removal functions.

The RCPB helium relief valve configuration used for the 350 MW(t) design is shown schematically on Figure 2-3. It has two relief trains, each consisting of a 100 percent-capacity relief valve followed by a rupture disk, which is provided solely to limit losses of hard-to-contain helium. The design proposed for the 450 MW(t) MHTGR is shown in Figure 2-4. It consists of three, 33⅓ percent capacity relief valves, preceded by rupture disks. (While the ASME B&PV Code does not currently permit a rupture disk upstream of a relief valve, a code inquiry to permit such an arrangement in the case of the MHTGR has been initiated.) The valve trains in both the 350 MW(t) and the 450 MW(t) designs discharge into the reactor building in the vicinity of the main circulator. Gasses are generally able to flow through the building and will open the vent and exit to the environment. Radionuclides which may be entrained in the discharged gasses are reduced during this process by deposition and decay within the building.

The reactor building for the proposed 450 MW(t) design is shown in Figures 2-5 through 2-9. It is a multicelled, embedded structure constructed of cast-in-place reinforced concrete. The degree of embedment was selected to serve a number of objectives, including reduced cost and complexity of construction, ease of operation, minimization of shielding, and good seismic performance. The operating floor of the power plant is set at site grade, with a common maintenance enclosure covering the operating area traversed by shared refueling equipment. There are two floors below grade with a rectangular footprint which are used to house mechanical, electrical, and instrumentation systems dedicated to each reactor. A number of additional mechanical and electrical systems which do not require radiation shielding or protection from external hazards are designed to be delivered to the site as prefabricated modules and located at grade outside the maintenance enclosure. The reactor building below elevation -30 ft is configured as a cylinder to enable it to resist soil and ground water pressure. The reactor, vessel, and heat transport systems are located within this space. Access to and from the cylindrical portion of the building for piping, electrical services, personnel, and the concentric RCCS ducting is made from the rectangular portion of the building between elevations -30 ft and grade. Access for refueling and for major maintenance activities is from the operating floor. There are two extensions of the reinforced concrete reactor building above grade. On the west side of the building, adjacent to the steam generator, an elevated portion of the building provides protection for the main steam relief valve discharge stacks and is part of the reactor building vent path. On the east side of the reactor building, the reinforced concrete portion of the building extends to elevation + 95 ft. 6 in. to serve as the Reactor Cavity Cooling System elevated inlet-outlet structure.

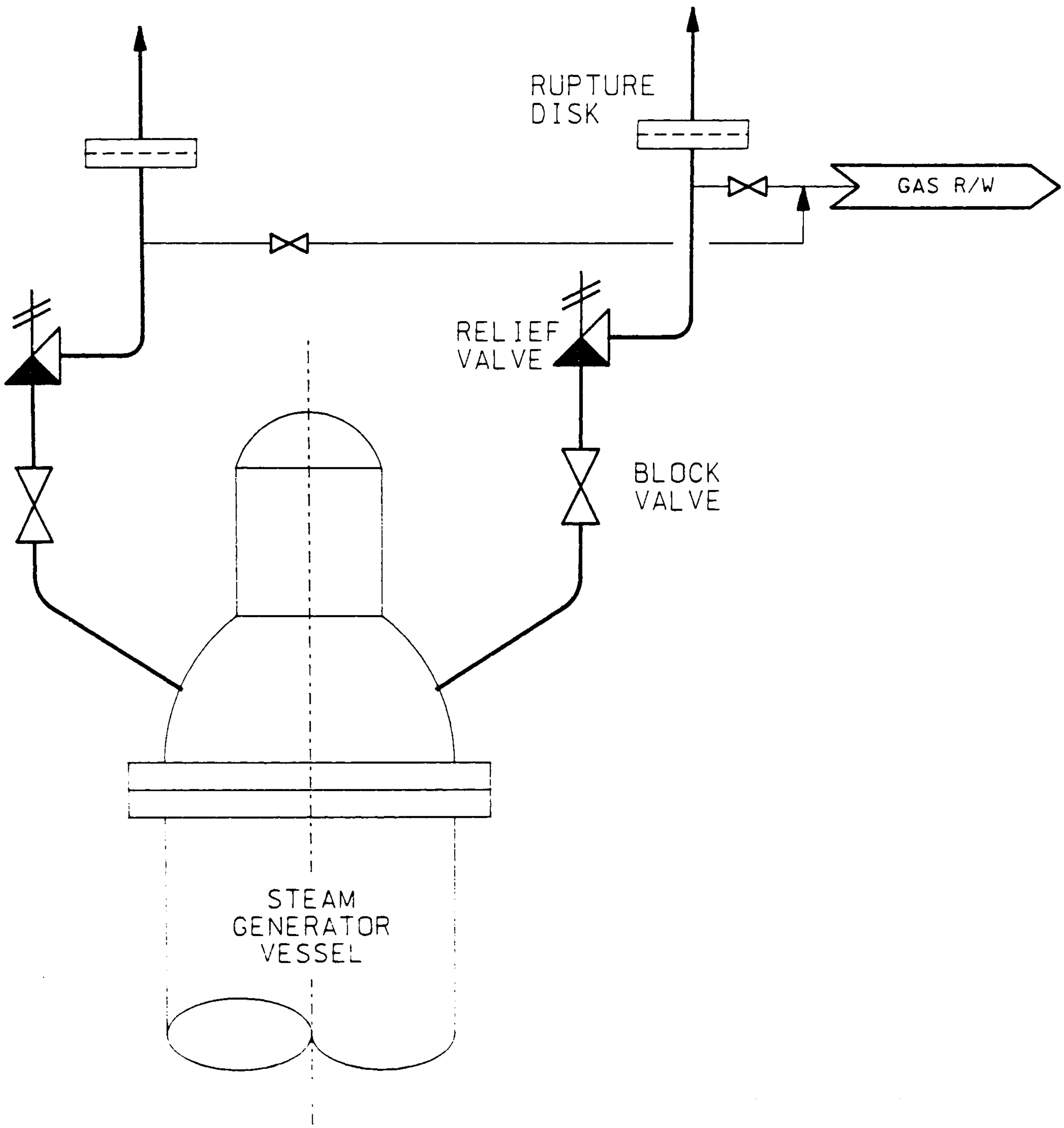


Figure 2-3 HELIUM RELIEF TRAIN CONFIGURATION - 350 MW(t) MHTGR

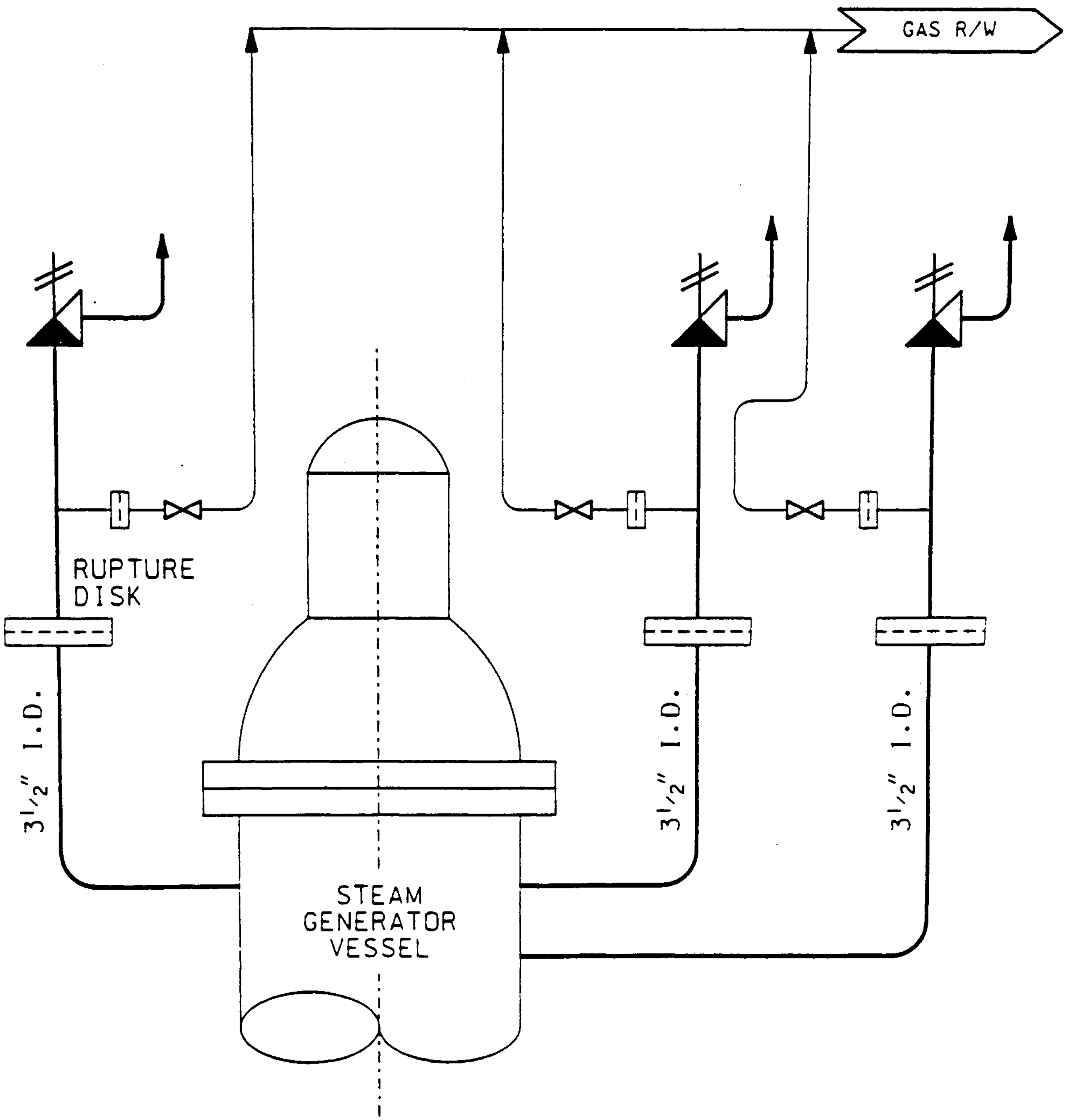


Figure 2-4 HELIUM RELIEF TRAIN CONFIGURATION 450 MW(t) MHTGR

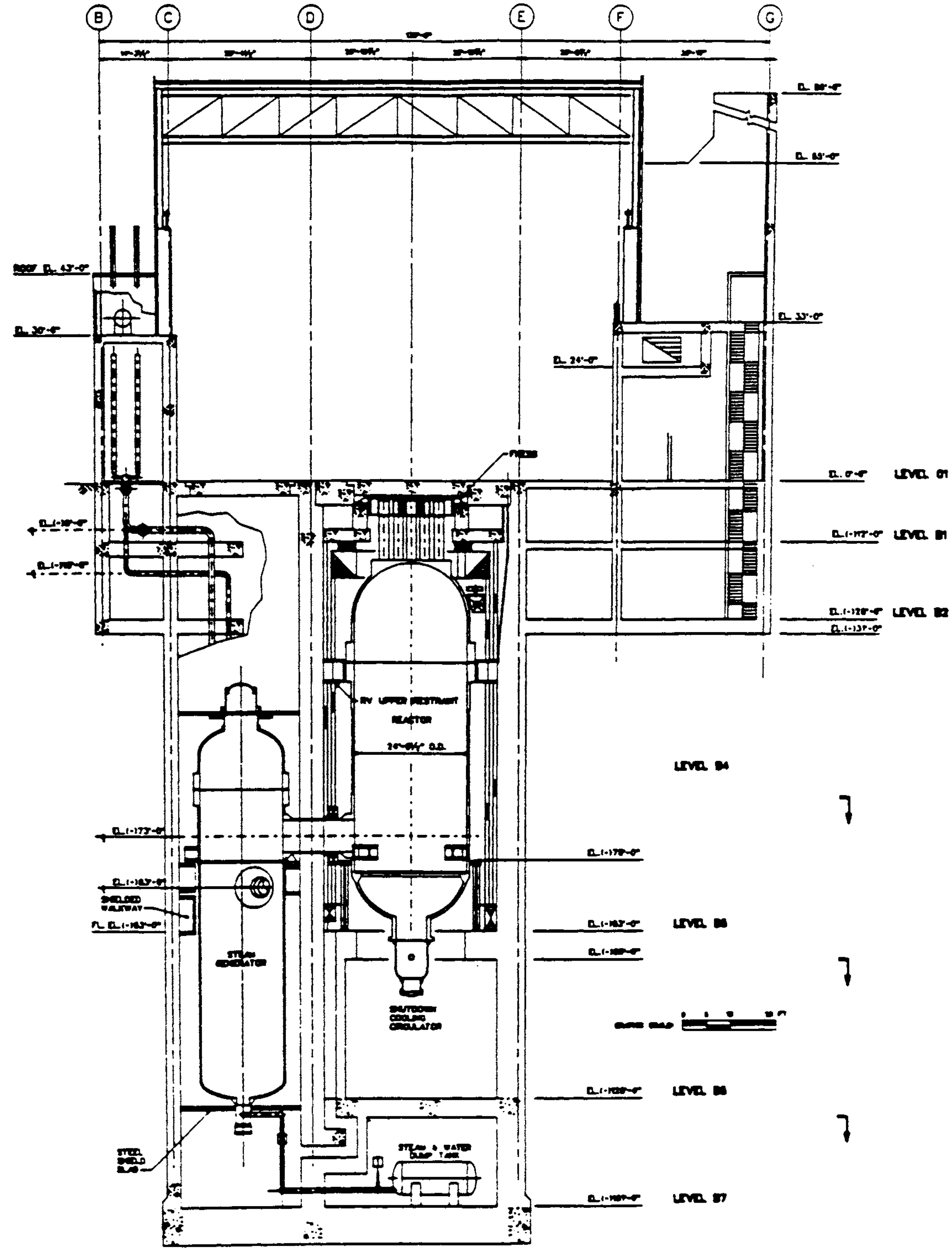


Figure 2-5 REACTOR BUILDING - ELEVATION VIEW THROUGH VESSELS

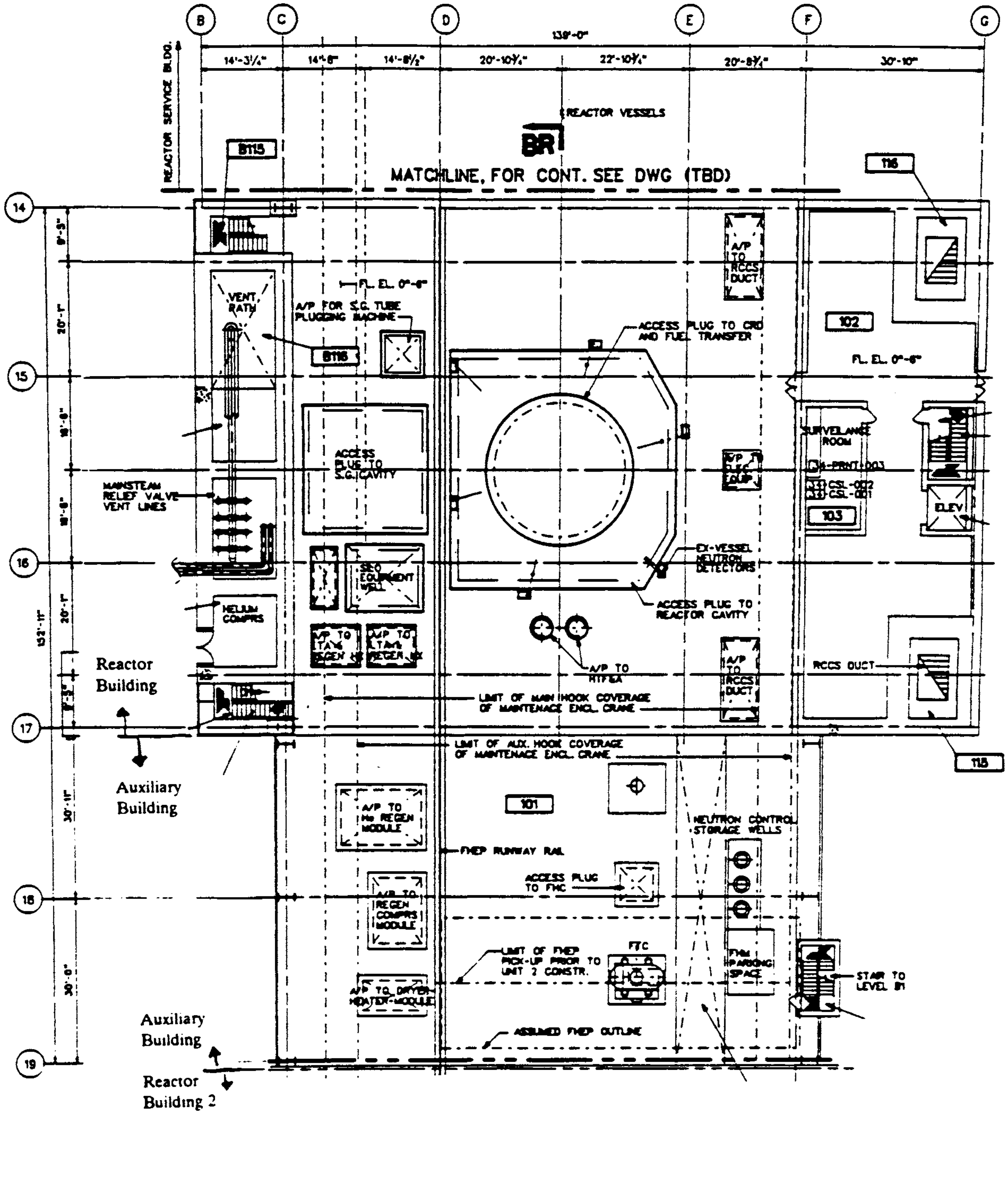


Figure 2-6 REACTOR BUILDING - PLAN AT GRADE

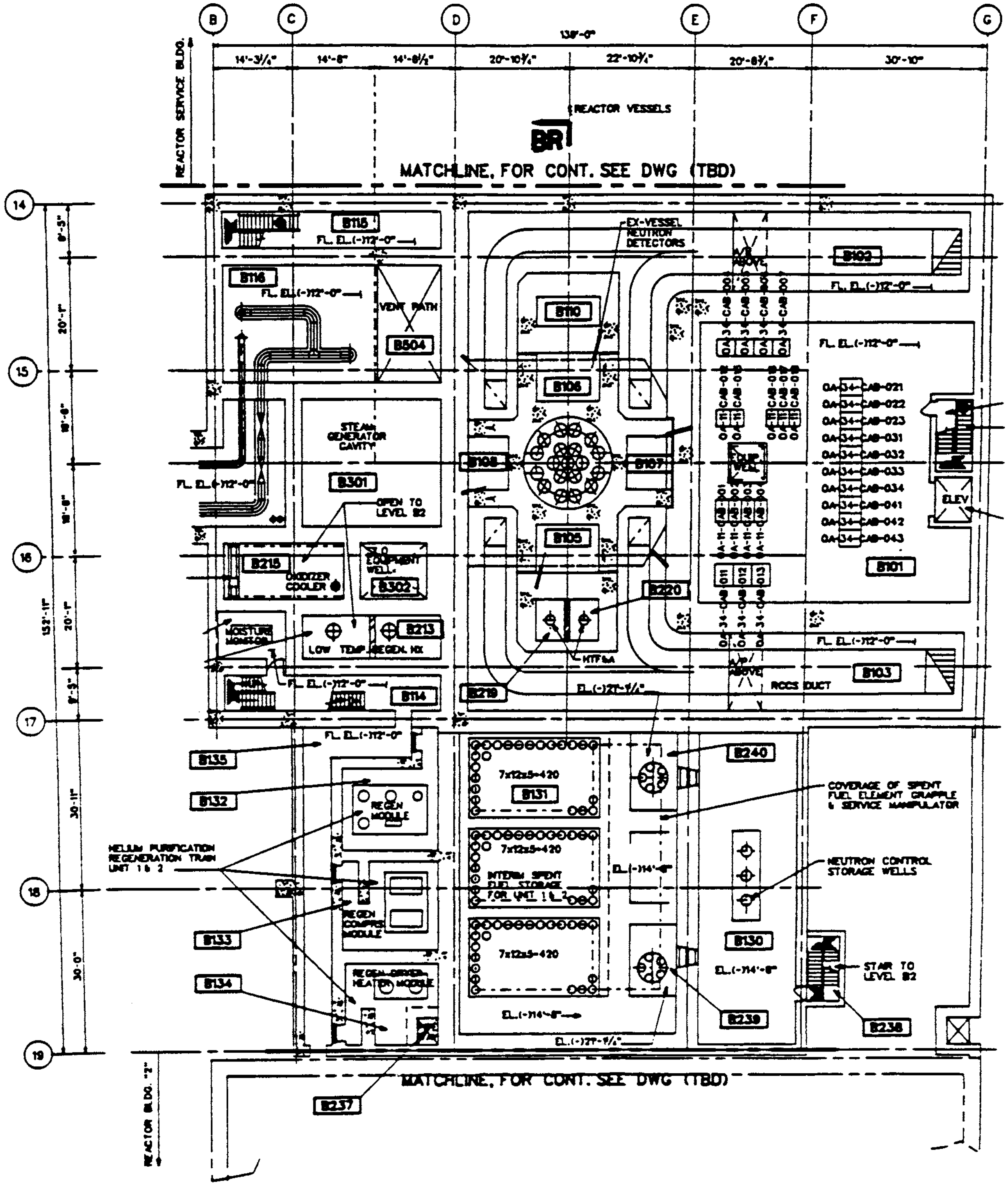


Figure 2-7 REACTOR BUILDING - PLAN AT LEVEL B1

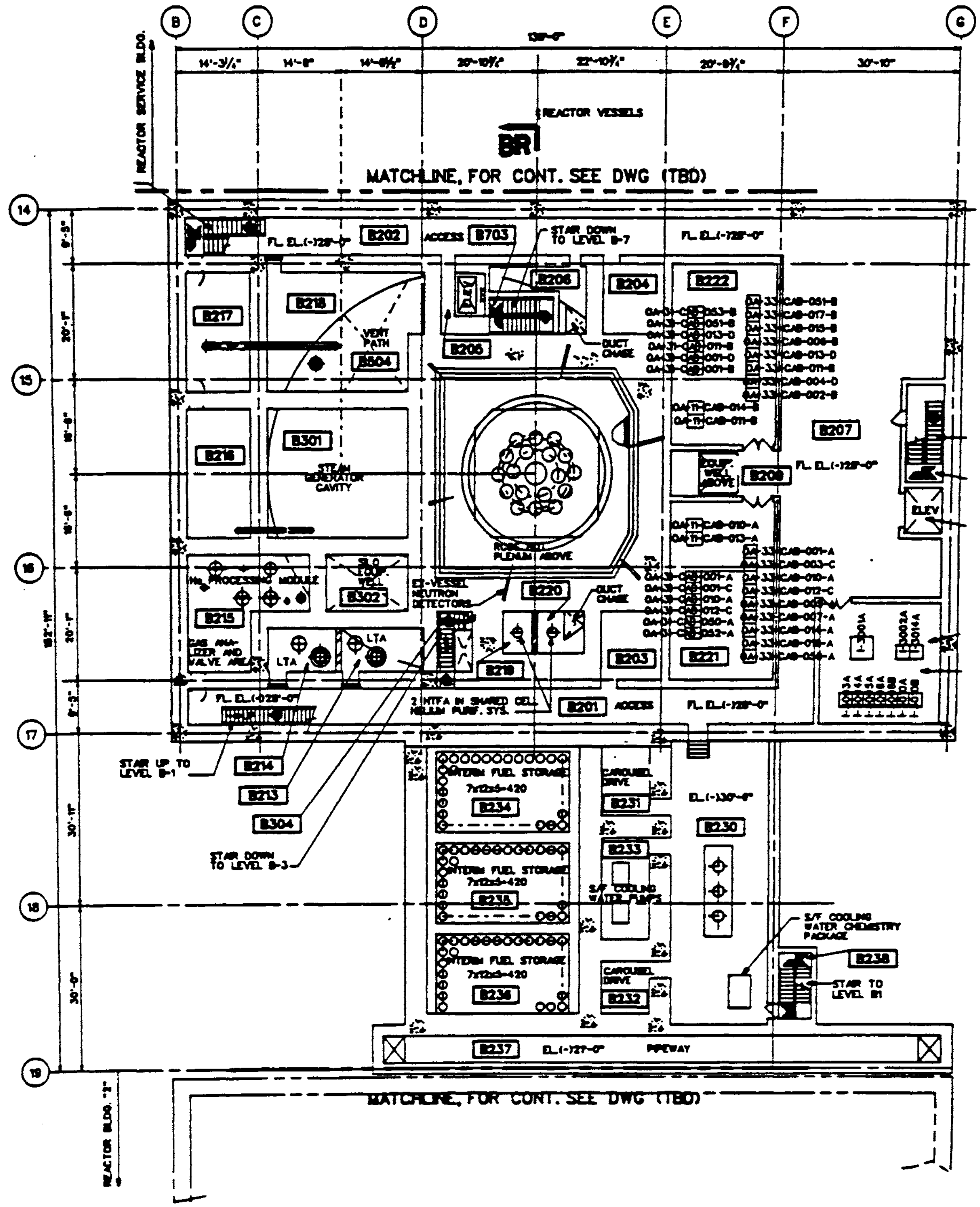
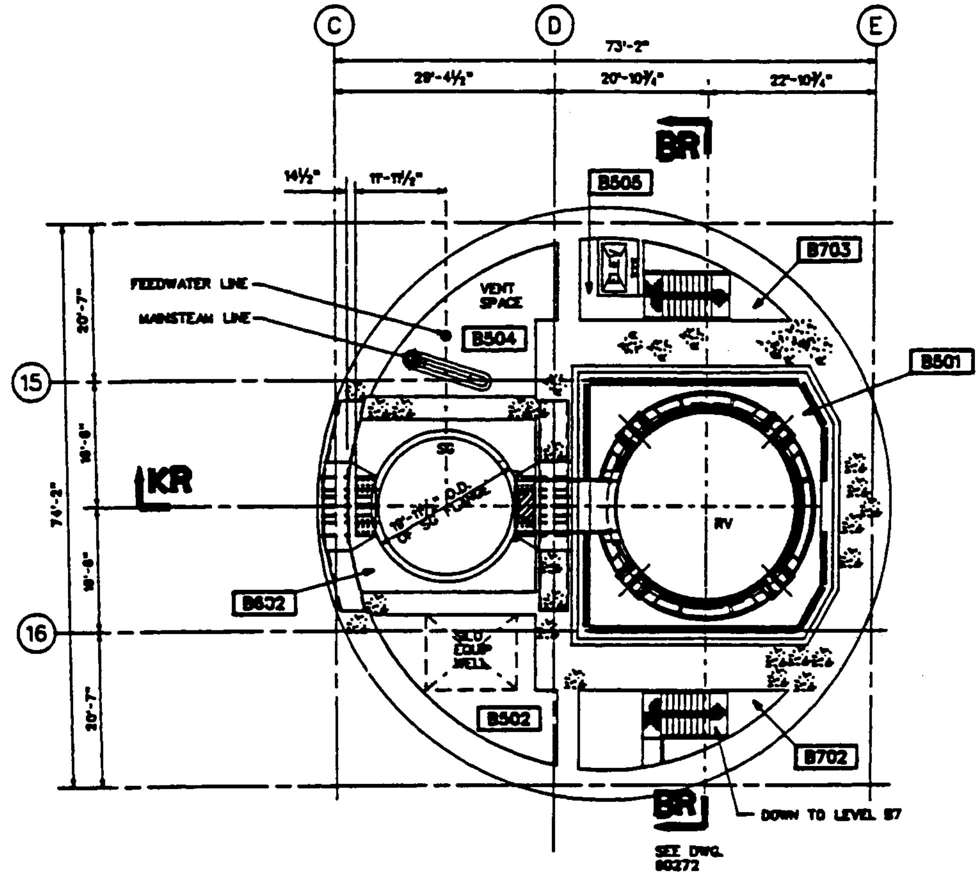


Figure 2-8 REACTOR BUILDING - PLAN AT LEVEL B2



LEVEL B4 PLAN AT EL.(-)78'-0"

Figure 2-9 REACTOR BUILDING - PLAN AT CROSS VESSEL ELEVATION

The reactor building has been divided into two distinct zones for purposes of the heating, ventilating, and air conditioning (HVAC) design. The cells containing the helium purification system, the vent path sections above grade west of the maintenance enclosure, and most of the cells in the cylindrical portion of the building have been designed to form a closed, interconnected space which is normally isolated from the environment. Air is recirculated internally and heat is removed by chilled water-cooled air handling units. The balance of the rectangular portion of the building, the personnel access stairways, the personnel elevator shaft into the silo portion of the building, and the space containing the steam and water dump tank have been designed to be conditioned by a once-through flow of heated or cooled air. The RCCS panels, where they enter the closed portion of the reactor building, are regarded as part of the vented low pressure containment boundary. In essence, air flowing inside the RCCS ducts and panels is outside of the reactor building. The walls, doors, plugs, and other barriers which separate the closed, recirculated portion of the building from the once-through cooled portion of the building or from the outside environment (including the RCCS panels and ducts) constitute the fourth containment barrier. Leakage from within this portion of the reactor building to the other part of the reactor building or to the environment has the potential to transport fission products from the containment to the environment. This space is also the portion of the reactor building which is affected by the specified building leak rate. The net free volume within this space is approximately 260,000 cubic ft. Figure 2-10 shows an expanded orthogonal visualization of this space. This space is designed to have a leak rate of no greater than 1 volume per day at an internal pressurization of 1 psid, and to vent whenever the internal pressure exceeds 1 psid. It is expected that essentially none of the leakage which occurs will be from the surfaces of the building which are in contact with the soil, and that the specified leak rate represents an upper bound on the exchange which could occur between the building interior and the environment, since the pressure and therefore the leakage will normally decrease over the course of an accident. Architectural features such as doors, gaskets around floor plugs, and penetrations are important to establishing the building leak rate but can be modified to achieve the specified value.

In the event of a large primary or secondary coolant leak within the closed portion of the reactor building, the internal pressure will exceed 1 psid. Gasses are able to flow from any compartment through the building and out the vent path relief valves or dampers to the atmosphere, as depicted on Figure 2-11. If a break were to occur in the reactor cavity, helium would be able to flow through a shielded labyrinth into the steam generator compartment, via a one-way damper. Energetic pipe failures are more likely in the steam generator compartment. If a blowdown occurs in the steam generator area, gasses are able to flow downward to the bottom of the steam generator compartment, and then to the north side of the silo. They will then follow the main feedwater and main steam lines upward to the building vent. The vent dampers are maintained in a closed position by gravity, and the weight of the damper plate determines the relief setpoint pressure, which is the internal pressure needed to open the damper. This design must be considered preliminary. The relief setpoint

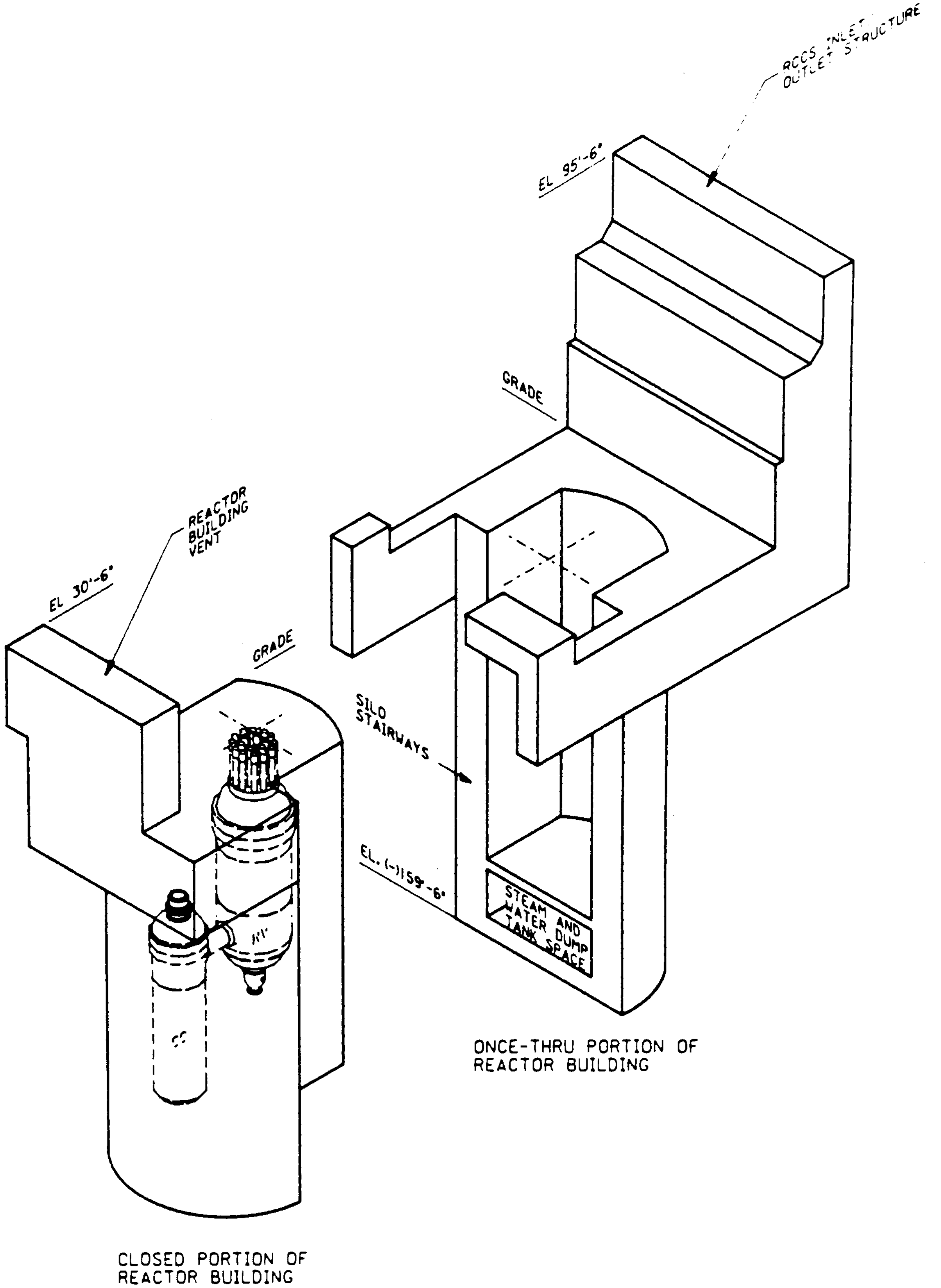


Figure 2-10 EXPANDED VIEW OF REACTOR BUILDING SHOWING HVAC STRATEGY

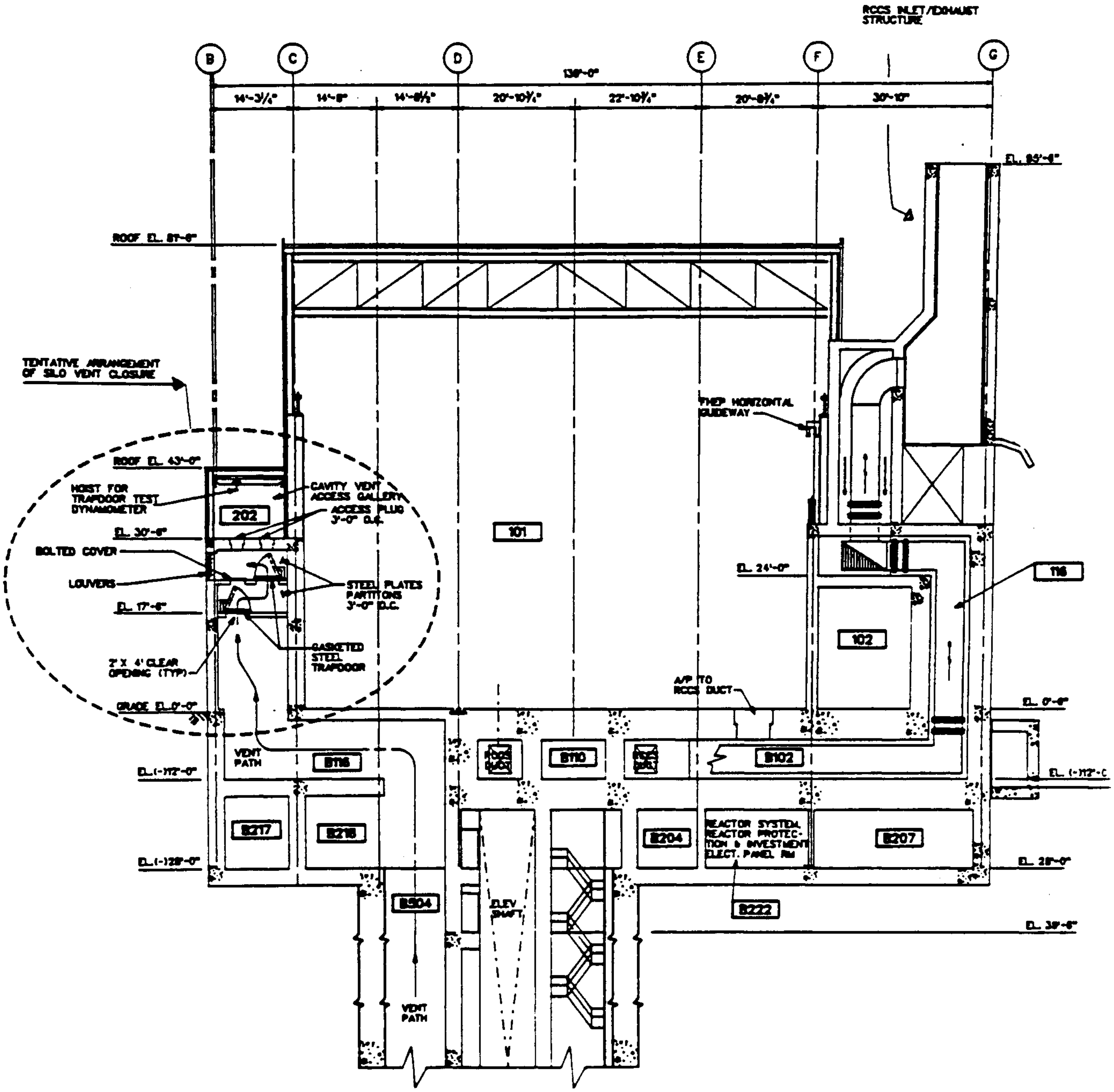


Figure 2-11 REACTOR BUILDING - ELEVATION THROUGH RCCS DUCT AND BUILDING VENT

pressure affects both the nominal reactor building leak rate and the building pressure transients following a large primary or secondary coolant leak. The building relief setpoint pressure and vent opening area can both be adjusted if needed to obtain satisfactory performance during a pressure transient. The reinforced concrete building and RCCS panels have been designed to withstand pressure transient loadings of 10 psid.

Both the leakage across the reactor building boundary and the gasses which are vented to the atmosphere via the reactor building vent are considered to be ground level releases. Radionuclides released from the MHTGR are assumed to travel a minimum of 425 meters to the site exclusion area boundary (EAB) before they result in the exposure of a member of the public. As a consequence, the concentration of radionuclides, which is a determinant of both the inhalation and immersion doses, is reduced in transit by atmospheric diffusion and, to a much lesser extent, deposition on the ground and decay. This effect is included in the atmospheric diffusion coefficient (χ/Q) in dose conversion calculations.

3. SOURCE TERM SUMMARY

3.1 EQUILIBRIUM RADIONUCLIDE INVENTORIES

The TRISO fuel coating is the first and most important barrier to fission product transport. There are approximately 10^{10} coated particles in the core of each reactor. Microspheres of blended uranium carbide and uranium oxide are coated with several layers of ceramic material, as shown in Figure 3-1. A negligible fraction of particles is expected to fail in service under normal conditions. The only mechanism which has been identified which can cause failure of intact particles is elevated temperature (exceeding 2900°F, 1600°C) for an extended period of time (Ref. 1). The design selections made for the core and heat removal systems are intended to make events which can result in fuel temperatures which approach the threshold of TRISO coating damage exceedingly unlikely. Therefore, essentially all of the fission products which arise within intact particles remain immobilized inside the particle coatings. The fission products which make up the source terms for both normal conditions and licensing basis events (LBEs) arise in fuel particles which have manufacturing defects.

Most of the fission products which are formed in defective particles also remain within the uranium oxy-carbide kernel. However, some of those fission products, as well as those from heavy metal contamination, may become mobile during normal operation and some accident events. Their migration from the kernel to the reactor coolant is slowed and inhibited by the second barrier to release, namely the fuel rod matrix and fuel element graphite. However, a small portion will eventually become mixed with the circulating helium coolant. Fission products which are not volatile at or below the core inlet (cold leg) helium temperature will plate out on the cooler surfaces within the reactor and steam generator vessels and on the submerged heat transport loop components. Fission products which enter the helium coolant and which are volatile below the cool helium temperature are continuously removed from the coolant, along with non-radioactive contaminants such as water, hydrogen, and hydrocarbon compounds, by the helium purification system. The amount of fission products in equilibrium at various locations within the MHTGR depends on the fuel defect fraction, the fraction of heavy metal contamination, the length of time the fuel has been in the core, the temperature history of the irradiated fuel, and the operating history ofthe helium purification system. Fission products are characterized as being resident in one of the following five locations:

- Circulating Activity. Fission products in this category are entrained in the flowing helium and will be released from the reactor vessel if the helium coolant escapes. The transport of this source term is inhibited by Containment System barriers 3 and 4.

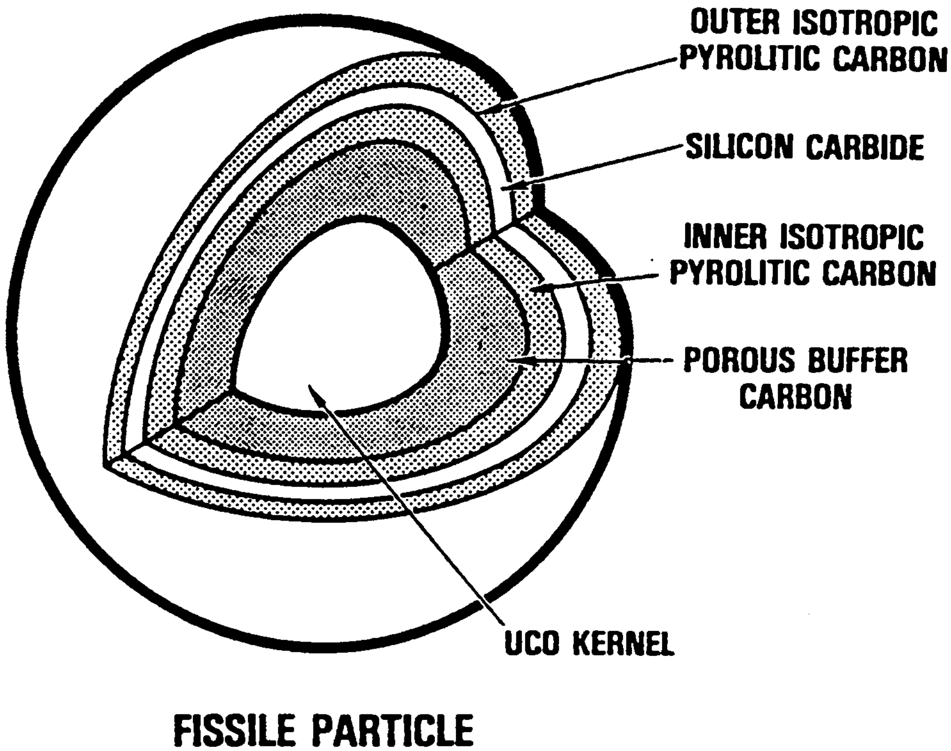


Figure 3-1 TRISO COATED PARTICLE

- Plateout Activity. Fission products in this category are located on the inside surfaces of the reactor coolant pressure boundary (RCPB) and on the heat transport and reactor system components in contact with the helium coolant. They tend to be concentrated on the cooler surfaces. A break in the primary coolant pressure boundary large enough to cause rapid blowdown of the helium could result in the re-entrainment of a fraction of these fission products and their release from the RCPB. Plateout activity could also be washed off and made available for release by the action of steam or water during an event involving a steam generator tube failure. The release of this source term is dependent on events that involve specific mechanical failures. The transport of this source term is also inhibited by Containment System barriers 3 and 4.

- Core Activity in Defective Particles. Fission products in this category are located within fuel kernels but are potentially mobile because they are not retained by fully intact TRISO coatings (Containment System barrier 1). High core temperatures or fuel hydrolysis following a water ingress event may exacerbate their transport from the kernels into the reactor coolant. The mechanisms which can transport this inventory across the fuel-graphite matrix barrier are generally time dependent. The transport of this inventory to the environment is inhibited by Containment System barriers 2, 3, and 4.

- Core Activity Due to Heavy Metal Contamination. Fission products in this category behave like the fission products in defective particles.

- Core Activity in Standard Particles. The vast majority of all fission products are in this category. There are no identified events which could credibly lead to the release of this activity into the helium coolant or, therefore, outside the vessel. Thus, while all four barriers of the MHTGR Containment System would act to inhibit the transport of this activity, its release is averted primarily by barrier 1, the TRISO fuel coating. The integrity of barrier 1 is not threatened by a failure of barrier 3, the RCPB, to maintain the helium coolant inventory.

As an indication of the relative radionuclide inventories in each of these locations, Table 3-1 summarizes the I-131 inventory distribution in the 450 MW(t) MHTGR reactor module.

The MHTGR fuel specification defines the fraction of radionuclide inventory outside the standard fuel particles (Ref. 1). The key fuel specification values for initial inventory quantification are the fraction of defective fuel with missing layers of porous buffer carbon (see Figure 3-1) and the fraction of heavy metal

Table 3-1
I-131[a] INVENTORY AVAILABLE FOR POTENTIAL RELEASE FROM
A SINGLE MODULE OF THE 450 MW(t) MHTGR

Source	Inventory (Ci)	Timing of Release	Release Mechanisms	
			From Core	From Primary Coolant Pressure Boundary
A. Circulating	0.03	Minutes	--	Leakage flow (He depressurization)
B. Plateout	26	Minutes	--	Leakage flow (He depressurization) Moisture (water ingress)
C. Outside standard particles				
1. Missing coatings (failed SiC and/or OPyC)[b]	600	Minutes-hours[c] Hours-days	Moisture (water ingress) Temperature (loss of forced cooling)	Relief valve (He depressurization) Leakage flow (He depressurization)
2. Contamination	120	Hours-days	Temperature (loss of forced cooling)	Leakage flow (He depressurization)
D. Standard particles	1.2×10^7	--	Temperature (no event identified)	--

[a] Contributes to thyroid dose.

[b] Silicon carbide (SiC); outer pyrolytic carbon (OPyC).

[c] About 42 curies of the inventory (representing the UC_2 in nontact fuel particles) is subject to release within minutes after hydrolyzing.

contamination outside the protective TRISO coatings. The fuel specification establishes the expected value for the missing buffer fraction at $\leq 5 \times 10^{-5}$ and the 95th percentile value at $\leq 2 \times 10^{-4}$. The expected heavy metal contamination fraction for compact lots is $\leq 1 \times 10^{-5}$, and the 95th percentile value is $\leq 2 \times 10^{-5}$.

The 450 MW(t) MHTGR source term data presented in Table 3-1 are estimated by scaling the 350 MW(t) MHTGR plant inventories. In this study, the nominal inventory of nonintact fuel particles is estimated to be equal to the missing buffer fraction of 5×10^{-5} times the total core inventory of 1.2×10^7 curies of I-131. The nominal heavy metal contamination inventory is estimated by multiplying 1×10^{-5} by the I-131 inventory.

The data for circulating and plateout expected inventories are based on nominal equilibrium plant operation at 100% power. Upper bound or 95th percentile values for I-131 are approximately a factor of four higher than the nominal circulating and plateout values shown in Table 3-1.

Additional calculations will be required to better reflect the changes proposed for the fuel cycle, including the replacement of thorium fertile particles in the 350 MW(t) design with natural uranium fertile particles in the 450 MW(t) design.

3.2 DESCRIPTION OF DOMINANT EVENTS FOR OFFSITE CONSEQUENCES AND RISK

A spectrum of possible licensing basis events (LBEs) for use in determining the site suitability source terms have been identified for the reference 350 MW(t) MHTGR through the application of a rigorous and structured analytical process (Refs. 1, 10). It is expected that the same set of events will also govern the evaluation of offsite radionuclide releases for the 450 MW(t) MHTGR. Table 3-2 lists the scenarios which are expected to lead to a release of radionuclides from the plant. The mean event frequency and the applicable criteria against which adequacy of fission product retention is measured are also identified in Table 3-2 (Ref. 10).

3.2.1 Release Characterization

The MHTGR configuration was developed to ensure the integrity of the standard (no defect and within specification) fuel particle such that the radionuclide inventory is retained within the fuel particles under all credibly conceivable events. Thus, the only available significant source for potential radionuclide release is outside the standard particles. As shown in Table 3-1, the smallest sources, the circulating and plateout activities within the primary circuit, have the potential for prompt release to the reactor building. Because this release is linked to the accidental leakage of the gaseous helium coolant from the vessel system, this release could characteristically occur within minutes during a coolant release.

The remaining sources of radionuclides are within the core graphite but outside of standard, intact particles and take longer to be released. Since the mechanisms leading to release from these defective fuel particles depend on core temperature, which increases very slowly due to the large heat capacity of the massive graphite moderator and low power density of the core, this delayed source term is characteristically released over a period from hours to many days in a manner which is highly specific to the accident scenario. However, it is possible for a small fraction of the inventory from fuel particles with as-manufactured defective coatings to be released rapidly. Such a rapid release (minutes to hours) from these defective particles is postulated to occur during events in which high core temperatures occur coincident with a large moisture ingress which provides

Table 3-2
350 MW(t) MHTGR RELEASE SCENARIOS
SELECTED FOR SITE SUITABILITY SOURCE TERMS

Release Scenarios	Frequency/Plant year	Applicable Criteria	Evaluation Confidence
Routine small releases associated with RCCS air and other service system sources	Normal plant operation	10CFR50 Appendix I	Best Estimate
Small primary coolant leak with forced core cooling (AOO-5)	3×10^{-1}	10CFR50 Appendix I	Best Estimate
Moisture inleakage without forced core cooling (DBE-7)	4×10^{-5}	10CFR100	High (95 th %ile)
		Lower PAG	High (95 th %ile)
Moderate Primary coolant leak with forced core cooling (DBE-10)	2×10^{-2}	10CFR100	High (95 th %ile)
		Lower PAG	High (95 th %ile)
Small Primary coolant leak without forced core cooling (DBE-11)	3×10^{-3}	10CFR100	High (95 th %ile)
		Lower PAG	High (95 th %ile)
Moisture ingress with delayed steam generator isolation and without forced core cooling (EPBE-1)	7×10^{-6}	Lower PAG	Best Estimate (50 th %ile)
Moisture ingress with delayed steam generator isolation and with forced core cooling (EPBE-2)	5×10^{-6}	Lower PAG	Best Estimate (50 th %ile)
Primary coolant leak in four modules without forced core cooling (EPBE-3)	3×10^{-5}	Lower PAG	Best Estimate (50 th %ile)

Legend
AOO = Anticipated Operational Occurrence
DBE = Design Basis Event
EPBE = Emergency Planning Basis Event

reactants that can hydrolyze the carbide portion (approximately 7%) of the UCO fuel. In summary, the potential activity releases, as shown in Table 3-1, can be grouped into two broad categories, a small prompt release and a larger delayed release.

3.2.2 Risk Dominant Event Identification

For the purpose of determining the adequacy of the MHTGR Containment System for a range of source terms, a set of risk dominant accidents representative of each of the MHTGR accident families with the potential for releasing fission products to the environment were selected. A mechanistic, internally consistent assessment was performed for each of these accidents, the most significant of which are listed in Table 3-2. All accidents, including the events in Table 3-2 have previously been grouped into four accident families. For purposes of this study these families will be represented by two conditions previously evaluated in the PSID namely, SRDC-6 and SRDC-11. The frequencies and resulting evaluation criteria required in order to ensure that the PAGs are not exceeded at the site boundary are shown in Table 3-3. The two limiting SRDCs are described briefly in Table 3-4 and more fully in Ref. 1. These two types of events, small RCPB leaks and water ingress, dominate the risk

Table 3-3
EVALUATION CRITERIA FOR ACCIDENT FAMILIES USED FOR THIS STUDY

Accident Family Based on 350 MWt PRA (Ref.2)	Accident Family Frequency per Plant Year	Consequences Represented by Safety Related Design Condition (SRDC)	Utility/User Requirement
Primary Coolant Leak With Forced Cooling	~ 0.35	Small Helium Leak with Depressurized Conduction Cooldown (SRDC-11)	Dose at 5×10^{-7} per Plant Year less than or equal to 5 Rem Thyroid or 1 Rem Whole Body
Primary Coolant Leak Without Forced Cooling	~ 3×10^{-5}		
Water Ingress With Forced Cooling	~ 7×10^{-5}	Water Ingress with Depressurized Conduction Cooldown (SRDC-6)	
Water Ingress Without Forced Cooling	~ 5×10^{-5}		

and consequences, respectively. The SRDC-11 events are small leaks in the reactor coolant pressure boundary with loss of all forced cooling, which are important because the helium coolant is still exhausting from the reactor vessel when the core temperatures reach a level high enough to cause the inventory outside standard particles (source C, Table 3-1) to be released from the core. The flow of helium transports some of the inventory out of the reactor vessel, and it is able to find its way into the environment. In events which involve larger breaks, the vessel is fully depressurized before the temperature rises significantly, so there is no significant outflow of helium from the vessel when the temperature causes radionuclides in this category to be released from the core.

Table 3-4
DESCRIPTION OF RISK DOMINANT ACCIDENTS

Accident Characterization	Initiating Event	Reactor Trips	HTS Cools	SCS Cools	SG Iso-lated	SG Dumped	Prim. Relief Valve Opens	RCCS Cools
Small helium leak with depressurized conduction cooldown (SRDC-11)	0.05 $in.^2$ primary coolant leak	Yes	No	No	N/A	N/A	No	Yes
Water ingress with depressurized conduction cooldown (SRDC-6)	Steam generator tube failure	Yes	No	No	Yes but Delayed	No	Yes	Yes

Legend
HTS = Heat Transport System
SCS = Shutdown Cooling System
RCCS = Reactor Cavity Cooling System

The second type of event, represented by SRDC-6, involves water ingress from a steam generator tube failure. Feedwater pressure in the MHTGR is approximately 3450 psi. This high pressure and a large head loss across orifices in the feedwater tube sheet are needed in order to assure uniform flow of feedwater in all tubes. However, this results in a secondary coolant pressure in the tubes which is more than 1000 psi greater than the primary helium coolant pressure. In this event, the release from the RCPB occurs when the ingressing water flashes to steam, causing the pressure to rise past the setpoint of the helium relief valves. If the steam generator tube failure is small, the pressure may rise slowly and the time before the pressure is sufficient to cause the relief valve to lift may be long enough that there will have been a significant temperature-induced evolution of radionuclides from the core (source C, Table 3-1). In addition, there will have been a significant time during which steam can react with and hydrolyze the exposed fuel kernels. There is a probability that the relief valve will open and reseat one or more times and then fail open. If the relief valve behaves in this way, it exacerbates the radioactive release by storing the energy which transports the radionuclides out of the vessel until a greater radionuclide inventory is available for release. SRDC-6 is based on conservative assessment of this possibility and assumes that the relief valve(s) sticks open 21 hours after the start of the event.

Table 3-5 presents the key I-131 source terms associated with these limiting events and, thus, used for assessing the capability of the MHTGR containment. These source terms are based on estimates of 450 MW(t) fission product equilibrium inventories, afterheat generated following full power operation, and the revised US/FRG fuel performance models (Ref. 11). The source terms shown in Table 3-5 are all releases to the primary coolant except those in the bottom row, which are the associated releases from the RCPB to the reactor building. Retention and decay inside the vessel system account for the difference between the sum of the activity released to the coolant and that released from the vessel system. The releases occur over a relatively long period of time. Figures 3-2 and 3-3 show the time dependence of the releases for SRDC-6 and SRDC-11.

Table 3-5
I-131 INVENTORY RELEASED BY LIMITING EVENTS - 450 MW(t) MHTGR
(Nominal Fuel Temperatures)

			Release Events			
			Depressurized loss of Forced Cooling (SRDC-11)		Water Ingress with Depressurized Loss of Forced Cooing (SRDC-6)	
Source	Inventory (Ci)	Timing	Mechanism	Release (Ci)	Mechanism	Release (Ci)
A. Circulating	0.03	Minutes	Helium blowdown	0.03	Helium blowdown	0.03
B. Plateout	26	Minutes	Liftoff	0.04	Steam washoff	15.4
C. Defective fuel						
1. Missing coatings	600	Minutes -hours			Hydrolysis	219
2. Contamination	120	Hours -days	Heatup	87	Heatup	72
Total	720			87		291
D. Standard particles	1.2×10^7	-	0		0	
	Total Release from Core			87		291
	Total Release from Vessel System			48		266

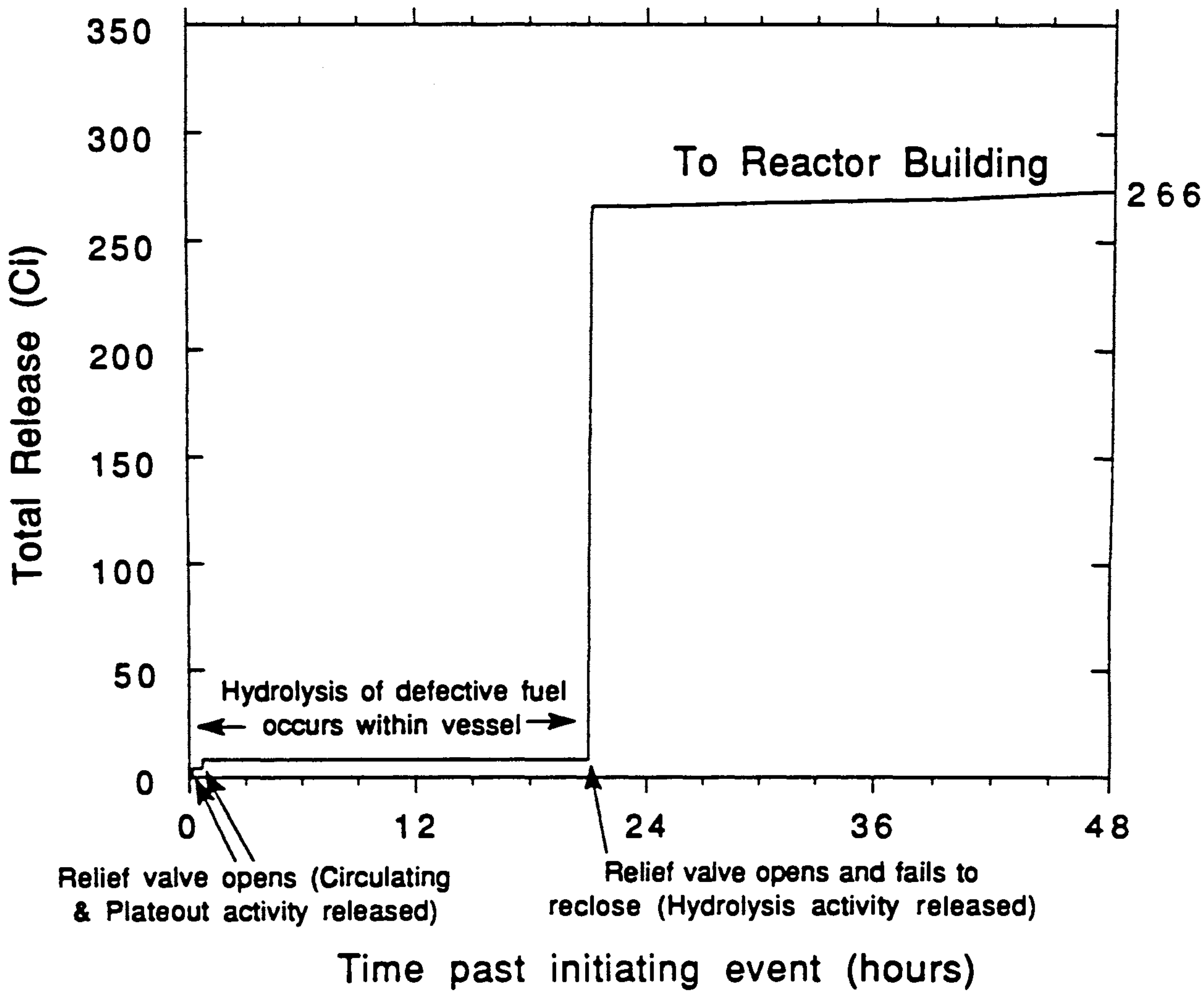


Figure 3-2 RELEASE OF I-131 DURING WATER INGRESS EVENT (SRDC-6)

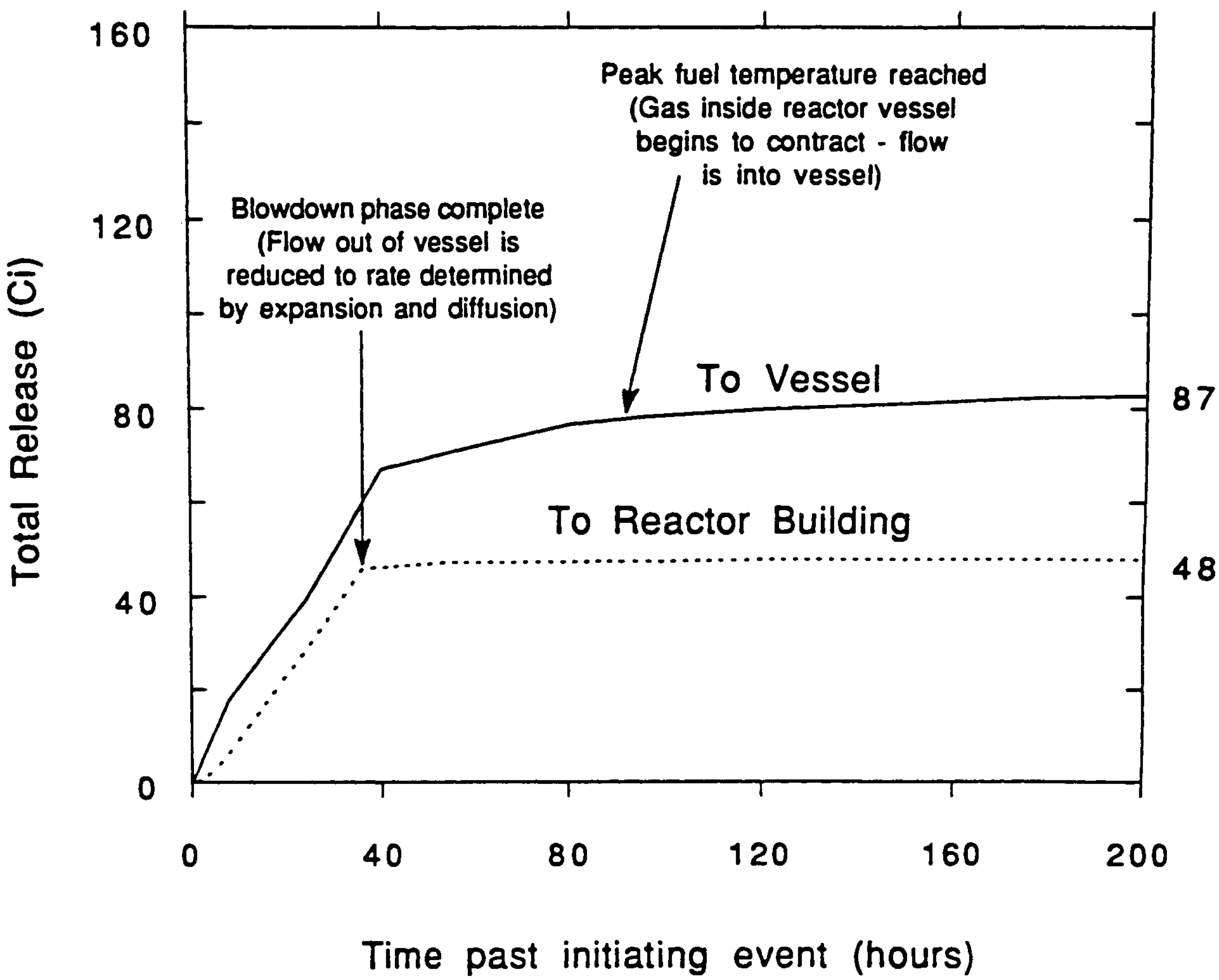


Figure 3-3 RELEASE OF I-131 DURING SMALL HELIUM RELEASE EVENT (SRDC-11)

3.3 METHODS FOR EVALUATING EVENTS WHICH CHALLENGE THE MHTGR CONTAINMENT SYSTEM

The selected scenarios, each a sequence of events, have been determined by probabilistic risk analysis to represent the risk dominant event in a family of potential sequences. The elements which are important to the MHTGR containment include reactivity and temperature transients, chemical reactions, and mass and energy transfer. The inventory of radionuclides which can potentially be released (sources A, B, and C, Table 3-1) and the containment barriers which must be traversed (fuel particle, fuel matrix, vessel system, and reactor building) are represented in a set of models and analyzed using computer codes. The analysis done of the 450 MW(t) MHTGR design for this report is similar to that which was done for the PSID of the 350 MW(t) design.

The MHTGR reactor building has been designed with dampers, which are gravity-closed vent valves. They will open whenever the building internal pressure exceeds the setpoint. The preliminary design concept uses two separate vent paths, each of which has two valves in series, to assure adequate protection against the possibility that one valve may either fail to open or fail to reclose. In analyses performed for the reference 350 MW(t) and the proposed 450 MW(t) designs, the vent is assumed to open when the building internal pressure equals 1 psid. The strategy to vent the reactor building results in the prompt release of gasses discharged into the reactor building whenever the internal pressurization of the building exceeds the vent setpoint. After the vent has reclosed following a pressure transient, the building continues to release gasses to the environment at a rate determined by its leakage characteristics. Currently, the building is designed not to exceed one volume per day at 1 psid internal pressure.

The radionuclides released upon a failure of the reactor coolant pressure boundary may be accompanied by a significant release of mass and energy. The MHTGR design must also accommodate the potential failure of a secondary coolant (main steam or feedwater) line, which has no direct radionuclide release consequences but which may impose significant structural or thermal loads on the reactor building and the Reactor Cavity Cooling System (RCCS). As gasses flow through the building, they induce subcompartment pressure transients which must be accommodated in the structural design of the building. The RCCS is part of the building boundary. In order to perform its decay heat removal function, it must also withstand the loadings induced by the blowdown of primary or secondary coolant. To determine the short-term pressure transient experienced by subcompartments within the reactor building, the building is modeled as a group of discrete volumes connected by vent paths. The size of the compartments and flow characteristics of the vent paths are based on the building design. The thermodynamic and hydrodynamic properties of the fluid released into the building are used to calculate the mass and energy release as a function of time. Using small discrete time intervals, the equilibrium conditions in and flow between compartments are alternately calculated. Small time steps and careful modeling result in a good approximation of dynamic behavior. This analytical process is described more fully in the Containment Study for the MHTGR (Ref. 4). The transient pressures are strongly affected by the design of the building vent and its setpoint pressure. In this report, the model is based on the design of the 450 MW(t) MHTGR and is adjusted to accommodate the effect various proposed reactor building features have on the analysis.

The transport of radionuclides from the exterior of the RCPB through the reactor building to the site boundary is evaluated using Regulatory Guide 1.4 models for atmospheric dispersion. Time average values for χ/Q were developed which reflect the long duration of the discharge period for SRDC-6 and SRDC-11. Radiation dose calculations are based on dose conversion factors taken from Regulatory Guide 1.109. The event evaluation process is described in detail in both the PSID (Ref. 1) and in the Containment Study for the MHTGR (Ref. 4).

4. RESPONSE OF THE 450 MW(t) MHTGR DESIGN TO KEY EVENTS

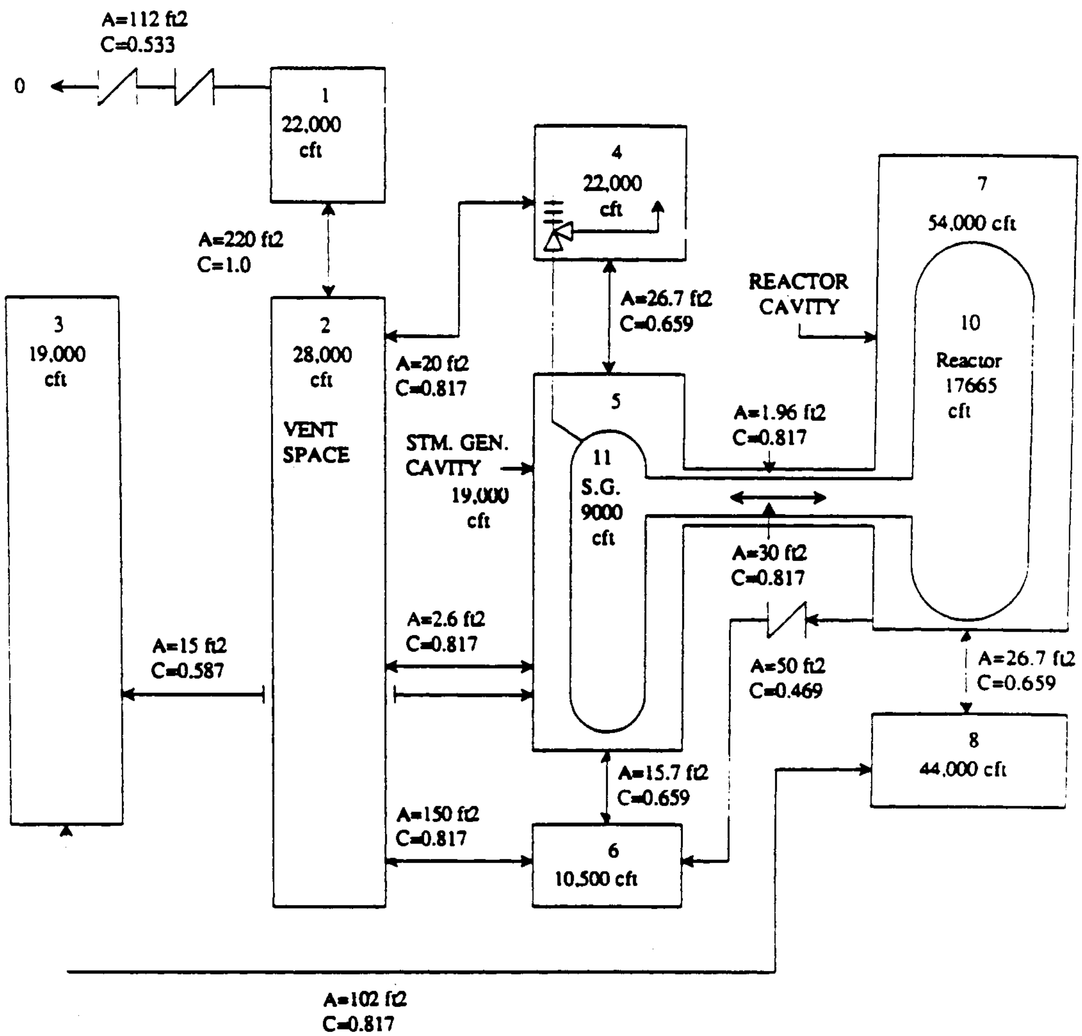
Using probabilistic risk assessment methods, a spectrum of events which can challenge the operation and safety functions of the MHTGR. The key events which challenge the plants ability to protect the offsite public from accidental radioactivity releases are described in Section 3.2. These events are accompanied by the release of mechanical energy in the form of expanding helium coolant. Restricting the release of helium coolant requires features which will also restrict the release of radiologically benign energy stored in the secondary system. In addition to evaluating the transport of radionuclides from the point of release to the point of exposure, the forces and loads exerted on the building by the expanding gas must be evaluated. The vented strategy taken in the MHTGR eliminates much of this concern, but it is still necessary to examine pressure transients.

4.1 PRESSURE TRANSIENT ANALYSIS

Based on previous analyses, it is clear that the most severe building pressure transient would occur following a complete failure (double-ended guillotine break) of the main steam line. This event has been modeled and analyzed for the proposed design. In this analysis, the building is modeled as a set of volumes or nodes connected by flow pathways. Blowdown mass and energy per unit time is input into the volume representing the break location. The code calculates equilibrium conditions in each node at the beginning of one time step and then calculates the flow between nodes for the time step. This process is then repeated for the next time step. By selecting very small time steps, a good approximation of the dynamic process can be made. Figure 4-1 shows the pressure transient model used in this analysis, including the important parameters for the 450 MW(t) reactor building and the main steam line break (MSLB). The peak internal pressure in each node is shown in Table 4-1. Curves showing the pressure transient history for key nodes are shown in Figure 4-2.

4.2 RADIOLOGICAL RESPONSE TO A SMALL HELIUM LEAK WITH DEPRESSURIZED CONDUCTION COOLDOWN (SRDC-11)

If a small helium break occurs coincident with a loss of all forced cooling, the source term described in section 3 for SRDC-11 is released into the reactor building. To give an indication of the relative magnitude of this release, Table 3-5 shows the quantity of I-131, the dose significant radionuclide, which will evolve from the reactor vessel. Using the complete set of 103 radiologically important isotopes, the fission product transport through the reactor building is calculated using a series of building volumes or nodes connected by flow paths. Rates of flow between nodes are calculated separately and input to the program. During each time interval, isotope-specific mechanisms such as birth, decay, settling, condensation, and plateout can act to affect the



Node Identification

1 Vent Path Space above Steam & Feed Piping
2 Vent Path Space from -155 ft to -15 ft
3 Equipment Shaft Space
4 Space above Main Circulator
5 Steam Generator Cavity
6 Space below Steam Generator, around Feed Nozzle
7 Reactor Cavity
8 Shutdown Cooling System Maintenance Space, below Reactor Cavity

A = Flow Path Area, ft^2
C = Flow Coefficient, dimensionless

Figure 4-1 PRESSURE TRANSIENT ANALYTICAL MODEL

Table 4-1
PRESSURE TRANSIENT PEAKS - 450 MW(t) MAIN STEAM LINE BREAK
REACTOR BUILDING VENT SETPOINT AT 1 PSID

Node Description	Peak Pressure (psig)	Time After Initiation (seconds)
Reactor cavity	0.9	2.63
Steam generator cavity	7.2	0.74
Feedwater nozzle area (below steam generator)	8.9	0.44
Circulator area (above steam generator)	7.6	0.70
Equipment access shaft	2.7	1.45
Lower vent passage (containing main steam and feedwater lines)	8.9	0.44
Upper vent passage (leading to vent opening)	8.7	0.44

distribution and quantity of radionuclides as a function of both position within the model and of time. In a second operation, the transport of radionuclides is calculated from the point of release to the point of exposure, including the mechanisms which can affect the quantity and concentration of radionuclides, and the exposure. The important parameters in the analyses include the physical properties of the building (volume, surface area), the flow rates within the model, and the coefficients for time-dependent depletion mechanisms such as plateout. The model is depicted schematically in Figure 4-3. The parameters used in conversion of releases to radiation dose are listed in Table 4-2.

The calculated radiation exposure to an individual at the site boundary for the proposed 450 MW(t) MHTGR are shown in Table 4-3. These results are based on a 1 psid building vent setpoint and the specified 1 volume per day leak rate. This leak rate is approximately three times greater than the calculated leakage at 1 psid, hence is considered an upper bound. In the analysis of SRDC-11, the reactor building leak rate is not significant because the majority of the release leaves the building through the vent rather than via building leaks, and the mechanisms which cause source term reduction (deposition and decay) are nearly the same whether the gas flows through the vent path or through leak points.

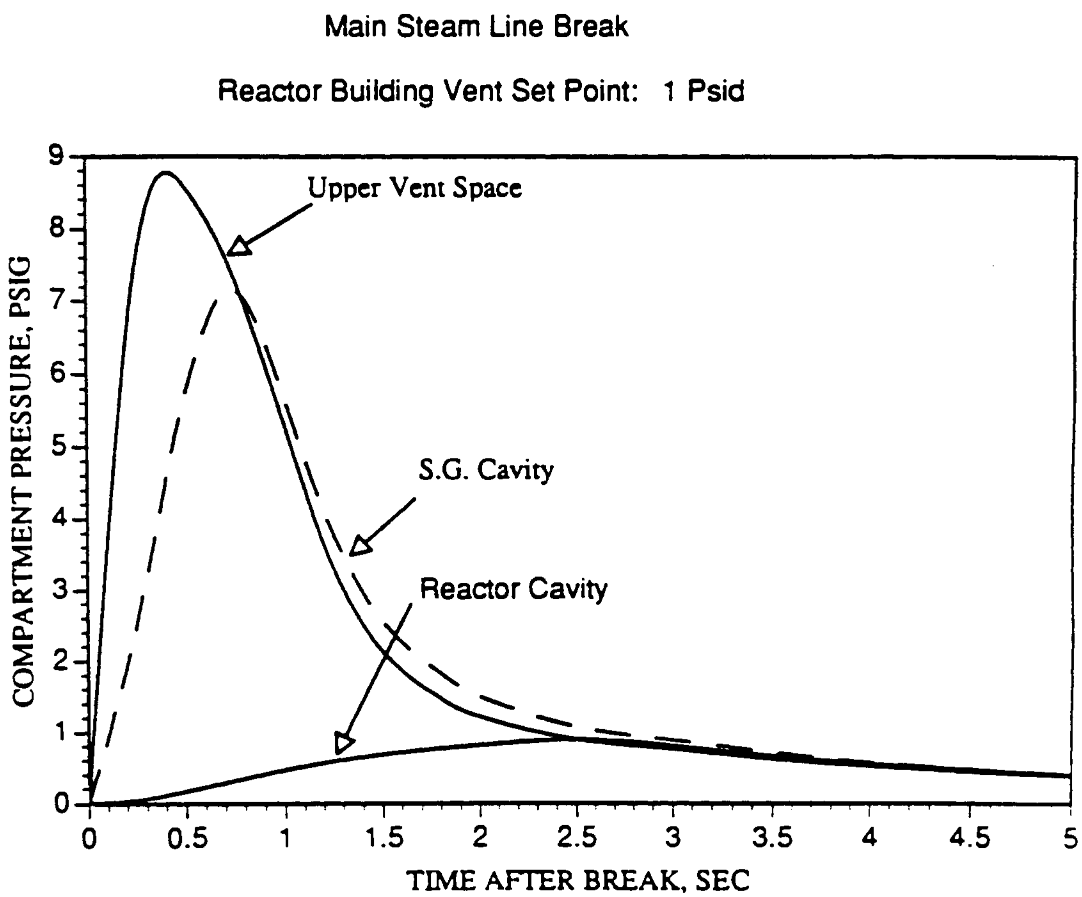
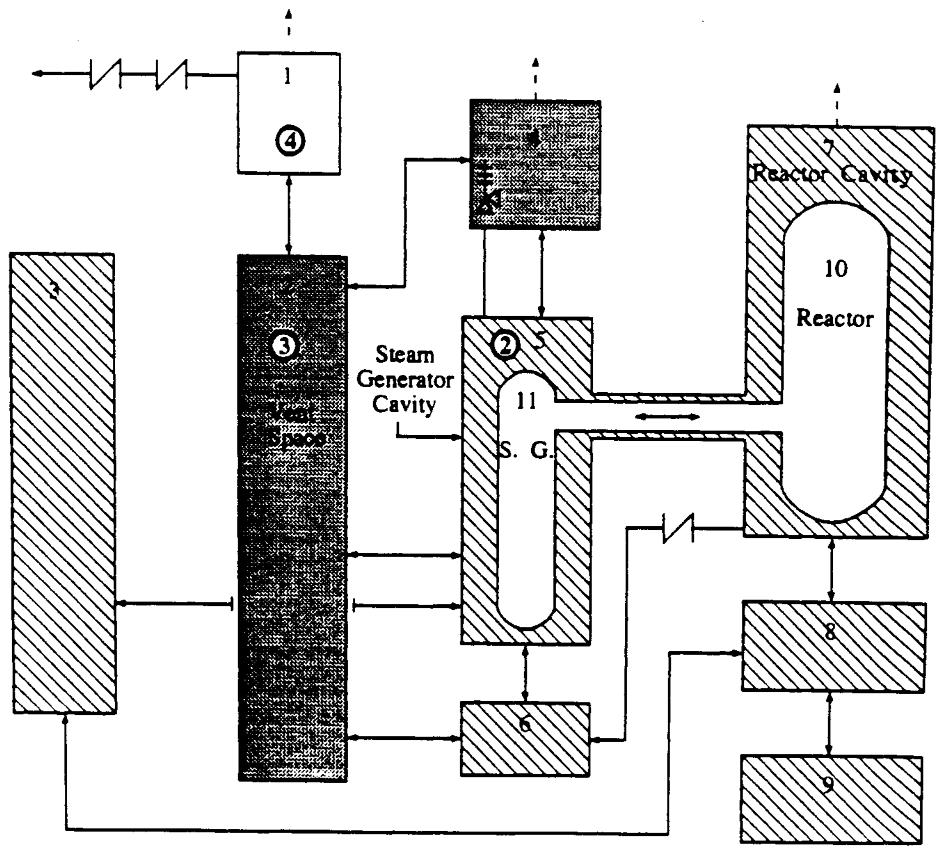


Figure 4-2 PRESSURE TRANSIENT ANALYTICAL RESULTS



2 = LOCADOSE. NE319 Node Number (Node 1 = Environment)

KEY TO FIGURE 4-3

Node Identification	Volume (cu ft)	Surface (sq ft)	Leakage (cfm)
1 Vent Path Space	22,000	6,000	32.15
2 Vent Path Space	28,000	9,500	--
3 Equipment Shaft	19,000	6,600	97.08
4 Main Circulator	22,000	5,100	10.63
5 Steam Generator Cavity	19,000	14,650	--
6 Below Steam Generator	10,500	3,000	--
7 Reactor Cavity	54,000	64,000	29.24
8 SCS Maintenance Space	44,000	9,165	--
Total	218,500	118,015	169.10

Removal Mechanism Constants (Units of hr^{-1})	Steam Condens.	Particulate Settling	Plateout Deposition
1 Vent Path Space	16.99	0.16	0.53
2 Vent Path Space	21.13	0.20	0.66
3 Equipment Shaft	21.63	0.20	0.68
4 Main Circulator	14.44	0.14	0.45
5 Steam Generator Cavity	48.02	0.45	1.50
6 Below Steam Generator	17.79	0.17	0.56
7 Reactor Cavity	73.81	0.69	2.31
8 SCS Maintenance Space	12.97	0.17	0.55

Figure 4-3 REACTOR BUILDING FISSION PRODUCT TRANSPORT MODEL

Table 4-2
WEATHER MODEL AND DOSE CONVERSION PARAMETERS

Parameter	Conservative Value	Best Estimate Value
Atmospheric diffusion, 0 - 8 hr (s/m^3)	1.22 x 10^{-3}	1.22 x 10^{-4}
Atmospheric diffusion, 8 hr - 30 days (s/m^3)	2.7 x 10^{-4}	2.7 x 10^{-5}
Breathing rate, 0 - 8 hr (m^3/s)	3.47 x 10^{-4}	3.47 x 10^{-4}
Breathing rate, 8 - 24 hr (m^3/s)	1.75 x 10^{-4}	1.75 x 10^{-4}
Breathing rate, 1 - 30 days (m^3/s)	2.32 x 10^{-4}	2.32 x 10^{-4}
Dose conversion factors	From Reg. Guide 1.109	From Reg. Guide 1.109

Table 4-3
SITE BOUNDARY DOSE CONSEQUENCES OF PRIMARY COOLANT LEAK EVENTS
(REPRESENTED BY SRDC-11; 0 - 30 DAYS)

	50th Percentile
Thyroid, Rem	0.142
Whole body, Rem	4.91 x 10^{-4}

4.3 RADIOLOGICAL RESPONSE TO A STEAM GENERATOR TUBE FAILURE (SRDC-6)

If an event occurs involving the failure of a steam generator tube accompanied by a loss of all forced cooling, a delay in secondary coolant isolation, and a failure to dump the steam generator, the source term associated with SRDC-6 will be discharged from the reactor coolant pressure boundary via the helium relief valve trains at the top of the steam generator compartment. This scenario is developed using probabilistic arguments and assumes that one relief valve train opens and reseats twice. The third time the valve opens, it is assumed to fail to reseat. The valve discharges into the reactor building, and each time that the valves open, the building pressurization is sufficient to cause the building vent to open. The flow rate of gas through the building is high enough that the attenuation due to plateout on building surfaces or to decay is minor. The results of this analysis are shown in Table 4-4.

Table 4-4
SITE BOUNDARY DOSE CONSEQUENCES OF SRDC-6

	50th Percentile
Thyroid, Rem	1.51
Whole body, Rem	1.08×10^{-2}

4.4 CUMULATIVE RISK OF RADIATION EXPOSURE

The results in Tables 4-3 and 4-4 clearly show that the reference 450 MW(t) MHTGR meets its design goals with regard to lower threshold PAG limits for individual events. The cumulative risk for all events for whole body dose is met. However, the MHTGR does not meet the User/Utility requirement that the cumulative thyroid dose be less than the PAG requirement for all events at event frequencies of 5×10^{-7}/plant year and higher. As shown on Figure 4-4, the primary coolant leak without forced cooling family of events, represented by SRDC-11, is the dominant contributor to this risk. There are several options for dealing with this shortcoming, including more detailed analyses, additional technology, or alterative design features. These alternative features are discussed in Section 6.

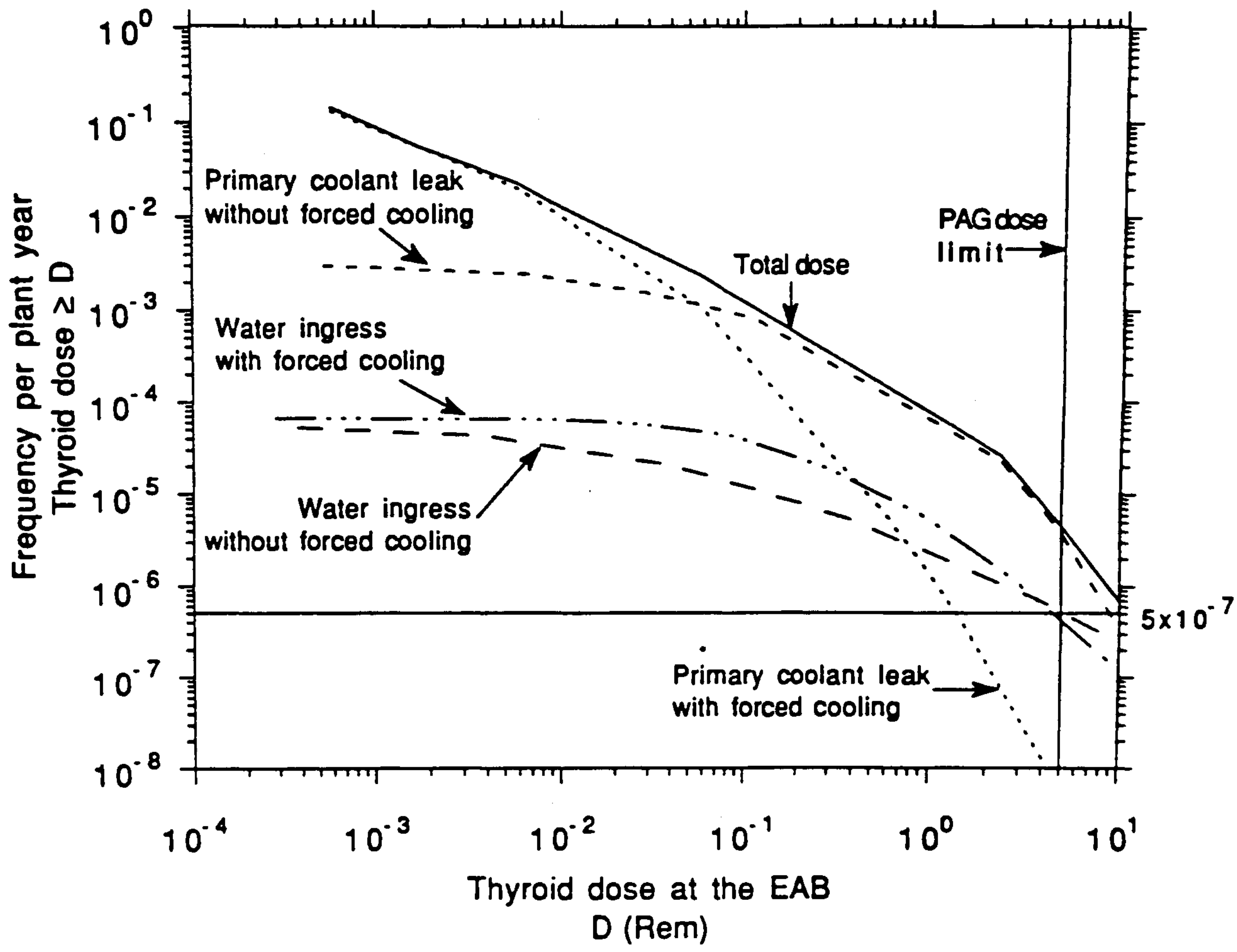


Figure 4-4 450 MW(t) CUMULATIVE DOSE RESULTS

5. SOURCE TERM ALTERNATIVES

In the NRC's Office of Nuclear Regulatory Research (RES) review of the Preliminary Safety Information Document (PSID, Ref.1) it was suggested that more conservative magnitudes be used for both "prompt and delayed fission product source terms." The NRC also questioned two aspects of the fuel and fission product behavior model (Ref. 5). The first aspect involves the rate of hydrolysis of exposed kernels. In the event of a water ingress, the uranium oxide and uranium carbide both react with moisture, releasing fission products. The NRC reviewers felt that the MHTGR should be evaluated, assuming that the rate of this reaction was much higher than calculated from reference MHTGR methods. The second area that the NRC reviewers felt should be evaluated was based on the potential impact of a breakdown in the manufacturing quality assurance of the fuel. It was suggested that the plant could experience a batch of weak fuel, containing an unexpected and undetected flaw. This flaw would manifest itself only after a period of operation and would result in the failure of some portion of intact, but flawed fuel particles.

In this section, the source terms associated with these NRC concerns are addressed. Source terms are assessed that correspond to lower fuel quality than the reference design, a bounding source term due to "rapid hydrolysis" is developed, and an arbitrary but large source term due to hypothesized weak fuel is described. In addition, a bounding prompt source term, related to excessive liftoff/washoff of the plateout radionuclide inventory from the inside surfaces of the RCPB are developed.

5.1 BASIS FOR BOUNDING LOWER LIMIT ON FUEL QUALITY

The reference MHTGR fuel has as-manufactured defects which are expected to result in a bare kernel fraction of nominally 5×10^{-5} after irradiation. To explore the capabilities of alternative MHTGR Containment System features to mitigate larger source terms, a lower fuel quality value has been explored. Although there is no intent to lessen the importance of achieving the quality specified for the reference fuel, it is appropriate to envelope the potential lower limit of as-manufactured fuel quality. Based on the manufacturing similarities between the MHTGR fuel design and the fuel used in the Fort St. Vrain (FSV) HTGR, it is reasonable to expect that the MHTGR fuel would not be lower in quality than the fuel manufactured for FSV. FSV fuel blocks are the same physical size and configuration and use the same approach of inserting fuel compacts into blind holes alternating with helium coolant holes. The last reload batch of fuel was manufactured for FSV to a specification which allowed a defect fraction which was approximately 10 times higher than the reference MHTGR specification. Operating experience for FSV fuel indicated that it had no hidden defects or unanticipated weaknesses over the range of its in-service operation. FSV-like fuel quality is proposed as a bounding fuel quality for evaluating alternative features. Table 5-1 shows the I-131 inventories which would exist in the 450

MW(t) MHTGR if this lower fuel specification were used. The time associated with accident releases with this fuel quality would be similar to the data indicated in Figures 3-2 and 3-3.

Table 5-1
I-131 INVENTORY RELEASED BY LIMITING EVENTS - 450 MW(t) MHTGR
LOWER BOUND FUEL QUALITY ASSUMED
(Nominal Fuel Temperatures)

Location	Inventory (Ci)	Timing of Release	Release Events			
			Small Helium Leak With Depressurized Cooldown (SRDC-11)		Water Ingress With Depressurized Cooldown (SRDC-6)	
			Release Mechanism	Curies I-131	Release Mechanism	Curies I-131
A. Circulating	0.3	Minutes	Helium Blowdown	0.3	Helium blowdown	0.3
B. Plateout	260	Minutes	Liftoff	0.4	Steam washoff	154
C. Defective fuel						
1. Missing coatings	5570	Hours-days				
2. Heavy metal contamination	1114	Hours-days	Hydrolysis	None	Hydrolysis	1996
			Heatup	858	heatup	976
Total	6684		Total	858	Total	2972
D. Standard particles	1.2×10^7	> days	None	None	None	None
Total release from vessel				471		2863

5.2 BASIS FOR BOUNDING SOURCE TERM ASSUMING RAPID FUEL KERNEL HYDROLYSIS

If the chemical reaction between water and uranium oxide and uranium carbide occurs at a rate which is much higher than is characterized in the reference MHTGR fuel behavior model, the source term associated with families of events represented by SRDC-6 could be larger and evolve more quickly. The worst case for the source term associated with this fuel behavior assumption would involve the release of all of the fission product inventory available from exposed kernels and heavy metal contamination to the helium reactor coolant during the first few minutes following a water ingress. Radionuclides would then be present in the coolant during each lifting of the helium relief valve. Table 5-2 gives an estimated magnitude of the I-131 releases which would occur if rapid hydrolysis is assumed for the reference and lower bound fuel quality. The results presented in this table are based on the assumption that all defective fuel hydrolyses and that the release from the vessel via the relief valves is the same percentage of fission products released to the coolant and from the vessel as used in the detailed analyses of SRDC-6 (Table 3-5). Figure 5-1 gives the estimated time of the release.

Table 5-2
I-131 RELEASED BY LIMITING EVENTS - 450 MW(t) MHTGR
RAPID FUEL HYDROLYSIS ASSUMED
(Nominal Fuel Temperature)

Location	Timing of Release	Water Ingress with DCC (SRDC-6)			
		Reference Fuel Quality		Lower Bound Fuel Quality	
		Release mechanism	Curies I-131	Release mechanism	Curies I-131
A. Circulating	Minutes	Helium blowdown	0.03	Helium blowdown	0.3
B. Plateout	Minutes	Steam washoff	15.4	Steam washoff	154
C. Defective Fuel 1. Missing coatings 2. Heavy metal contamination Total	 Minutes Minutes	Hydrolysis heatup Total	600 72 672	Hydrolysis heatup Total	5570 976 6546
D. Standard particles	> days	None	None	None	None
Total release from vessel			637		6305

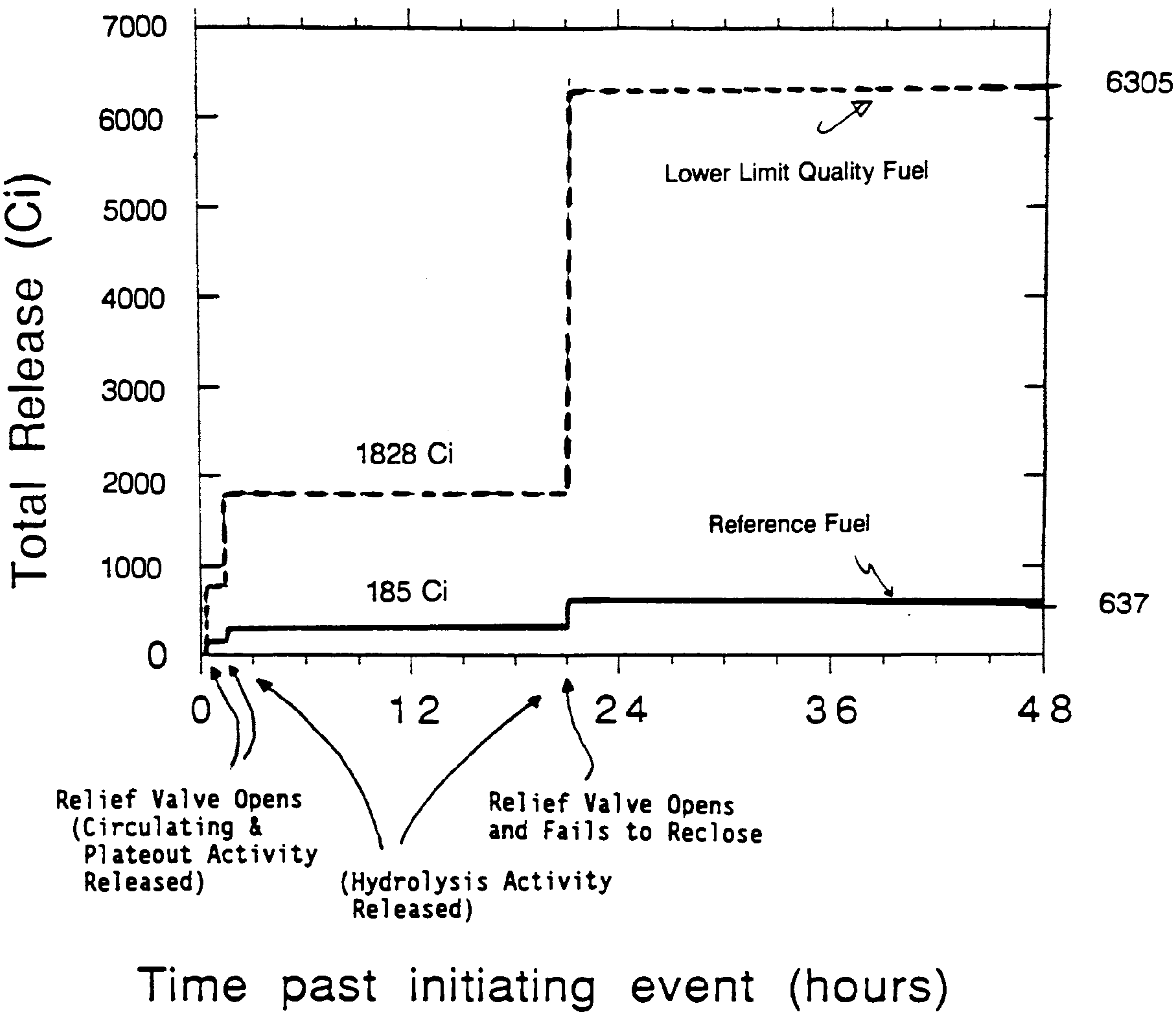


Figure 5-1 RELEASE AS A FUNCTION OF TIME, ASSUMING RAPID HYDROLYSIS (SRDC-6)

5.3 BASIS FOR ARBITRARY SOURCE TERM ASSUMING WEAK FUEL

A set of arbitrary assumptions regarding the weak fuel concern has been made: Half of a core reload contains weak fuel at the time a loss of cooling event like SRDC-11 occurs, and weak fuel (which is undetected during normal operation) fails above 1200°C. Above its failure temperature, the TRISO coating on weak fuel fails, and the fuel behavior is like a bare or defective kernel. Reference models for bare kernels are then applied except that it was conservatively assumed that fission products would be released rapidly, in a manner similar to releases from heavy metal contamination.

SRDC-11 is based on a 0.05 $in.^2$ break, and takes approximately 40 hr to exhaust to atmospheric pressure. Flow out of the vessel decreases significantly after that time, although core temperatures continue to increase until about 80 hr after the start of the event. The peak fuel temperatures distributed within the core during an event like SRDC-11 will be lower than those experienced during a depressurized cooldown. If it is assumed that the peak core temperatures are reached at 40 hours, instead of 80 hours, and a linear interpolation between the pressurized and depressurized cases is made, approximately 25 percent of the fuel blocks will reach a temperature exceeding 1200°C. If the reload is half the core, and half the reload contains the weakness, then around six percent of the core will both be weak and reach 1200°C. During heatup events, two percent of the inventory in defective fuel is released. Using this value as a guideline, the total I-131 which would be released into the coolant during an event like SRDC-11, asuming weak fuel, is 15,000 Curies. Dose consequences from weak fuel are estimated by scaling the dose from detailed calculations of SRDC-11, using a ratio based on the release of I-131.

5.4 BASIS FOR BOUNDING PROMPT SOURCE TERM ASSUMING ALL PLATEOUT RELEASE

The MHTGR prompt source term is comprised of both circulating and plateout inventories. Circulating inventories are characterized as fission products that are entrained in the flowing helium coolant and could be released from the reactor vessel in the event helium coolant escapes. The plateout inventories are fission products that are located on the inside surfaces of the reactor coolant pressure boundary (RCPB). A break in the RCPB large enough to cause a rapid blowdown of helium could result in the re-entrainment of a fraction of these fission products and their release from the RCPB. Plateout activity could also be washed off and made available for release by the action of steam or water during an event involving a steam generator tube failure.

The magnitude of the prompt source term is a result of power operation which generates circulating and plateout radionuclide levels. The circulating and plateout inventories available for release in a helium depressurization which comprise the prompt source term will be limited by Technical Specifications imposed on reactor operations. The amount of radionuclides actually released from the reactor vessel will depend on the inventories and liftoff/steam washoff parameters such as shear force, species adherence characteristics and amount of water ingress.

The bounding prompt source term is developed from an arbitrary assumption that all the plateout activity is released either by liftoff during helium leaks or by washoff during water ingress events.

In addition to SRDC-6 and SRDC-11, an additional rapid depressurization event, represented by SRDC-10, is analyzed. Past studies using reference methods have shown that the prompt releases from the SRDC-10 accident class are the most severe. SRDC-10 is a rapid helium depressurization characterized by moderate openings of the RCPB of up to 13 square inches. A break of this size results in a rapid blowdown (blowdown complete within minutes) followed by a conduction cooldown with relatively slow transport of gases from the vessel due only to thermal expansion of the coolant as the core heats up and thermal contraction as the core cools down. It has the largest and most rapid release of the prompt source term. Moderate break events release a much smaller portion of the delayed source term than small leaks (SRDC-11) because the helium coolant is essentially all exhausted before the core has heated significantly, and released the delayed source term into the vessel.

Table 5-3 shows the I-131 inventory which would be released during key events from the 450 MW(t) MHTGR reactor coolant pressure boundary if the bounding prompt source term were used. When compared to the I-131 inventory released during SRDC-11 and SRDC-6, as shown in Table 3-5, the only changes are the increased liftoff or steam washoff of the plateout. Note that a smaller fraction of the source term is released from the RCPB in SRDC-10 as compared to SRDC-11 due to the lack of a driving force from the vessel system during the conduction cooldown.

5.5 SUMMARY OF ALTERNATIVE SOURCE TERMS

The release of radionuclides from the MHTGR coolant pressure boundary is determined by the accident scenario, which is critical to the transport mechanism across the RCPB; the in-situ radionuclide inventories, which are determined primarily by the fuel quality or defect fraction; and the fuel and fission product behavior model, including the liftoff/washoff fraction and the rate of hydrolysis for water ingress events. Releases have been calculated for three event families and for the spectrum of source term alternatives including the arbitrary weak fuel scenario. Table 5-3 summarizes the I-131 releases over the range of event and source term alternatives analyzed.

Table 5-3
I-131 INVENTORY RELEASED BY LIMITING EVENTS - 450 MWt MHTGR
REFERENCE FUEL QUALITY AND BOUNDING PROMPT SOURCE TERM ASSUMED
(Nominal Fuel Temperatures)

Location	I-131 Inv. (Ci)	Timing of Release	Release Events					
			Rapid Depressurization and Loss of Forced Cooling (SRDC-10)		Slow Depressurization and loss of Forced Cooling(SRDC-11)		Water Ingress with Depressurized loss of Forced Cooling (SRDC-6)	
			Release Mechanism	I-131 Rel. (Ci)	Release Mechanism	I-131 Rel. (Ci)	Release Mechanism	I-131 Rel. (Ci)
A. Circulating	0.03	Minutes	Helium Blowdown	0.03	Helium Blowdown	0.03	Helium Blowdown	0.03
B. Plateout[(a)]	26	Minutes	Liftoff	26	Liftoff	26	Steam Washoff	26
C. Defective Fuel 1. Missing Coatings 2. Heavy Metal Contamination Total	 600 120 720	 min-hr hr-days	Heatup	87	Heatup	87	Hydro-lysis Heatup	219 72
D. Standard Particles	1.2E7			0		0		0
Total Release From the Core				87		87		291
Total Release From the RCPB				37		73		276

Note: (a) The best estimate liftoff of I-131 plateout during SRDC-10 and SRDC-11 are 0.05 and 0.04 Curies, respectively. The best estimate steam washoff of I-131 plateout during SRDC-6 is 15.4 Curies.

Table 5-4
ALTERNATIVE SOURCE TERMS
Curies of I-131 Released from the Reactor Coolant Pressure Boundary

Alternative Fuel Scenario	SRDC-6	SRDC-10	SRDC-11
Reference Fuel Quality with Expected Liftoff/Washoff	266	12	48
Reference Fuel Quality with 100% Liftoff/Washoff	276	37	73
Lower Limit Fuel Quality with Expected Liftoff/Washoff	2,863	~120	471
Lower Limit Fuel Quality with 100% Liftoff/Washoff	~2,965	~370	~720
Reference Fuel Quality with Rapid Hydrolysis	637	N/A	N/A
Lower Limit Fuel Quality with Rapid Hydrolysis	6,305	N/A	N/A
"Weak Fuel" arbitrary assumptions	N/A	N/A	15,000

6. ALTERNATIVE MHTGR OFFSITE DOSE REDUCTION STRATEGIES

The focus of this report is on MHTGR containment barriers 1 and 4. Other strategies that would affect the frequencies of events or which focus on barrier 3, the RCPB, have not been addressed. An example of modifications to the reference design which are outside the scope of this study is a more reliable and automatic means of intentional depressurization.

The alternative design strategies which have been identified for further evaluation are listed in Table 6-1. Providing a tall vent stack (alternative 1) to which all releases would be directed would have no effect on the event scenarios nor on the performance of any of the Containment System barriers. This alternative would act entirely by increasing the atmospheric dispersion, thereby reducing the concentration of radionuclides in the air at the point of exposure. Event scenarios and containment barrier performance are also unchanged by increasing the site Exclusion Area Boundary (EAB) distance (alternative 2). This strategy acts entirely by providing additional distance and, therefore, time over which atmospheric dispersion, decay, and deposition will reduce the concentration of radionuclides in the air at the point of exposure.

The other four alternatives listed in Table 6-1 act primarily by changing the performance of the barriers in the containment system. The introduction of a piping network and a filter downstream of the primary helium relief valve trains (alternative 3) which discharges outside of the reactor building boundary would create a new fission product retention system, distinct from and operating in parallel with the reactor building. This alternative clearly has potential merit. This option will act exclusively on events which release fission products through the helium relief train. SRDC-6 is the surrogate for the family of events in this category. This alternative is described more fully below and is evaluated using reference fuel quality in Section 7, and with alternative source term assumptions in Section 8.

The establishment of an alternative filtered flow path for gasses which would otherwise pass into the environment via the reactor building vents or leaks may also have potential merit. The introduction of a filtered pathway influences the way the building vent responds. It is possible to devise a filter suitable for even the largest breaks, including the limiting main steam line. A very large filter could allow the elimination of the building vent valves. However, the very large size of a filter capable of receiving the flow from a main steam line failure or even the largest credible helium break could make it difficult to arrange and very costly. Furthermore, the events which result in the most difficult flow conditions are not the most challenging in terms of fission product release. If the filtered path is parallel to the main vent path, the pressure loss as the gasses flow through a filter bed must be less than the building vent setpoint pressure, otherwise the vents will open and

Table 6-1
ALTERNATIVE DESIGN STRATEGIES FOR REDUCED OFFSITE RADIATION EXPOSURE

Alt	1	2	3	4	5	6
	Increase radio-nuclide release point elevation	Increase EAB	Provide He relief valve blowdown filter	Provide reactor building alt. flow path with filter/scrubber	Control reactor building leak rate	Control reactor building vent relief pressure setpoint
Base case	Ground level release	EAB = LPZ = 425 m.	Decontamination factor = 1 (no filter)	Decontamination factor = 1 (no filter or scrubber)	Leak rate = 1 vol/d; flow alts: vent or leaks	Two trains with two valves in series open at 1 psid
Option 1	Provide tall stacks = 90 m (300 ft)	EAB = LPZ = 800 m (1/2 mile)	DF=1 (noble gas) DF=10 (halo-gens) DF=10 (particulates)	DF=1 (noble gas) DF=3 (halogens) DF=3 (particulates)	Leak rate = 1 vol/d; flow alts: vent, filter (90%), leaks (10%)	Same Valve arrangement, Valves open at 5 psid
Option 2			DF=1 (noble gas) DF=100(halo-gens) DF=100 (particulates)	DF=1 (noble gas) DF=30 (halogens) DF=30 (particulates)	Leak rate = 1 vol/d; flow alts: vent, filter (99%), leaks (1%)	
Option 3			DF=1 (noble gas) DF=100(halo-gens) DF=1000 (particulates)			

the filter will be bypassed. An increase in the building pressure during transients, achieved by raising the vent opening pressure as high as possible, will facilitate the design of a filter. The head loss through the filter bed can be controlled by adjusting the bed depth and frontal area of the filter. It is possible to select filter parameters such that helium breaks greater than a certain size will result in opening the building vents, while breaks of a smaller size will be relieved through the filter and by building leakage without opening the vents. The building leak rate is also strongly affected by the peak design pressure. At 1 psid, the architectural features needed to establish a 1 volume per day leak rate, such as door or floor plug gaskets and electrical or piping penetrations seals, are typical of conventional power plant construction. If the building pressure is increased without any changes in the architectural details, the leak rate will increase. The gases which escape the reactor building boundary via leakage bypass the filter. In order to establish an effective reactor building filter capability, it may be necessary to specify a reactor building leak rate which is different from the current value of 1 volume per day, and that may lead to additional requirements for architectural features to limit leakage. Containment feature alternatives 4, 5, and 6 are interdependent, and must be examined in a coordinated manner. Building filter design alternatives, including the implications for the vent design and the building leak rate, are described more fully below and are evaluated using reference fuel models in Section 7 and with alternative source term assumptions in Section 8.

6.1 PROVIDE A FILTERED RELEASE PATHWAY FROM THE REACTOR BUILDING

This design option is depicted schematically in Figure 6-1. The building filter capacity, in terms of the size of the blowdown to be accommodated, is a parameter which can be selected, and a set of constraints on the filter and other building aspects (vent setpoint and leakrate) can then be determined. The head loss through the filter must be less than the maximum building vent opening pressure, 5 psid. The head loss is a function of flow, and as the blowdown proceeds, the flow rate and head loss through the filter will gradually decrease until the building returns to atmospheric pressure. The physical and chemical properties of the blowdown gases, particularly the temperature, place limits on the type of filter medium which is suitable. Sand or gravel, and packed glass fibers both meet the need to withstand the expected temperature (800°F) and to be suitably inert over a long standby period.

A candidate filter is specified by a limited set of parameters, specifically the filter medium, bed depth, and frontal area. The performance of the filter will vary with the flow rate, which is determined by the break size. The decontamination factor of the filter increases as the length of the flow path (filter bed depth) increases, and is inversely proportional to the flow velocity through the bed. The pressure loss through the filter is proportional to the flow velocity. For purposes of this report, several candidate filters have been chosen based on judgement. A preliminary set of scoping calculations indicate that a bed consisting of packed glass fibers will usually be more efficient in terms of head loss with depth, but sand may be superior for decontamination factor per unit pressure loss. Table 6-2 indicates candidate filter design parameters using glass fiber as a filter medium. The bed depth and frontal area of candidate filters have been selected and the flow characteristics and decontamination performance calculated for various helium break sizes. If selected for incorporation into the plant design, this filter design must be optimized. The physical location of the vented, low pressure containment (VLPC) filter can be within the building, at or below grade, or outdoors. A likely location is outdoors at grade near the power conditioning modules, west of the maintenance enclosure. Each VLPC would have its own filter arrangement. A low pressure rupture disk at the filter outlet is proposed as a means of excluding moisture and foreign matter during standby conditions expected to persist over the life of the plant.

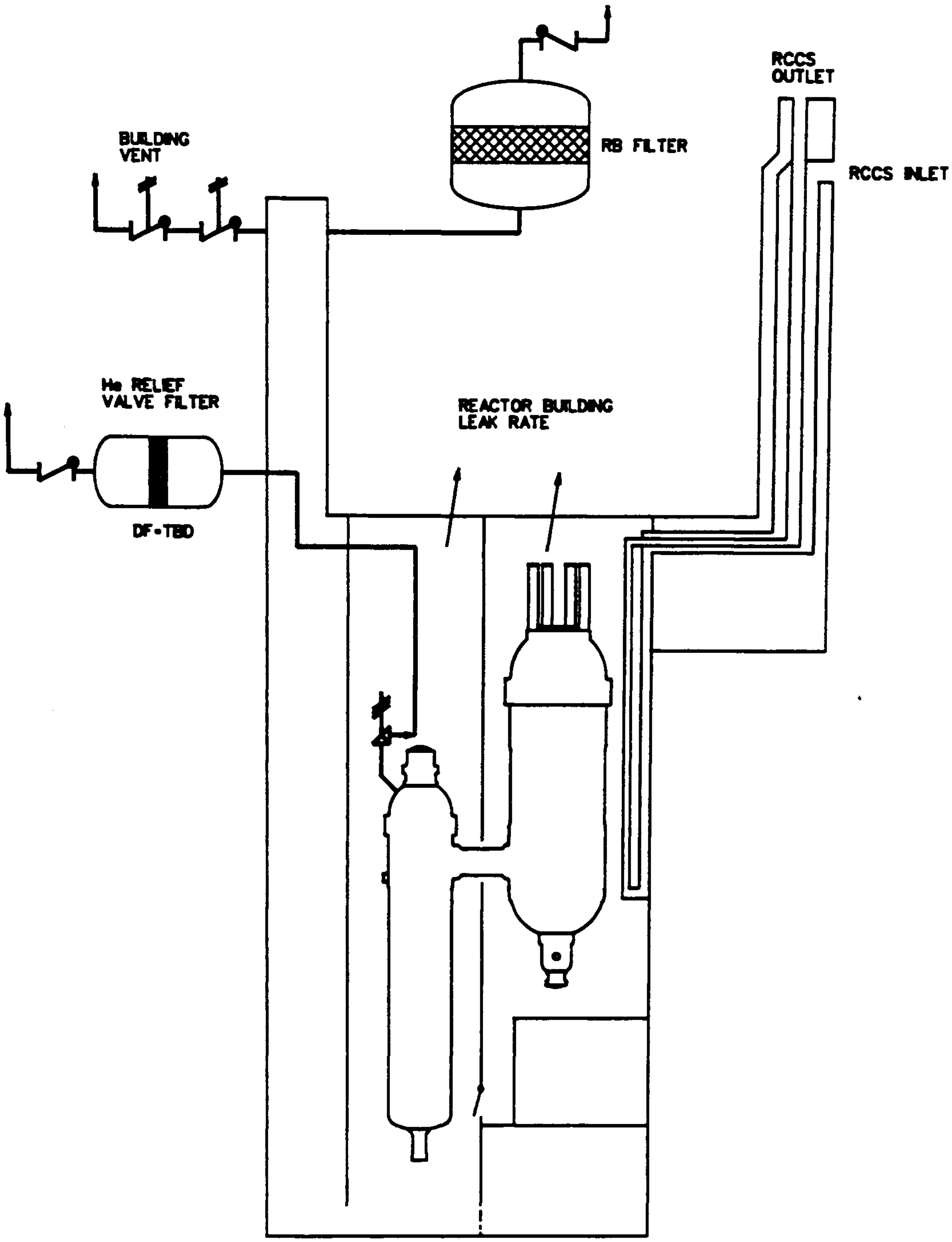


Figure 6-1 VLPC BUILDING FILTER - SCHEMATIC ARRANGEMENT

Table 6-2
CANDIDATE FILTERS, GLASS FIBER MEDIUM

Bed Depth	Bed Diam.	Primary Coolant Leak Size	Max Flow (CFM, 1000's)	Head Loss, ΔP (psi)	Decontamination Factors: Noble Gas	Decontamination Factors: Halogens	Decontamination Factors: Particulates
12 in.	5 ft.	10 in^2	464	> 5	<— Vent opens - filter bypassed —>		
		1 in^2	46.4	0.4	1	~ 1	1.5
		0.05 in^2	2.3	0.02	1	~ 1	3
	15 ft.	10 in^2	464	> 5	<— Vent opens - filter bypassed —>		
		1 in^2	46.4	0.04	1	~ 1	2
		0.05 in^2	2.3	0.002	1	~ 1	20
	45 ft.	10 in^2	464	0.05	1	~ 1	2
		1 in^2	46.4	0.005	1	~ 1	10
		0.05 in^2	2.3	0.0002	1	4.4	1,200
120 in.	5 ft.	10 in^2	464	> 5	<— Vent opens - filter bypassed —>		
		1 in^2	46.4	4	1	~ 1	20
		0.05 in^2	2.3	0.2	1	50	20,000
	15 ft.	10 in^2	464	> 5	<— Vent opens - filter bypassed —>		
		1 in^2	46.4	0.44	1	4.3	1,200
		0.05 in^2	2.3	0.02	1	> 1000	> 10^4
	45 ft.	10 in^2	464	0.5	1	3.5	1,000
		1 in^2	46.4	0.05	1	1000	10^4
		0.05 in^2	2.3	0.002	1	>> 1000	>> 10^4

Leakage across the building boundary to the environment will bypass the filter. The flow through cracks and small openings will vary in the same manner as the filter, as a function of the pressure gradient between the inside and the outside of the building, and the flow coefficient for the opening. Over the range of flow expected in the events which are radiologically important, the flow coefficients for the filter and for the building leaks can be assumed to be constant. Therefore, the percentage of flow which goes through the filter and the percentage of flow which bypasses the filter by leaking through the building boundary are constant regardless of the pressure variation over the entire blowdown event. This percentage can be controlled by selecting the filter characteristics or by altering the architectural features in the reactor building to reduce the leak rate. It is probably easier and more efficient to increase the percentage of flow passing through the filter by adding more filter area than it would be by adding features to reduce the building leak rate.

6.2 PROVIDE A FILTER ON THE HELIUM RELIEF LINE

This design option is depicted schematically in Figures 6-2 and 6-3. The constraints on the design of this alternative include the following:

- The RCPB vessels are all interconnected and are designed to Section III of the ASME Boiler & Pressure Vessel (B&PV) Code. Events which can cause overpressure in the MHTGR vessels can result from an increase in the temperature of the helium or from mass addition which could be either helium or water. The required capacity of the relief valves in the MHTGR is controlled by mass addition due to water ingress, and is based on the failure of a steam generator tube. With design margin, this results in a required capacity of 25 lbm/sec of steam or 42 lbm/sec of an equimolar mixture of carbon monoxide and hydrogen produced when the ingressing steam reacts completely with core graphite. The valves actually selected for the MHTGR may have a greater capacity, which must be accommodated in the design of the filter. The gas flowing to the filter could vary from pure helium to pure steam, as well as the carbon monoxide-hydrogen reaction product mixture.

- The code specifies requirements for redundancy, which could be met with 2 or more valve trains, and for the maximum allowable head loss in the inlet piping to the valve. The code allows outlet piping attached to the valves which can have a head loss of no more than 30 percent of the valve setpoint. Allowing for losses in the piping, the maximum allowable head loss in the filter train can be about 300 psi.

The physical location for the filter is not constrained although it may be necessary to surround it with radiation shielding and to protect it form external hazards. It is desirable to locate its discharge outside of the reactor building and to connect the relief valves to the filter discharge with piping which penetrates the reactor building boundary. There are two advantages to locating the filter discharge outdoors. An exterior discharge

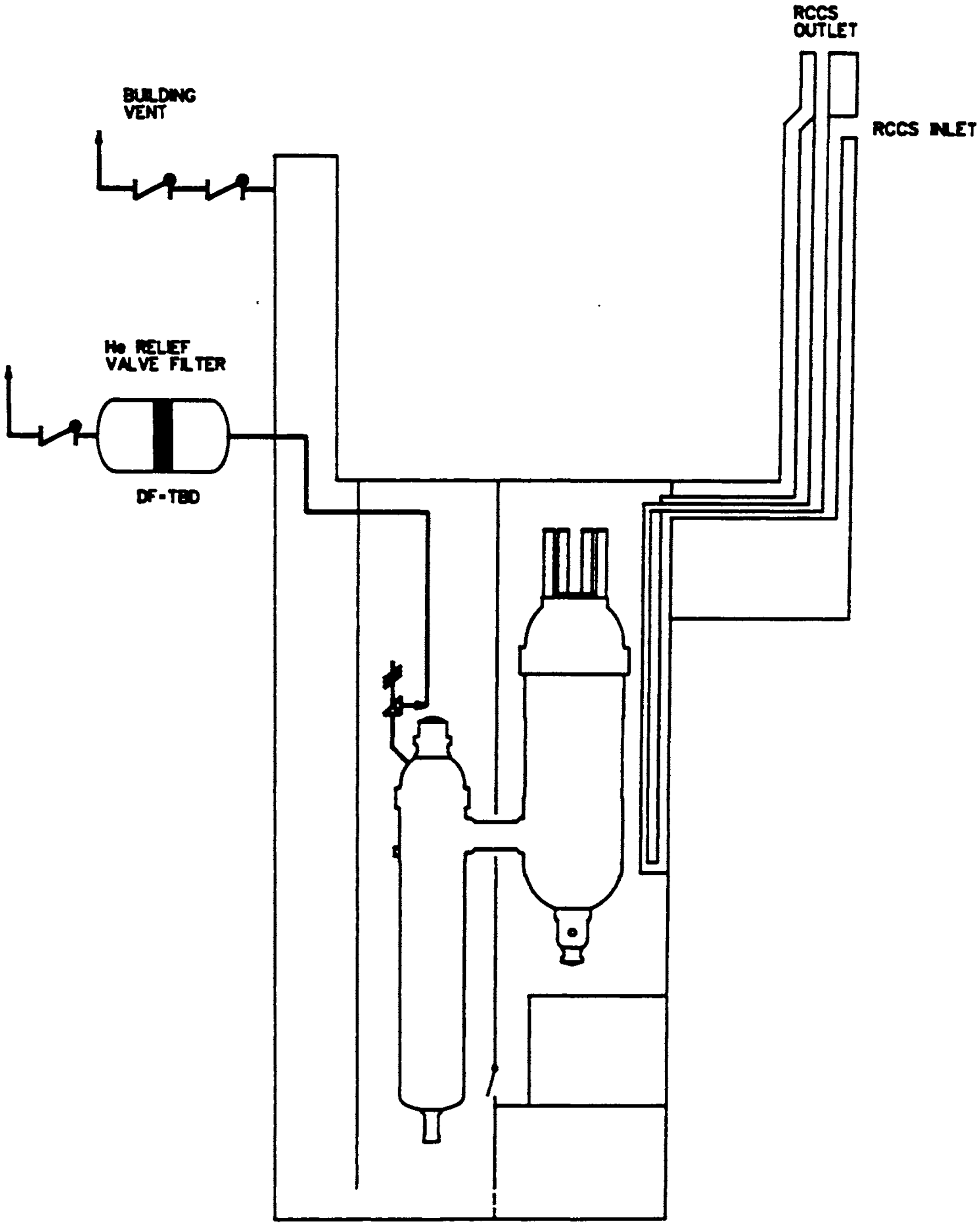


Figure 6-2 HELIUM RELIEF VALVE FILTER TRAIN - SCHEMATIC ARRANGEMENT

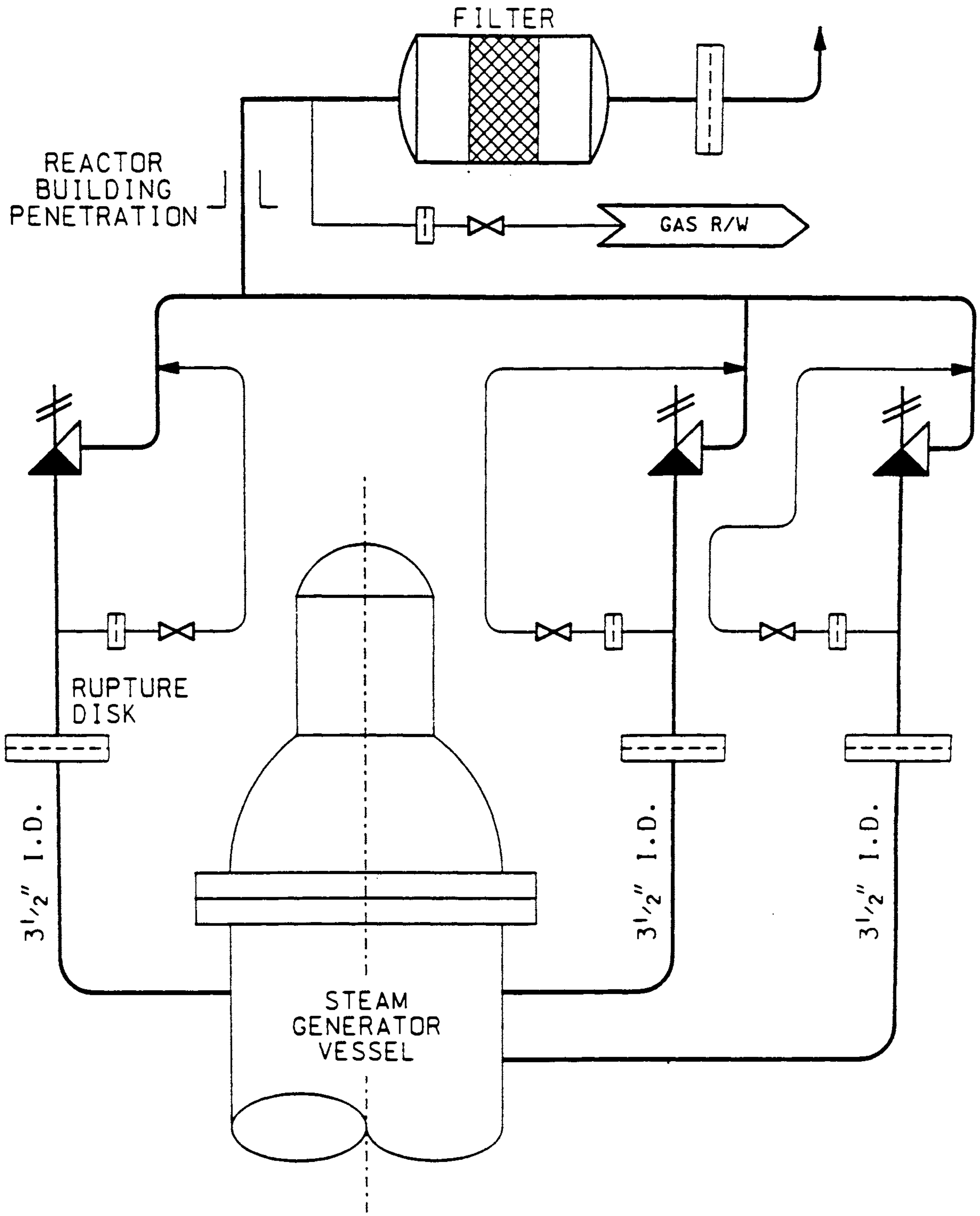


Figure 6-3 HELIUM RELIEF VALVE FILTER TRAIN - DETAIL ARRANGEMENT

will reduce the hazards to personnel who might be suffocated or burned by helium if they were inside the silo portion of the VLPC when the relief valves lifted. It will also preclude contamination of the silo interior and the need for subsequent cleanup. If the filter is effective, the advantage for contamination control is very small.

The piping network which connects the relief valve trains to the filter can be headered from each valve to a common line connecting to the filter. The sizing of the line must be such that it does not exceed the maximum allowable head loss requirements. The present expectation is that the 450 MW(t) MHTGR design will be equipped with 3 identical relief valves, configured as shown in Figure 6-3. The low pressure rupture disk shown on the outlet of the filter vessel is proposed as a means of excluding moisture foreign material from the filter during standby conditions which are expected to persist throughout the life of the plant. Potential parameter sets for the filter itself are shown in Table 6-3.

Table 6-3
HELIUM RELIEF FILTER TRAIN PARAMETERS BASED ON
RELIEF VALVE SIZING CRITERIA
(42 pounds per second maximum mass addition rate)

Filter Medium	Glass Fiber	Glass Fiber	Sand/Gravel	Sand/Gravel	Sand Gravel
Bed depth	35 ft	40 ft	104 in.	92 in.	60 in.
Bed characteristics	11 lb/cu ft.	10 lb/cu ft	Granule size 0.04 in.	Granule size 0.06 in.	Granule size 0.024 in.
Decontamination factors:					
Noble gas	1	1	1	1	1
Halogens	4	100	100	10	1000
Particulates	1,500	4,000	100 +	10 +	1000 +
Frontal area ft^2	19.6	78.5	4200	1400	4200
Head loss during max. blowdown, psi	50 psi	23 psi	122 psi	288 psi	88 psi
Vessel size					
Diameter, ft	5	10	73	43	73
Length, ft	45	45	19	18	15

6.3 CHANGE THE SITE AND BUILDING VENT CONDITIONS

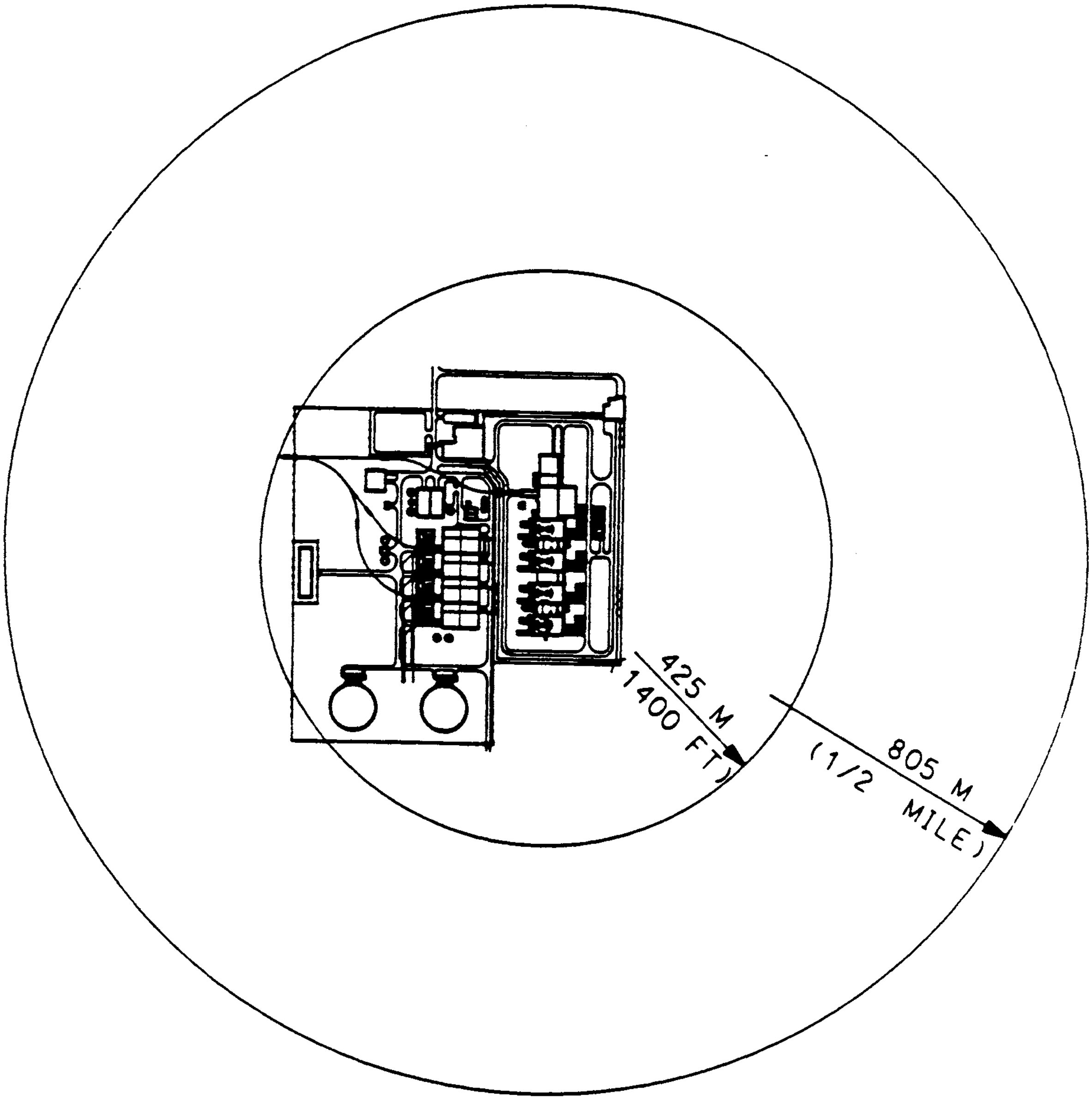
The MHTGR Overall Plant Design Specification (Ref. 12) stipulates that the minimum exclusion radius (EAB) shall be set at 425 m (1394 ft). The passive light water reactors being designed as part of a government-utility effort are being designed to have a minimum distance to the EAB of 805 m (0.5 mile). The MHTGR seeks to fill a role in the power industry which is complementary to the passive LWRs, and it would be reasonable to design the MHTGR to the same generic site requirements. Changing the specified distance to the EAB from 425 m to 805 m would reduce all offsite doses by a factor of approximately three. Figure 6-4 illustrates the effect this would have on the total site area required for the MHTGR, and shows the increase in distance which would be imposed on any process heat or related industrial facility which might be affiliated with the MHTGR.

Another way to change the evaluated offsite doses for all events would be to equip the MHTGR with a tall exhaust stack. This strategy would allow the MHTGR to take credit for an elevated release for all radioactive exhaust streams which could be assured of traveling up the stack. Exhaust streams such as building leakage would not be included in this benefit. It would be feasible to direct the nuclear island HVAC exhaust to the stack, and the VLPC vent could also be directed to the stack. This would allow a lower calculated dose for all important events, since the vent is opened by the discharges to the VLPC in the reference design for all blowdown events including event families characterized by SRDC-6 and SRDC-11. If the VLPC is equipped with a filter (described in Section 6.1, above) and a relief train filter (described in Section 6.2, above) the exhaust from both of these features could be directed to a stack. This configuration is shown on Figure 6-5.

Both of these site related strategies are available. They are both relatively modest in cost and their efficacy can be easily evaluated. Sections 7 and 8 include their effect on calculated offsite doses.

6.4 COMBINATIONS OF ALTERNATIVE CONTAINMENT FEATURES

Clearly, the alternative choices for incremental VLPC features can be combined in many different ways. Each feature has its own cost-effectiveness, as well as its own effectiveness at reducing site boundary doses from individual events and overall radiation dose risk to the offsite public. In Sections 7 and 8, the effectiveness of each feature is evaluated for the appropriate range of source terms. The effects of combinations of features are not evaluated exhaustively, but instead doses are calculated for groups of features chosen on the basis of judgement. Table 6-4 lists the subset of these feature sets which are reported in the executive summary.



X/Q Values (0-8 hrs)	50th%ile	95th%ile
425 Meters	1.22×10^{-4}	1.22×10^{-3}
805 Meters	3.99×10^{-5}	3.99×10^{-4}

Figure 6-4 EXCLUSION AREA BOUNDARY RADIUS OPTIONS

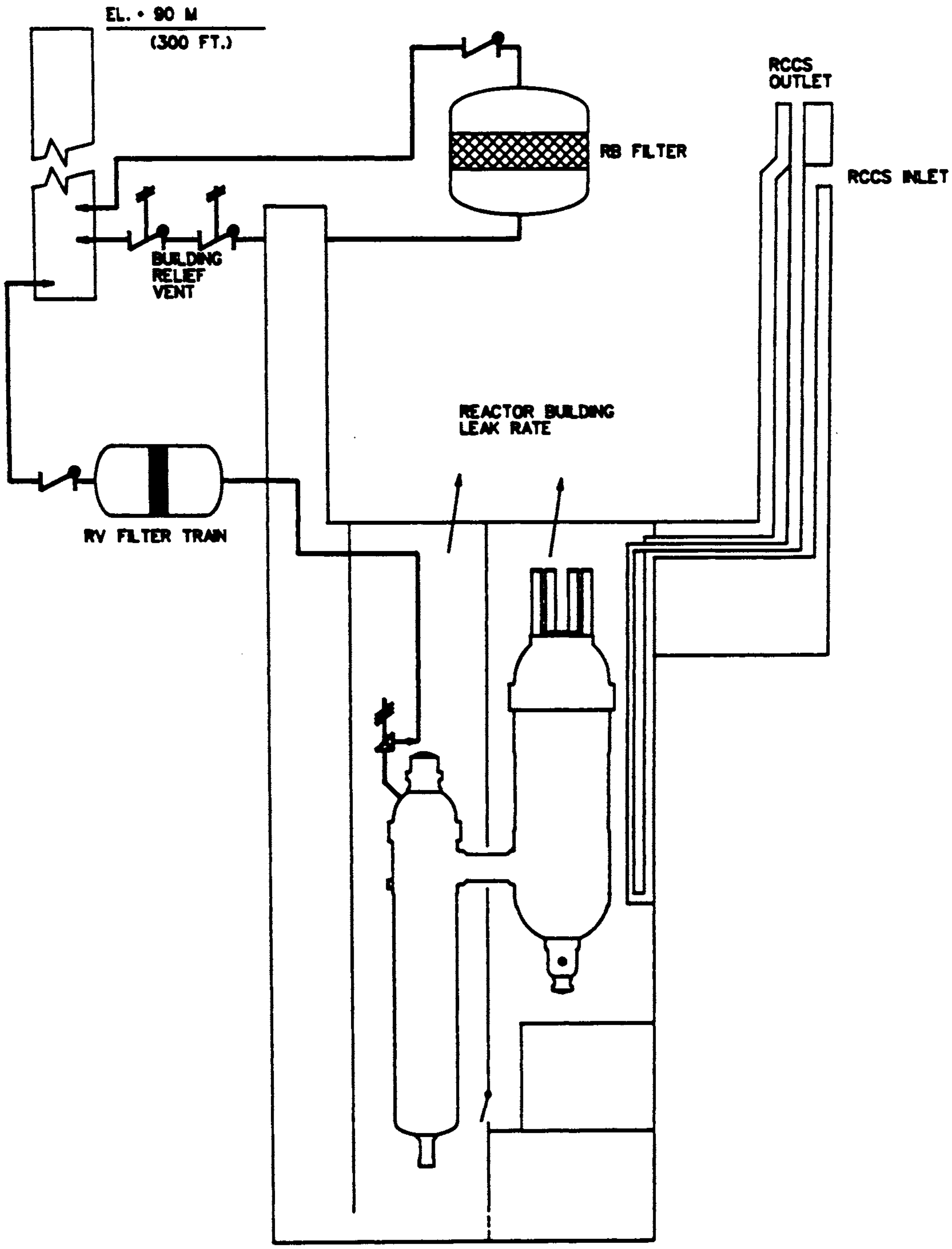


Figure 6-5 ELEVATED RELEASE POINT - SCHEMATIC ARRANGEMENT

Table 6-4
MITIGATION MEASURES CONSIDERED

Vented, Low Pressure Containment	Low-leakage, High Pressure Containment
Case A is the current, reference reactor building. Its vent path is designed to open at 1 psid and it can withstand a 10 psi internal pressure transient load. Fission products which leak into the building are reduced by plateout and deposition before release to the environment via the vent path.	Case F is a conventional, high pressure containment structure based on the design included in the previous Containment Report. It has a 5 vol%/day leak rate. The internal design pressure would need to be on the order of 55 psig, and alternative possibly active decay heat removal and containment cooling systems, would be required.
The building has a leak rate of 1 volume per day. Helium relief valves discharge into the building. Building leakage and the vent path discharge are considered ground level releases, and the site boundary distance is 425 m.	Case G is identical to Case F except that a containment leak rate of 0.5 vol%/day has been evaluated at a site boundary distance of 805 m.
Case B adds a system to collect relief discharges and conduct them to a simple filter located outside the building. It also adds a second building vent path which goes to another simple filter. This filtered path will relieve the pressure from small leaks, which are radiologically important, so that the larger building vent will not open.	
Case C is the same as Case B, except that it uses more efficient filters at both locations. Increasing the area of the reactor building filter improves the decontamination factor and also affects the percent of release to the environment through building leaks which bypass the filter.	
Case D is the same as Case C, except that the site boundary distance is increased from 425 m (1400 ft) to 805 m (2640 ft)	
Case E modifies Case D by adding a tall stack, so that the releases can be considered as elevated releases. The stack, which must be three times as tall as the tallest building, will be about 90 m (300 ft). The discharge from both filters and the main building vent is routed to the stack. Discharges from building leaks are still considered as ground level releases.	

7. EVALUATION OF ALTERNATIVE CONTAINMENT FEATURES WITH REFERENCE FUEL

When the recommended 450 MW(t) MHTGR plant is evaluated for the design and licensing basis events, as developed during the design and analysis of the 350 MW(t) MHTGR, it meets all of the operability and safety goals for pressure transient and offsite radiation dose requirements, with the exception of the Utility/User requirement related to co-locating the Emergency Planning Zone (EPZ) Boundary with the Exclusion Area boundary (EAB). In that case, the expected cumulative site boundary dose exceeds the 5 Rem PAG dose goal at a frequency of 5×10^{-7} per plant year. Alternatively stated, the expected frequency with which the PAG lower threshold thyroid dose will be exceeded is greater than 5×10^{-7} per plant year. It is possible that the containment alternatives described in Section 6 may sufficiently reduce the radiological consequences of the risk dominant accident families (represented by SRDC-11) to reduce the calculated cumulative risk at 5×10^{-7} per plant-year to below 5 rem. That possibility is examined in this section. Refinements to the calculational methods or to the User/Utility requirement may also be able to relieve this circumstance. That aspect of the MHTGR containment performance will be addressed in future work. This section reports on calculated VLPC pressure transient responses for various alternatives, and on calculated site boundary doses assuming reference fuel quality and behavior with alternative VLPC features.

7.1 RESPONSE OF CANDIDATE DESIGN STRATEGIES TO PRESSURE TRANSIENT EVENT INITIATORS

The peak short term pressures developed within the vented reactor building are due to a hypothetical main steam line break (MSLB). Other pressure transient event initiators are the feedwater line failure and failures of the RCPB. There is no offsite dose consequence to either the main steam line failure or the feedwater line failure; hence, the fuel quality does not enter into the evaluation of limiting pressure transient events. The main steam failure gives a higher mass and energy release rate, and is the more severe of the two events.

Credible failures which could occur in the RCPB are determined by the largest attached pipe. In the reactor cavity, this is a 2 in. (inside diameter) line connecting the RCPB to the helium purification system. In the steam generator compartment, the largest attached pipe is the inlet to the helium relief valves. In the proposed configuration for the 450 MW(t) MHTGR, this is a 3.5 in. (inside diameter) line which results in a 10 in^2 break area.

The candidate filters described in Section 6-1 are connected to the reactor building boundary near the vent point and would be affected by the peak building pressure at that point. The pressure at that point is also affected by the building vent setpoint. Table 7-1 shows the worst case pressure transient results for 1 psid and 5 psid vent opening pressures, assuming no filter is added. If a filter is added, it provides an additional pressure relief path and, thus, reduces the pressure which operates on the vent valves. However, even if the largest filter studied (45 ft in diameter) with the minimum bed depth (12 in.) is added, the calculated pressure from a MSL break is 5.92 psi; hence, the addition of the filter offering the greatest pressure relief still results in the vent opening when a main steam line failure occurs.

Table 7-1
PEAK PRESSURE IN SELECTED BUILDING COMPARTMENTS
(ASSUMING NO ADDITIONAL REACTOR BUILDING FILTER)

Building Vent Setpoint	Break Size	Peak Pressure (psig)		
		Node 7 Reactor Cavity[a]	Node 5 Steam Generator Cavity	Node 1 Building Vent
1 psid	Main steam line break	0.9	7.2	8.7
	10 in.^{2} RCPB break	0.25	1.2	1.0
	1 in.^{2} RCPB break	0.74	1.0	1.0
5 psid	Main steam line break	0.9	7.5	9.1
	10 in.^{2} RCPB break	2.3	5.0	5.0
	1 in.^{2} RCPB break	2.39	2.55	2.55

(a) See Figure 4-1 for location of calculational model nodes.

From these data and the head loss shown for helium breaks in Table 6-3, it appears unlikely that increasing the building vent setpoint pressure from the reference value of 1 psid to the maximum feasible pressure of 5 psid is beneficial. The limiting pressure transient from a main steam line break is made slightly worse by this strategy; meanwhile, if a building filter is added, the radiologically important helium breaks (represented by SRDC-11) will not result in a pressure transient sufficient to open vents set at even 1 psid. Building leakage, which would bypass the filter, is not affected by the pressure setpoint. There is no clear requirement to specify the building leak rate at the same pressure as the vent opening; however, if both were set at 5 psid, and the building leak rate were maintained at 1 volume per day, the building would require additional, potentially expensive, features in order to meet its specifications. From this it is concluded that it is not desirable to increase the containment vent pressure setpoint.

7.2 RESPONSE OF CANDIDATE DESIGN STRATEGIES TO SMALL HELIUM BREAK EVENTS

The addition of an alternate filtered VLPC exhaust vent affects small helium break events, represented by SRDC-11. The addition of a filter capability to the helium relief lines does not enter into the analysis of these events because the break is presumed to occur at a location in the RCPB other than through the helium relief train and would bypass such a filter. The filter parameters and building leak rate together affect the percentages of discharged gases which flow through the filter and which flow through the cracks in the building, bypassing the filter. This class of events is also affected by the extension of the EAB radius and by the addition of a tall stack needed to establish an elevated release point. The reactor building leakage rate is important in the evaluation of a tall stack because leakage across the building boundary bypasses both a filter and a tall stack. As described in Section 6, the portion of the flow which could go through the filter and up a stack is a constant percentage of the total flow. Table 7-2 presents the results of site boundary dose calculations for individual events with the addition of individual features. Table 7-3 shows the MHTGR response to SRDC-11 assuming selected feature sets, including two conventional containment alternatives.

The analytical process used for these cases is the same as that used for the reference plant described in Section 3. The analytical model, shown in Figure 7-1, includes all of the features described in Section 6. For each of the vented cases, the building leak rate has been kept constant at 1 volume per day. The parameters assumed in this analysis span the candidate filter designs described in Table 6-2, providing assurance than an optimized design will fall within both the hydraulic and radiological performance factors assumed. Tentative correlation of the decontamination factors used in the cases reported in Tables 7-2 and 7-3 with filter bed depth (t) and diameter (D) result in the following: Case 11-2 relates to a filter with t = 120 in. and D = 5 ft. Case 11-3 relates to a filter with t = 12 in. and D = 45 ft. Case 11-4 relates to a filter with t=120 in. and D = 15 ft. Case 11-5 relates to t = 120 in. and D = 45 ft. Additional decontamination is theoretically possible; however, there is only very limited operating experience with filters of types suitable for this application. A selection study and testing program would be required to confirm the selection of a filter for this process; however, it would appear that decontamination factors greater than the cases reported in Tables 7-2 and 7-3 could be justified.

Table 7-2
MHTGR RESPONSE TO SMALL HELIUM LEAK EVENTS (SRDC-11)
EFFECTS OF INDIVIDUAL CONTAINMENT ALTERNATIVES
(Nominal Doses, Assuming Reference Quality Fuel)

Case[a]	Filter DF[b]	Filter Bypass(%)	EAB (meters)	Stack	0-30 day Thyroid Dose (Rem)	0-30 day Whole Body Dose (Rem)
11-1[c]	N/A	N/A	425	No	1.42×10^{-1}	4.91×10^{-4}
11-2	3	10	425	No	7.41×10^{-2}	3.52×10^{-4}
11-3	3	1	425	No	6.73×10^{-2}	3.38×10^{-4}
11-4	30	10	425	No	4.37×10^{-2}	2.89×10^{-4}
11-5	30	1	425	No	3.39×10^{-2}	2.69×10^{-4}
11-1	N/A	N/A	425	Yes	4.03×10^{-3}	1.49×10^{-5}
11-1	N/A	N/A	800	No	7.47×10^{-2}	2.61×10^{-4}

[a] Case numbers are related to the case identification in supporting calculations.
[b] Decontamination factors are applied to halogens and particulates. The filter does not retain any noble gasses.
[c] Case 11-1 is the reference VLPC design, without addition of any filters.

7.3 RESPONSE OF CANDIDATE DESIGN STRATEGIES TO WATER INGRESS EVENTS

Events which lead to radiological release via the helium relief valve train, typified by SRDC-6, are the most severe in terms of the magnitude of individual event doses at the site boundary. This class of events is treated by the proposed filter train on the helium relief lines, and by strategies which would change the site and release conditions. SRDC-6 has been analyzed for the reference quality fuel for the same range of alternative containment features and feature combinations as described in Section 6.2. Tables 7-4 and 7-5 show the results of these calculations.

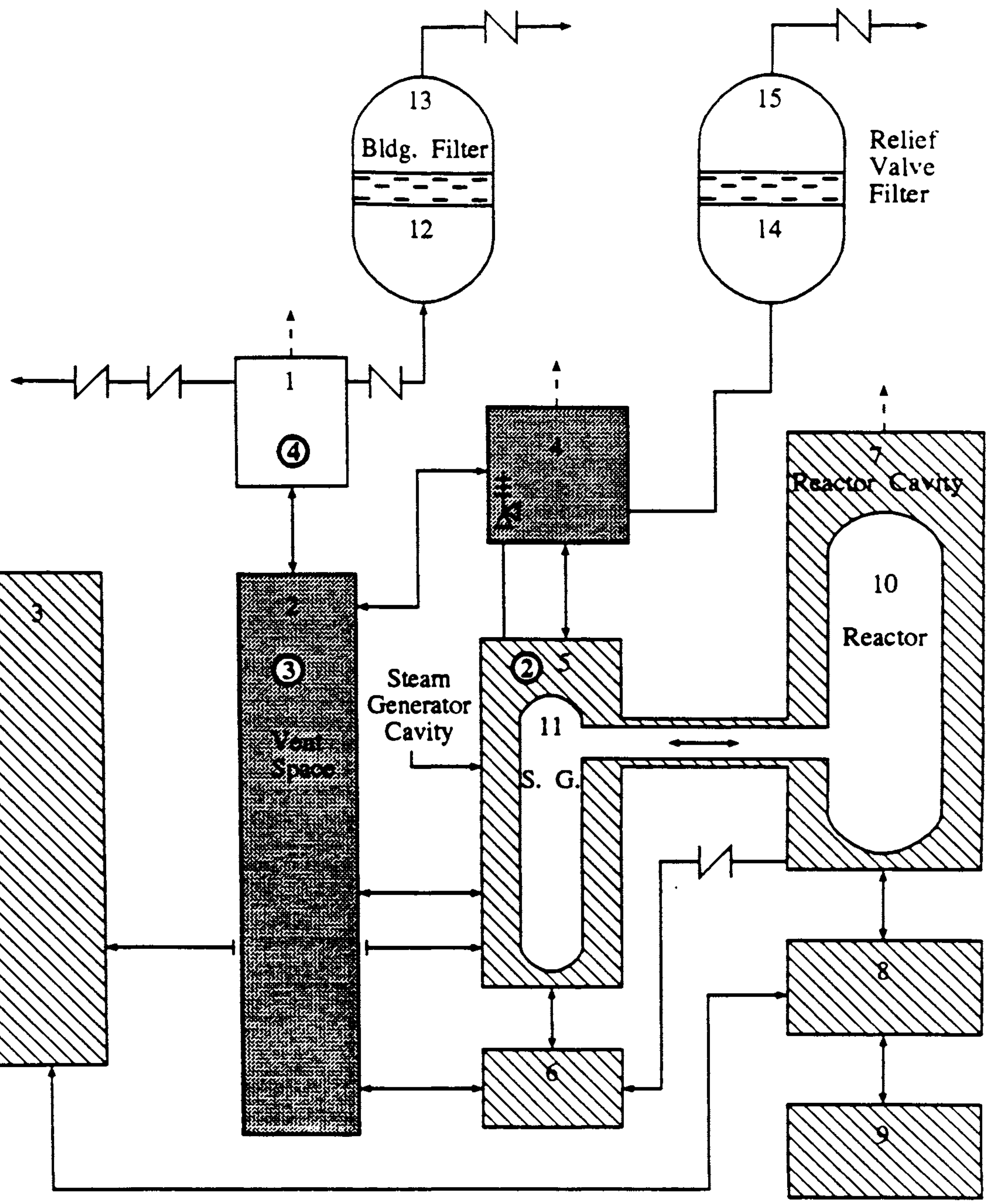
It is unlikely that the proposed filters could be configured to fit within the reactor building. In these analyses, it has been assumed that the filter is located outdoors and that the relief valve train filter discharges to the environment. The filters are specified by the decontamination factor which they achieve on halogens and particulates. As with all non-adsorptive filter mediums, the proposed glass fiber or sand beds do not reduce

Table 7-3
MHTGR RESPONSE TO SMALL HELIUM LEAK EVENTS (SRDC-11)
EFFECTS OF COMBINED ALTERNATIVE FEATURES
(Nominal Doses, Assuming Reference Quality Fuel)

Case (a)	Set (b)	Filter DF (c)	Filter By-pass	EAB (m)	Stack	Contain. Leak Rate % vol/d	0-30 day Thyroid Dose (Rem)	0-30 day Whole Body Dose (Rem)
11-1[d]	A	N/A	N/A	425	No	Vented	1.42×10^{-1}	4.91×10^{-4}
11-2	B	3	10	425	No	Vented	7.41×10^{-2}	3.52×10^{-4}
11-3		3	1	800	No	Vented	3.51×10^{-2}	1.79×10^{-4}
11-5	C	30	1	425	No	Vented	3.39×10^{-2}	2.69×10^{-4}
11-5		30	1	425	Yes	Vented	1.27×10^{-3}	8.83×10^{-6}
11-5	D	30	1	800	No	Vented	1.73×10^{-2}	1.42×10^{-4}
11-5	E	30	1	800	Yes	Vented	7.51×10^{-4}	5.47×10^{-6}
11-6	F	N/A	N/A	425	N/A	5%/d	1.77×10^{-3}	1.36×10^{-5}
11-6		N/A	N/A	800	N/A	5%/d	9.07×10^{-4}	4.65×10^{-6}
11-7		N/A	N/A	425	N/A	0.5%/d	1.77×10^{-4}	1.60×10^{-6}
11-7	G	N/A	N/A	800	N/A	0.5%/d	9.08×10^{-5}	4.87×10^{-7}

[a] Case numbers are related to the case identification in supporting calculations.
[b] Set numbers are related to the Feature Sets in the Executive Summary.
[c] Decontamination factors are applied to halogens and particulates. The filter does not retain any noble gasses.
[d] Case 11-1 is the reference VLPC building design, without any additional filters.

noble gasses. The addition of a system which collects the discharge from the helium relief valves and discharges it to the environment via a filter has the additional benefit of enhancing plant operator safety by eliminating hazards due to the filling of the reactor building spaces with suffocating and potentially hot gas. It also prevents the contamination of interior building spaces by fission products entrained in the helium coolant, and would thereby contribute to reducing the radiation dose to plant operators.



② = LOCADOSE, NE319 Node Number (Node 1 = Environment)

Figure 7-1 DOSE CALCULATION MODEL, INCLUDING ALTERNATIVE CONTAINMENT FEATURES

Table 7-4
MHTGR RESPONSE TO WATER INGRESS EVENTS (SRDC-6)
EFFECTS OF INDIVIDUAL ALTERNATIVE FEATURES
(Nominal Doses, Assuming Referemce Quality Fuel)

Case[a]	Filter DF[b]	Filter Medium	EAB (meters)	Stack	0-30 day Thyroid Dose (Rem)	0-30 day Whole Body Dose (Rem)
6-1[c]	N/A	N/A	425	No	1.51	1.08×10^{-2}
6-2	10	Sand	425	No	3.08×10^{-1}	3.49×10^{-3}
6-3	100	Glass Fiber	425	No	3.08×10^{-2}	2.01×10^{-3}
6-4	100[d]	Sand	425	No	3.02×10^{-2}	1.97×10^{-3}
6-1	N/A	N/A	425	Yes	1.32×10^{-1}	1.02×10^{-3}
6-1	N/A	N/A	805	No	1.11	8.16×10^{-3}

[a] Case numbers are related to the case identification in supporting calculations.

[b] Decontamination factors are applied to halogens and particulates. The filter does not retain any noble gasses.

[c] Case 6-1 is the reference VLPC building design, without any additional filters.

[d] Case 6-4 uses DF=100 for halogens, DF=1000 for particulates.

7.4 EFFECT OF CANDIDATE STRATEGIES ON THE FREQUENCY OF EXCEEDING PAG THRESHOLD LIMITS

The impact of the design alternatives on the 450 MW(t) MHTGR's potential for satisfying the User/Utility offsite sheltering or evacuation requirement has been assessed. This requirement is interpreted to mean that the lower level EPA PAG for sheltering during accidents is not exceeded at or above a frequency of 5×10^{-7} per plant year. In this study, the plant risk of exceeding the PAG has been estimated by using SRDC-6 and SRDC-11 predicted doses and the uncertainty assessments performed in the 350 MW(t) probabilistic risk assessment (PRA) performed in 1988 (Ref. 2). Inherent in these assumptions is that the accident doses and uncertainties for the 27 accident families identified in the PRA can be scaled based on the two dominant events. The estimated risk based on this approach reflects uncertainties in both the accident frequency and consequences as developed for the 350 MW(t) MHTGR. The accident frequency uncertainty assessment considers the initiating event frequency and the probabilities that the transient proceeds in a given manner. Consequence uncertainties consider variations in the as-manufactured fuel quality, core temperatures, fuel performance, fission

Table 7-5
MHTGR RESPONSE TO WATER INGRESS EVENTS (SRDC-6)
EFFECTS OF COMBINED ALTERNATIVE FEATURES
(Nominal Doses, Assuming Reference Quality Fuel)

Case (a)	Set (b)	Filter DF (c)	EAB (m)	Stack	Contain. Leak Rate % vol/d	0-30 day Thyroid Dose (Rem)	0-30 day Whole Body Dose (Rem)
6-1[d]	A	N/A	425	No	Vented	1.51	1.08×10^{-2}
6-2	B	10	425	No	Vented	3.08×10^{-1}	3.49×10^{-3}
6-3	C	100	425	No	Vented	3.08×10^{-2}	2.01×10^{-3}
6-3	D	100	800	No	Vented	2.24×10^{-2}	1.49×10^{-3}
6-3	E	100	800	Yes	Vented	1.45×10^{-3}	9.88×10^{-5}
6-4		100[e]	425	Yes	Vented	2.65×10^{-3}	1.79×10^{-4}
6-4		100[e]	800	No	Vented	2.20×10^{-2}	1.46×10^{-3}
6-5	F	N/A	425	N/A	5%/d	1.80×10^{-3}	7.58×10^{-5}
6-5		N/A	800	N/A	5%/d	9.83×10^{-4}	2.69×10^{-5}
6-6		N/A	425	N/A	0.5%/d	1.80×10^{-4}	8.82×10^{-6}
6-6	G	N/A	800	N/A	0.5%/d	9.85×10^{-5}	2.81×10^{-6}

[a] Case numbers are related to the case identification in supporting calculations.
[b] Set numbers are related to the Feature Sets in the Executive Summary.
[c] Decontamination factors are applied to halogens and particulates. The filter does not retain any noble gasses.
[d] Case 6-1 is the reference design.
[e] Case 6-4 uses DF=100 for halogens, DF=1000 for particulates.

product transport mechanisms, and weather. The results of this approach should be judged as an indication of plant risk trends. Since these two events are representative of risk dominant accident family contributors to the 350 MW(t) plant risk, they are also expected to be important to the 450 MW(t) plant risk.

Risk was estimated for two alternative design options to show the effect of the possible alternatives on reducing plant risk. In option 1, it was assumed that the relief train filters associated with Case 6-3 (Table 7-4) are included in the design. This design will effectively reduce SRDC-6 releases. However, SRDC-11 consequences remain unchanged with this design modification so Case 11-1 dose results at 425 m are applicable (Table 7-2).

Based on the 350 MW(t) PRA, at higher frequencies for option 1, above about 10^{-2} per plant year, primary coolant leaks with forced cooling are dominant. In the frequency range below about 10^{-3} per plant year primary coolant leaks without forced cooling (events similar to SRDC-11) dominate. The thyroid dose at the frequency of 5×10^{-7} per plant has been reduced from about 15 Rem (see base case, Fig. 4-4) to between 10 and 14 Rem. A range is presented because of the approximations made in this assessment. This modest change is to be expected since primary coolant leaks dominate the risk curve and are unaffected by the relief valve filter modification.

In the second option relief train filters from Case 6-3 and a building filter representative of Case 11-3 (Table 7-2) were assumed to be included in the design. The thyroid dose at a frequency of 5×10^{-7} per plant has been reduced from about 15 to between 4 and 7 Rem. This more significant change is to be expected since primary coolant leaks are dominant contributors to risk and building filters reduce these doses.

For both options considered, the results indicate that the User/Utility goal is more closely approached for the case with relief train and building filters. Since both options considered reduce plant risk, a more careful assessment of the cumulative risk curves considering these modifications are warranted. The approach used in this assessment was simplified to provide some insight into risk reduction capabilities of the alternatives considered. Additional design review of these alternatives, a detailed assessment of the 450 MW(t) accident families frequencies, and a 450 MW(t) dose uncertainty assessment will likely produce a design that can meet the restrictive PAG requirement.

7.5 CONCLUSIONS REGARDING CONTAINMENT ALTERNATIVES ASSUMING REFERENCE FUEL QUALITY

At this level of design review it is recommended that a filtered relief train be included into the design. Additional work is necessary to select a filtered building design which coupled with stacks, increased EAB or methods improvements could further improve the plant's potential for meeting the PAGs.

8. RESPONSE OF VLPC ALTERNATIVES WITH ALTERNATIVE SOURCE TERMS

Because the alternative source terms are themselves low probability occurances, when coupled with the accident frequencies associated with the MHTGR licensing bases (see Section 3.2.2 for accident family frequencies), the resulting calculated dose consequences should not be judged against the PAGs or top level regulatory requirements. Instead, postulated MHTGR accidents including consideration of the alternative source terms should be compared against dose levels that lead to prompt fatalities. This level is about 600 Rem to the whole body. Prompt thyroid fatalities would not be expected; however, non-fatal acute effects would be expected for doses over 3000 Rem. This comparison will provide insight into the plant's capability to mitigate residual risk for beyond licensing basis events.

8.1 ACCIDENT CONSEQUENCE ASSUMING LOWER LEVEL FUEL QUALITY

8.1.1 Response of Candidate Containment Design Strategies to Small Primary Coolant Leak

The alternative containment features have been evaluated assuming the lower bound fuel quality, in a fashion similar to the analysis using reference fuel quality discussed in Section 7.2. Table 8-1 presents the results of site boundary dose calculations for SRDC-11 like events with the addition of individual features. Table 8-2 shows the MHTGR response to SRDC-11-like events assuming selected feature sets, including two conventional containment alternatives. The analytical methods used to assess these cases, and the correlation to potential filter designs, is the same as used for the reference fuel, as described in Sections 3 and 7.

8.1.2 Response of Candidate Design Strategies to Water Ingress Events

Events which lead to radiological release via the helium relief valve train, typified by steam generator leaks like SRDC-6, are the most severe in terms of individual event doses at the site boundary. This class of events has also been evaluated for the efficacy of various alternative containment features, assuming lower bound fuel quality as described in Section 5.1. Tables 8-3 and 8-4 show the results of these calculations.

8.2 PLANT RESPONSE TO RAPID HYDROLYSIS AND WEAK FUEL HYPOTHESES

Both the rapid hydrolysis and weak fuel scenarios are considered beyond the licensing basis for the MHTGR. They are believed to be lower in expected probability than the Bounding Event Sequences which were examined by the MHTGR Program in response to NRC questions, although it is extremely difficult to accurately

Table 8-1
MHTGR RESPONSE TO SMALL HELIUM LEAK EVENTS (SRDC-11)
EFFECTS OF INDIVIDUAL CONTAINMENT ALTERNATIVES
(Nominal Doses Assuming Lower Limit Fuel Quality)

Case[a]	LPVC Filter DF[b]	Filter Bypass (%)	EAB (Meters)	Stack	0-30 Day Thyroid Dose (Rem)	0-30 Day Whole Body Dose (Rem)
11-15[c]	N/A	N/A	425	No	1.35	3.88×10^{-3}
11-16	3	10	425	No	7.08×10^{-1}	2.62×10^{-3}
11-17	3	1	425	No	6.44×10^{-1}	2.50×10^{-3}
11-18	30	10	425	No	4.20×10^{-1}	2.06×10^{-3}
11-19	30	1	425	No	3.27×10^{-1}	1.87×10^{-3}
11-15	N/A	N/A	425	Yes	3.87×10^{-2}	1.28×10^{-4}
11-15	N/A	N/A	805	No	7.11×10^{-1}	2.08×10^{-3}

(a) Case numbers are related to the case identification in supporting calculations.
(b) Decontamination factors are applied to halogens and particulates. The filter does not retain any noble gasses.
(c) Case 11-15 is the reference containment system design.

assess the likelihood of the phenomena described. These alternative source terms have been reviewed to determine if they have unacceptably severe consequences. The analyses performed in making this assessment are only approximate. The dose consequences for these phenomena have been estimated by taking the nominal value for thyroid doses due to SRDC-6 and SRDC-11, and adjusting them by a ratio based on the I-131 source term released from the reactor vessel.

Section 5.2 developed a basis for the release of I-131 from the reactor vessel which is 2.2 to 2.3 times greater than the release calculated using reference MHTGR hydrolysis methods. If it is assumed that SRDC-6 is otherwise unchanged by this assumption, then the resultant doses for SRDC-6 will be as shown in Tables 8-5 and 8-6. The containment feature sets, including correlation to filter candidates and conventional containment alternative leak rates are the same as are used in Section 7. These results show that the MHTGR VLPC strategy

Table 8-2
MHTGR RESPONSE TO SMALL HELIUM LEAK EVENTS (SRDC-11)
EFFECTS OF COMBINED ALTERNATIVE FEATURES
(Nominal Doses Assuming Lower Limit Fuel Quality)

Case (a)	Set (b)	Filter DF (c)	Filter By-pass	EAB (m)	Stack	Contain. Leak Rate % vol/d	0-30 day Thyroid Dose (Rem)	0-30 day Whole Body Dose (Rem)
11-15	A	N/A	N/A	425	No	Vented	1.35	3.88×10^{-3}
11-16	B	3	10	425	No	Vented	7.08×10^{-1}	2.62×10^{-3}
11-17		3	1	805	No	Vented	3.36×10^{-1}	1.33×10^{-3}
11-18	C	30	1	425	No	Vented	3.27×10^{-1}	1.87×10^{-3}
11-19		30	1	425	Yes	Vented	1.24×10^{-2}	7.14×10^{-5}
11-19	D	30	1	805	No	Vented	1.67×10^{-1}	9.95×10^{-4}
11-19	E	30	1	805	Yes	Vented	7.29×10^{-3}	4.26×10^{-5}
11-20	F	N/A	N/A	425	N/A	5%/d	1.71×10^{-2}	1.24×10^{-4}
11-20		N/A	N/A	805	N/A	5%/d	8.75×10^{-3}	4.04×10^{-5}
11-21		N/A	N/A	425	N/A	0.5%/d	1.71×10^{-3}	1.47×10^{-5}
11-22	G	N/A	N/A	805	N/A	0.5%/d	8.77×10^{-4}	4.25×10^{-6}

(a) Case numbers are related to the case identification in supporting calculations.
(b) Set numbers are related to the Feature Sets in the Executive Summary. Set A is the reference containment design.
(c) Decontamination factors are applied to halogens and particulates. The filter does not retain any noble gasses.

has a high tolerance for the rapid hydrolysis behavior hypothesis. Even when rapid hydrolysis is assumed to coincide with lower than specified fuel quality, dose results are far below 10CFR100 limits. If a filter is added to the relief train (case 6-33 or 6-34), the consequences of SRDC-6 are below Utility/User thyroid dose goals of 5 Rem. The MHTGR appears to be highly tolerant of uncertainty regarding rapid hydrolysis.

Section 5.3 developed a rough estimate of the consequences of a breakdown in manufacturing quality control, which has been suggested as a possible mechanism leading to a undetected flaw in a substantial amount

Table 8-3
MHTGR RESPONSE TO WATER INGRESS EVENTS (SRDC-6)
EFFECTS OF INDIVIDUAL ALTERNATIVE FEATURES
(Nominal Doses Assuming Lower Limit Fuel Quality)

Case[a]	Relief Train Filter DF[b]	Relief Train Filter Medium	EAB (meters)	Release Elev.	0-30 day Thyroid Dose (Rem)	0-30 day Whole Body Dose (Rem)
6-19[c]	N/A	N/A	425	Ground	19.5	0.101
6-20	10	Sand	425	Ground	4.07	3.53×10^{-2}
6-21	100	Glass Fiber	425	Ground	4.07×10^{-1}	2.08×10^{-2}
6-22	100[d]	Sand	425	Ground	4.02×10^{-1}	2.05×10^{-2}
6-19	N/A	N/A	425	Elevated	1.41	8.61×10^{-3}
6-19	N/A	N/A	805	Ground	13.3	7.34×10^{-2}

(a) Case numbers are related to the case identification in supporting calculations.
(b) Decontamination factors are applied to halogens and particulates. The filter does not retain any noble gasses.
(c) Case 6-19 is the reference design.
(d) Case 6-22 uses DF=100 for halogens, DF=1000 for particulates.

of fuel judged to meet the quality specification. If 50 percent of a reload batch failed at 1200°C, an estimated 15,000 Ci of I-131 would be released into the reactor coolant. If it is assumed that the limiting event (SRDC-11) is unchanged, then an estimate of the dose consequences of this source term can be made by developing a ratio of the I-131 released and applying it to the calculated results of SRDC-11. This hypothesis leads to an incremental dose which must be added to the results due to fuel particles which met the earlier definition of defective fuel. Table 8-7 presents the results of this estimating process.

The vented low pressure MHTGR containment can be augmented with fission product retention or offsite dose reduction features to mitigate dose consequences from fuel behavior based on the weak fuel hypothesis as proposed by NRC reviewers. The set of best possible features includes the addition of a tall stack, extension of the EAB to 800 meters, selection of the best decontamination factors identified in the survey of candidate filters, including a large frontal area, so that only a very small fraction of the gas released from the

Table 8-4
MHTGR RESPONSE TO WATER INGRESS EVENTS (SRDC-6)
EFFECTS OF COMBINED ALTERNATIVE FEATURES
(Nominal Doses Assuming Lower Limit Fuel Quality)

Case[a]	Set[b]	Filter DF[c]	EAB (m)	Stack	Contain. Leak Rate % vol/d	0-30 day Thyroid Dose (Rem)	0-30 day Whole Body Dose (Rem)
6-19	A[d]	N/A	425	No	Vented	19.5	0.101
6-20	B	10	425	No	Vented	4.07	3.53×10^{-2}
6-21	C	100	425	No	Vented	4.07×10^{-1}	2.08×10^{-2}
6-21	D	100	805	No	Vented	2.76×10^{-1}	1.48×10^{-2}
6-21	E	100	805	Yes	Vented	1.60×10^{-2}	9.33×10^{-4}
6-22		100[e]	425	Yes	Vented	2.85×10^{-2}	1.68×10^{-3}
6-22		100[e]	805	No	Vented	2.73×10^{-1}	1.46×10^{-2}
6-23	F	N/A	425	N/A	5%/d	3.36×10^{-2}	7.54×10^{-4}
6-23		N/A	805	N/A	5%/d	1.80×10^{-2}	2.53×10^{-4}
6-24		N/A	425	N/A	0.5%/d	3.37×10^{-3}	8.87×10^{-5}
6-24	G	N/A	805	N/A	0.5%/d	1.80×10^{-3}	2.66×10^{-5}

(a) Case numbers are related to the case identification in supporting calculations.
(b) Set numbers are related to the Feature Sets in the Executive Summary.
(c) Decontamination factors are applied to halogens and particulates. The filter does not retain any noble gasses.
(d) Set A is the reference design.
(e) Case 6-22 uses DF=100 for halogens, DF=1000 for particulates.

RCPB goes directly to the environment, bypassing both the filter and the stack. Making the building into a pressure retaining structure instead of adding a filter and a stack would reduce the site boundary dose to a level comparable to the best VLPC feature set, assuming that the containment functions correctly. Based on LWR experience, the probability that the containment fails to isolate when required may be as low as 1 per 100 demands. While the VLPC strategy and the conventional containment approach lead to comparable site boundary doses for individual events, it is expected that the cumulative dose at low frequencies will be lower for the VLPC than for the conventional containment, because of the latter's relatively high failure rate.

Table 8-5
MHTGR RESPONSE TO WATER INGRESS EVENTS (SRDC-6)
EFFECTS OF COMBINED ALTERNATIVE FEATURES
(Nominal Doses Assuming Reference Quality Fuel, Rapid Hydrolysis)

Case[a]	Set[b]	Filter DF[c]	EAB (m)	Stack	Contain. Leak Rate % vol/d	0-30 day Thyroid Dose (Rem)	0-30 day Whole Body Dose (Rem)
6-13	A[d]	N/A	425	No	Vented	3.61	2.59×10^{-2}
6-14	B	10	425	No	Vented	7.38×10^{-1}	8.35×10^{-3}
6-15	C	100	425	No	Vented	7.38×10^{-2}	4.81×10^{-3}
6-15	D	100	805	No	Vented	5.35×10^{-2}	3.57×10^{-3}
6-15	E	100	805	Yes	Vented	3.48×10^{-3}	2.37×10^{-4}
6-16		100[e]	425	Yes	Vented	6.35×10^{-3}	4.28×10^{-4}
6-16		100[e]	805	No	Vented	5.28×10^{-2}	3.50×10^{-3}
6-17	F	N/A	425	N/A	5%/d	4.31×10^{-3}	1.82×10^{-4}
6-17		N/A	805	N/A	5%/d	2.35×10^{-3}	6.45×10^{-5}
6-18		N/A	425	N/A	0.5%/d	4.32×10^{-4}	2.11×10^{-5}
6-18	G	N/A	805	N/A	0.5%/d	2.36×10^{-4}	6.72×10^{-6}

(a) Case numbers are related to the case identification in supporting calculations.
(b) Set numbers are related to the Feature Sets in the Executive Summary.
(c) Decontamination factors are applied to halogens and particulates. The filter does not retain any noble gasses.
(d) Set A is the reference design.
(e) Case 6-16 uses DF=100 for halogens, DF=1000 for particulates.

Table 8-6
MHTGR RESPONSE TO WATER INGRESS EVENTS (SRDC-6)
EFFECTS OF COMBINED ALTERNATIVE FEATURES
(Nominal Doses Assuming Lower Limit Fuel Quality, Rapid Hydrolysis)

Case[a]	Set[b]	Filter DF[c]	EAB (m)	Stack	Contain. Leak Rate % vol/d	0-30 day Thyroid Dose (Rem)	0-30 day Whole Body Dose (Rem)
6-31	A[d]	N/A	425	No	Vented	35.7	0.256
6-32	B	10	425	No	Vented	7.30	8.26×10^{-2}
6-33	C	100	425	No	Vented	0.730	4.76×10^{-2}
6-33	D	100	805	No	Vented	0.530	3.53×10^{-2}
6-33	E	100	805	Yes	Vented	3.45×10^{-2}	2.34×10^{-3}
6-34		100[e]	425	Yes	Vented	6.28×10^{-2}	4.23×10^{-3}
6-34		100[e]	805	No	Vented	5.22×10^{-1}	3.46×10^{-2}
6-35	F	N/A	425	N/A	5%/d	4.26×10^{-2}	1.80×10^{-3}
6-35		N/A	805	N/A	5%/d	2.33×10^{-2}	6.39×10^{-4}
6-36		N/A	425	N/A	0.5%/d	4.27×10^{-3}	2.09×10^{-4}
6-36	G	N/A	805	N/A	0.5%/d	2.33×10^{-3}	6.66×10^{-5}

(a) Case numbers are related to the case identification in supporting calculations.
(b) Set numbers are related to the Feature Sets in the Executive Summary.
(c) Decontamination factors are applied to halogens and particulates. The filter does not retain any noble gasses.
(d) Set A is the reference VLPC design.
(e) Case 6-34 uses DF=100 for halogens, DF=1000 for particulates.

8.3 PLANT RESPONSE TO BOUNDING PROMPT SOURCE TERM WITH REFERENCE FUEL QUALITY

8.3.1 Response of the Reference VLPC design

The analytical methods used to assess this case are the same as that used for the reference plant, as described in Section 4. The analytical model is the same as shown in Figure 7-1, with the building leak rate kept

Table 8-7
MHTGR RESPONSE TO SMALL HELIUM LEAK EVENTS (SRDC-11)
EFFECTS OF COMBINED ALTERNATIVE FEATURES
(Nominal Incremental Doses Assuming Weak Fuel)

Case[a]	Set[b]	Filter DF[c]	Filter Bypass (%)	EAB (m)	Stack	Contain. Leak Rate % vol/d	0-30 day Thyroid Dose (Rem)	0-30 day Whole Body Dose (Rem)
11-29	A[d]	N/A	N/A	425	No	Vented	24.4	8.47×10^{-2}
11-30	B	3	10	425	No	Vented	12.8	6.07×10^{-2}
11-31		3	1	800	No	Vented	6.06	3.08×10^{-2}
11-33	C	30	1	425	No	Vented	5.84	4.64×10^{-2}
11-33		30	1	425	Yes	Vented	0.219	1.52×10^{-3}
11-33	D	30	1	800	No	Vented	2.99	2.44×10^{-2}
11-33	E	30	1	800	Yes	Vented	0.129	9.43×10^{-4}
11-34	F	N/A	N/A	425	N/A	5%/d	0.305	2.35×10^{-3}
11-34		N/A	N/A	800	N/A	5%/d	0.156	8.02×10^{-4}
11-35		N/A	N/A	425	N/A	0.5%/d	3.06×10^{-2}	2.76×10^{-4}
11-35	G	N/A	N/A	800	N/A	0.5%/d	1.57×10^{-2}	8.40×10^{-5}

(a) Case numbers are related to the case identification in supporting calculations.
(b) Set numbers are related to the Feature Sets in the Executive Summary.
(c) Decontamination factors are applied to halogens and particulates. The filter does not retain any noble gasses.
(d) Set A is the reference VLPC design.

constant at 1 volume per day. The calculated radiation exposure to an individual at the site boundary with the bounding prompt source term are shown in Table 8-8. These are also compared to the best estimate source term dose consequences of SRDC-11 and SRDC-6 presented in Table 4-3 and Table 4-4 respectively. Despite SRDC-10 having a smaller cumulative release from the vessel than SRDC-11, its dose is larger due to the larger prompt source term and the weather model (see Table 4-2), which assumes that the atmospheric dispersion is

Table 8-8
SITE BOUNDARY DOSE CONSEQUENCES OF LIMITING EVENTS - 450 MWt MHTGR
REFERENCE FUEL QUALITY ASSUMED, REFERENCE AND BOUNDING PROMPT SOURCE
(50th Percentile Doses)

	Release Events		
	Rapid Depressurization and loss of Forced Cooling (SRDC-10)	Depressurized loss of Forced Cooling (SRDC-11)	Water Ingress with Depressurized loss of Forced Cooling (SRDC-6)
Best Estimate Prompt Source Term			
Thyroid Dose (Rem)	1.48×10^{-2}	0.142	1.51
W. Body Dose (Rem)	2.81×10^{-4}	4.91×10^{-4}	1.08×10^{-2}
Bounding Prompt Source Term			
Thyroid Dose (Rem)	0.660	0.555	1.71
W. Body Dose (Rem)	2.48×10^{-3}	1.69×10^{-3}	1.16×10^{-2}

worse during the initial part of the event. The way in which weather is modeled for SRDC-10 is conservative, and consistent with NRC accepted practice. At some point in the future, it may be appropriate to re-examine the weather models used for SRDC-11 and SRDC-6, to ensure that they are also conservative. SRDC-11, which has a slow, long term release, is relatively insensitive to the weather model.

8.3.2 Response of Candidate Containment Design Strategies to a Small Helium Leak

According to past studies, events which lead to radiological release via a small helium leak in the RCPB, typified by SRDC-11, are risk dominant. This class of events is treated by the proposed filter release pathway from the reactor building, and by strategies which would change the site and release conditions. SRDC-11 has been analyzed for the reference quality fuel with a bounding prompt source term for the same range of alternative containment features and feature combinations as described in Section 6. Table 8-9 shows the results of these assessments.

Table 8-9
MHTGR RESPONSE TO SMALL HELIUM LEAK EVENTS (SRDC-11)
EFFECTS OF ALTERNATIVE CONTAINMENT FEATURES
(Reference Fuel, Bounding Prompt Source Term Assumed)

Case (a)	Set (b)	Filter DF (c)	Filter Bypass (%)	EAB (m)	Stack	Contain. Leak Rate vol.%/d	0-30 day Thyroid Dose (Rem)	0-30 day W. Body Dose (Rem)
11-8	A(d)	1	N/A	425	No	Vented	0.555	$1.69x10^{-3}$
11-9	B	3	10	425	No	Vented	0.263	$9.08x10^{-4}$
11-10		3	1	805	No	Vented	0.121	$4.38x10^{-4}$
11-12	C	30	1	425	No	Vented	$8.94x10^{-2}$	$4.42x10^{-4}$
11-12		30	1	425	Yes	Vented	$2.54x10^{-3}$	$1.39x10^{-5}$
11-12	D	30	1	805	No	Vented	$4.63x10^{-2}$	$2.35x10^{-4}$
11-12	E	30	1	805	Yes	Vented	$1.63x10^{-3}$	$8.75x10^{-6}$
11-13	F	N/A	N/A	425	N/A	5%/d	$4.81x10^{-3}$	$2.46x10^{-5}$
11-13		N/A	N/A	805	N/A	5%/d	$2.50x10^{-3}$	$1.07x10^{-5}$
11-14		N/A	N/A	425	N/A	0.5%/d	$4.82x10^{-4}$	$2.71x10^{-6}$
11-14	G	N/A	N/A	805	N/A	0.5%/d	$2.50x10^{-4}$	$10.9x10^{-6}$

(a) Case numbers are related to case identification in supporting calculations.
(b) Set numbers are related to the Feature Sets in the Executive Summary.
(c) Decontamination factors are applied to halogens and particulates. The filter does not retain any noble gasses.
(d) Feature Set A is the reference VLPC design, without any additional filters.

8.3.3 Response of Candidate Containment Design Strategies to Water Ingress Event

Events which lead to radiological release via the helium relief valve train, typified by SRDC-6, are the most severe in terms of the magnitude of individual event doses at the site boundary. This class of events is treated by the proposed filter train on the helium relief lines, and by strategies which would change the site and release conditions. SRDC-6 has been analyzed for the reference quality fuel with a bounding prompt source term for the same range of alternative containment features and feature combinations as described in Section 6. Table 8-10 shows the results of these assessments.

Table 8-10
MHTGR RESPONSE TO WATER INGRESS EVENTS (SRDC-6)
EFFECTS OF COMBINED ALTERNATIVE CONTAINMENT FEATURES
(Reference Fuel, Bounding Source Term Assumed)

Case (a)	Set (b)	Filter DF (c)	EAB (m)	Stack	Contain. Leak Rate vol.%/d	0-30 day Thyroid Dose (Rem)	0-30 day W. Body Dose (Rem)
6-7	A[d]	N/A	425	No	Vented	1.71	$1.16x10^{-2}$
6-8	B	10	425	No	Vented	0.345	$3.62x10^{-3}$
6-9	C	100	425	No	Vented	$3.45x10^{-2}$	$2.02x10^{-3}$
6-9	D	100	805	No	Vented	$2.44x10^{-2}$	$1.50x10^{-3}$
6-9	E	100	805	Yes	Vented	$1.53x10^{-3}$	$9.92x10^{-5}$
6-10		100[e]	425	Yes	Vented	$2.75x10^{-3}$	$1.79x10^{-4}$
6-10		100[e]	805	No	Vented	$2.39x10^{-2}$	$1.47x10^{-3}$
6-11	F	N/A	425	N/A	5%/d	$2.51x10^{-3}$	$7.94x10^{-5}$
6-11		N/A	805	N/A	5%/d	$1.35x10^{-3}$	$2.88x10^{-5}$
6-12		N/A	425	N/A	0.5%/d	$2.52x10^{-4}$	$9.81x10^{-6}$
6-12	G	N/A	805	N/A	0.5%/d	$1.35x10^{-4}$	$3.00x10^{-6}$

(a) Case numbers are related to case identification in supporting calculations.
(b) Set numbers are related to the Feature Sets in the Executive Summary.
(c) Decontamination factors are applied to halogens and particulates. The filter does not retain any noble gasses.
(d) Feature Set A is the reference VLPC design, without any additional filters.
(e) Case 6-10 uses DF=100 for halogens and DF=1000 for particulates.

8.3.4 Response of Candidate Containment Design Strategies to Large Helium Leak

According to past studies, events which lead to radiological release via a large helium leak in the RCPB, typified by SRDC-10, are the most prompt. This class of events are typically treated by strategies which would change the site and release conditions. Only the largest of the proposed filter release pathway from the reactor building may have any effect, as the blowdown is designed to bypass smaller sized reactor building filters. SRDC-10 has been analyzed for the reference quality fuel with a bounding prompt source term for the same range of alternative containment features and feature combinations as described in Section 6. Table 8-11 shows the results of these assessments.

Table 8-11
MHTGR RESPONSE TO LARGE HELIUM LEAK EVENTS (SRDC-10)
EFFECTS OF COMBINED ALTERNATIVE CONTAINMENT FEATURES
(Reference Fuel Quality, Bounding Prompt Source Term)

Case (a)	Set (b)	Filter DF (c)	Filter Bypass (%)[e]	EAB (m)	Stack	Contain. Leak Rate vol.%/d	0-30 day Thyroid Dose (Rem)	0-30 day W. Body Dose (Rem)
10-8	A[d]	1	N/A	425	No	Vented	0.660	2.48×10^{-3}
10-9	B	3	10	425	No	Vented	0.635	2.40×10^{-3}
10-10		3	1	805	No	Vented	0.320	1.21×10^{-3}
10-12	C	30	1	425	No	Vented	0.620	2.35×10^{-3}
10-12		30	1	425	Yes	Vented	1.08×10^{-2}	4.13×10^{-5}
10-12	D	30	1	805	No	Vented	0.313	1.19×10^{-3}
10-12	E	30	1	805	Yes	Vented	8.18×10^{-3}	3.12×10^{-5}
10-13	F	N/A	N/A	425	N/A	5%/d	5.59×10^{-3}	2.56×10^{-5}
10-13		N/A	N/A	805	N/A	5%/d	2.18×10^{-3}	1.24×10^{-5}
10-14		N/A	N/A	425	N/A	0.5%/d	5.61×10^{-4}	2.62×10^{-6}
10-14	G	N/A	N/A	805	N/A	0.5%/d	2.81×10^{-4}	1.25×10^{-6}

(a) Case numbers are related to case identification in supporting calculations.
(b) Set numbers are related to the Feature Sets in the Executive Summary.
(c) Decontamination factors are applied to halogens and particulates. The filter does not retain any noble gasses.
(d) Feature Set A is the reference VLPC design, without any additional filters.
(e) Approximately 90% of the prompt source term bypasses the filter, since the pressure transient is sufficient to open the primary building vent.

The analytical methods used to assess these cases is the same as that used for the reference plant, as described in Section 4. The analytical model is the same as shown in Figure 7-1, and includes the features described in Section 6. For all of the vented cases, the building leak rate is kept constant at 1 volume per day. The results from Table 8-11 show only very small effects of the filters. In both Set B and set C the large helium leak causes the main building vents to open and most of the blowdown will bypass the filters. Most of the dose from SRDC-10 is a result of the release which occurs during the first few minutes of the event.

8.4 PLANT RESPONSE TO BOUNDING PROMPT SOURCE TERM WITH LOWER LEVEL FUEL QUALITY

8.4.1 Response of the Reference VLPC design

The analytical methods used to assess the site boundary doses are again the same as were used for the reference plant, as described in Section 4. The analytical model is the same as shown in Figure 7-1, with the building leak rate kept constant at 1 volume per day. The calculated radiation exposure to an individual at the site boundary with the bounding prompt source term are shown in Table 8-12.

Table 8-12
SITE BOUNDARY CONSEQUENCES OF LIMITING EVENTS - 450 MWt MHTGR
LOWER LEVEL FUEL QUALITY AND BOUNDING PROMPT SOURCE TERM ASSUMED
(50th Percentile Doses)

	Release Events		
	Rapid Depressurization and loss of Forced Cooling (SRDC-10)	Depressurized loss of Forced Cooling (SRDC-11)	Water Ingress with Depressurized loss of Forced Cooling (SRDC-6)
Best Estimate Prompt Source Term			
Thyroid Dose (Rem)	0.148	1.35	19.5
W. Body Dose (Rem)	2.81×10^{-3}	3.88×10^{-3}	0.101
Bounding Prompt Source Term			
Thyroid Dose (Rem)	6.60	5.47	22.0
W. Body Dose (Rem)	2.48×10^{-2}	1.67×10^{-2}	0.109

8.4.2 Response of Candidate Containment Design Strategies to a Small Helium Leak

According to past studies, events which lead to radiological release via a small helium leak in the RCPB, typified by SRDC-11, are risk dominant. This class of events is treated by the proposed filter release pathway from the reactor building, and by strategies which would change the site and release conditions. SRDC-11 has been analyzed for the reference quality fuel with a bounding prompt source term for the same range of alternative containment features and feature combinations as described in Section 6. Table 8-13 shows the results of these assessments.

Table 8-13
MHTGR RESPONSE TO SMALL HELIUM LEAK EVENTS (SRDC-11)
EFFECTS OF COMBINED ALTERNATIVE CONTAINMENT FEATURES
(Lower Level Fuel Quality, Bounding Prompt Source Term)

Case (a)	Set (b)	Filter DF (c)	Filter Bypass (%)	EAB (m)	Stack	Contain. Leak Rate vol.%/d	0-30 day Thyroid Dose (Rem)	0-30 day W. Body Dose (Rem)
11-22	A[d]	1	N/A	425	No	Vented	5.47	1.67×10^{-2}
11-23	B	3	10	425	No	Vented	2.59	8.95×10^{-3}
11-24		3	1	805	No	Vented	1.19	4.32×10^{-3}
11-26	C	30	1	425	No	Vented	0.882	4.36×10^{-3}
11-26		30	1	425	Yes	Vented	2.5×10^{-2}	1.37×10^{-4}
11-26	D	30	1	805	No	Vented	0.456	2.32×10^{-3}
11-26	E	30	1	805	Yes	Vented	1.61×10^{-2}	8.63×10^{-5}
11-27	F	N/A	N/A	425	N/A	5%/d	4.74×10^{-2}	2.43×10^{-4}
11-27		N/A	N/A	805	N/A	5%/d	2.46×10^{-2}	10.5×10^{-4}
11-28		N/A	N/A	425	N/A	0.5%/d	4.25×10^{-3}	2.67×10^{-5}
11-28	G	N/A	N/A	805	N/A	0.5%/d	2.47×10^{-3}	1.08×10^{-5}

(a) Case numbers are related to case identification in supporting calculations.
(b) Set numbers are related to the Feature Sets in the Executive Summary.
(c) Decontamination factors are applied to halogens and particulates. The filter does not retain any noble gasses.
(d) Feature Set A is the reference VLPC design, without any additional filters.

8.4.3 Response of Candidate Containment Design Strategies to Water Ingress Event

Events which lead to radiological release via the helium relief valve train, typified by SRDC-6, are the most severe in terms of the magnitude of individual event doses at the site boundary. This class of events is treated by the proposed filter train on the helium relief lines, and by strategies which would change the site and release conditions. SRDC-6 has been analyzed for the reference quality fuel with a bounding prompt source term for the same range of alternative containment features and feature combinations as described in Section 6. Table 8-14 shows the results of these assessments.

8.4.4 Response of Candidate Containment Design Strategies to Large Helium Leak

According to past studies, events which lead to radiological release via a large helium leak in the RCPB, typified by SRDC-10, are the most prompt. This class of events are typically treated by strategies which would change the site and release conditions. Even the largest proposed alternative filter release pathway from the reactor building will have very little effect, because most of the released gasses will escape through the main building vent. SRDC-10 has been analyzed for the lower limit fuel quality with a bounding prompt source term for the same range of alternative containment features and feature combinations as described in Section 6. Table 8-15 shows the results of these assessments.

Table 8-14
MHTGR RESPONSE TO WATER INGRESS EVENTS (SRDC-6)
EFFECTS OF COMBINED ALTERNATIVE CONTAINMENT FEATURES
(Lower Level Fuel Quality, Bounding Prompt Source Term)

Case (a)	Set (b)	Filter DF (c)	EAB (m)	Stack	Contain. Leak Rate vol.%/d	0-30 day Thyroid Dose (Rem)	0-30 day W. Body Dose (Rem)
6-25	A[d]	N/A	425	No	Vented	22.0	0.109
6-26	B	10	425	No	Vented	4.56	3.67×10^{-2}
6-27	C	100	425	No	Vented	0.456	2.09×10^{-2}
6-27	D	100	805	No	Vented	0.300	1.49×10^{-2}
6-27	E	100	805	Yes	Vented	1.68×10^{-2}	9.36×10^{-4}
6-28		100[e]	425	Yes	Vented	2.96×10^{-2}	1.68×10^{-3}
6-28		100[e]	805	No	Vented	0.295	1.47×10^{-2}
6-29	F	N/A	425	N/A	5%/d	4.69×10^{-2}	7.89×10^{-4}
6-29		N/A	805	N/A	5%/d	2.46×10^{-2}	2.71×10^{-4}
6-30		N/A	425	N/A	0.5%/d	4.70×10^{-3}	9.23×10^{-5}
6-30	G	N/A	805	N/A	0.5%/d	2.47×10^{-3}	7.75×10^{-5}

(a) Case numbers are related to case identification in supporting calculations.
(b) Set numbers are related to the Feature Sets in the Executive Summary.
(c) Decontamination factors are applied to halogens and particulates. The filter does not retain any noble gasses.
(d) Feature Set A is the reference VLPC design, without any additional filters.
(e) Case 6-28 uses DF = 100 for halogens, DF = 1000 for particulates.

Table 8-15
MHTGR RESPONSE TO LARGE HELIUM LEAK EVENTS (SRDC-10)
EFFECTS OF COMBINED ALTERNATIVE CONTAINMENT FEATURES
(Lower Level Fuel Quality, Bounding Prompt Source Term)

Case (a)	Set (b)	Filter DF (c)	Filter Bypass (%)[e]	EAB (m)	Stack	Contain. Leak Rate vol.%/d	0-30 day Thyroid Dose (Rem)	0-30 day W. Body Dose (Rem)
10-22	A[d]	1	N/A	425	No	Vented	6.60	2.48×10^{-2}
10-23	B	3	10	425	No	Vented	6.35	2.40×10^{-2}
10-24		3	1	805	No	Vented	3.20	1.21×10^{-2}
10-26	C	30	1	425	No	Vented	6.20	2.35×10^{-2}
10-26		30	1	425	Yes	Vented	0.108	4.13×10^{-4}
10-26	D	30	1	805	No	Vented	3.13	1.19×10^{-2}
10-26	E	30	1	805	Yes	Vented	8.18×10^{-2}	3.12×10^{-4}
10-27	F	N/A	N/A	425	N/A	5%/d	5.59×10^{-2}	2.56×10^{-4}
10-27		N/A	N/A	805	N/A	5%/d	2.81×10^{-2}	1.24×10^{-4}
10-28		N/A	N/A	425	N/A	0.5%/d	5.61×10^{-3}	2.62×10^{-5}
10-28	G	N/A	N/A	805	N/A	0.5%/d	2.81×10^{-3}	1.25×10^{-5}

(a) Case numbers are related to case identification in supporting calculations.
(b) Set numbers are related to the Feature Sets in the Executive Summary.
(c) Decontamination factors are applied to halogens and particulates. The filter does not retain any noble gasses.
(d) Feature Set A is the reference VLPC design, without any additional filters.
(e) Approximately 90% of the prompt source term bypasses the filter, since the pressure transient is sufficient to open the primary building vent.

The results from Table 8-15 show almost no effect of the filters. In both Set B and Set C the large helium leak causes the vents to open and bypass the filters. About 10% of the prompt release, and all of the delayed release remain inside the reactor building after the main vents reclose, and the filters are able to have an effect on the fission products entrained in the gas which is captured in the building.

9. SUMMARY, CONCLUSIONS, AND RECOMMENDATIONS

The Modular HTGR relies on a design strategy which is based more on the prevention of challenging events rather than on their mitigation. The selection of key materials and parameters lead to a low power density reactor with a very large heat capacity, which in turn results in transients which occur over a time frame of days. The use of a graphite moderator, an uninsulated steel reactor vessel, and a passive decay heat rejection system make the system tolerant of reactor coolant leaks. The selection of a ceramic coating system and inert coolant make core melting events impossible. The simplicity and robustness of the design is based on a goal of relieving the plant operator from the burden of required actions during emergencies and also of making the MHTGR tolerant of any operator mis-actions.

The MHTGR design is required to meet licensing requirements and utility/user goals, including the requirement that the plant avoid any intrusion into the normal day-to-day activities of the public. This goal requires that the offsite dose from any event shall be less than the lower threshold radiation dose at which the public would be required to evacuate or take shelter. The US Environmental Protection Agency has set this goal at 5 Rem to the thyroid and 1 Rem to the whole body. This goal has also been interpreted to mean that the frequency with which any event might result in a dose at the plant boundary equal to the PAG limit should be very low. The MHTGR designers have followed the method used in the light water reactor industry to establish an Emergency Planning Zone (EPZ) but with the zone at the plant boundary for all events above a frequency of 5×10^{-7} per plant-year.

The MHTGR fuel is designed to rigorous standards. The specification for acceptable fuel quality allows for only a minute fraction of defective fuel particles and fissile material outside of coatings. When this fuel quality is met, calculations indicate that the proposed reference 450 MW(t) MHTGR can meet nearly all of its goals. Individual events result in dose consequences far below the legal requirements listed in 10CFR100, when analyzed at the 95th percentile. Individual events are also below the 5 Rem thyroid, 1 Rem whole body dose requirements from the Protective Action Guidelines, when analyzed at the 50th percentile. The reference design meets the goal that the dose for all events at an event frequency of 5×10^{-7} per plant year be less than or equal to the whole body PAG dose of 1 Rem, but does not meet the thyroid PAG dose of 5 Rem. There are several aspects to this goal and to the method used to calculate the plant's performance which require further study and which may reverse this latter conclusion.

9.1 CONTAINMENT ALTERNATIVES ASSUMING REFERENCE FUEL QUALITY

There are a number of design alternatives which the MHTGR could employ to enhance fission product retention while continuing to utilize a VLPC. These alternatives rely on passive features which are consistent with the overall plant design goals. While increasing the burden on the plant operator by requiring that the additional equipment be adequately maintained, no actions or decisions are required by the plant operating staff in order to implement any of the identified alternatives during upset conditions. The capital costs of the various alternatives were examined briefly during the Cost Reduction Study (Ref. 7). Neither the capital costs, nor the operation and maintenance implications of the VLPC alternative features reviewed are expected to have a significant negative impact on the operability or cost of power generated by the MHTGR.

The addition of a filtered pathway to the reactor building boundary is expected to contribute to reduced doses due to dry helium leaks, represented by SRDC-11. The design of the vented building and the passive RCCS result in a restricted operating pressure range and allowable pressure loss across the bed of such a filter, with attendant limitations on the achievable decontamination factors. Individual event dose consequences were acceptable without this addition, and its addition may be sufficient to reduce the dose at event frequencies of 5×10^{-7} per plant-year to the 5 Rem thyroid dose goal. This alternative feature is a viable means to reduce offsite doses in a very reliable manner.

The connection of a filtered pathway from the helium relief valve discharge point to the environment would contribute to reduced offsite doses from water ingress events, represented by SRDC-6. Again, the individual events meet the design goals without this feature. However, this feature would have several advantages, and only modest cost implications, if incorporated into the plant design for other reasons. It would improve the safety of plant operators by eliminating the possibility of a relief valve discharge of hot gasses into the reactor building space which could burn or suffocate personnel in the area; and it could reduce the operator exposure to in-plant radiation by eliminating the possibility of a discharge into the building which would result in contamination of the building interior. For both these reasons, it is recommended that the helium relief valve filter train be included in the 450 MW(t) MHTGR design.

With the combination of a filter on a relief line that discharges to the environment and a filter on the vented containment, the relatively clean, pressurized primary coolant can be released to the environment eliminating the transport mechanism for any subsequent releases from the core which can then be effectively filtered.

Options relating to an increase in the Exclusion Area Boundary (EAB) and the addition of a tall stack are clearly possible. Either option alone would have the result of lowering the dose from any event at the site boundary. The reduction afforded by either option alone would probably be sufficient to reduce the dose at 5 x 10^{-7} per plant-year to the 5 Rem goal, and support the co-location of the EAB and the EPZ. These alternatives could have a negative impact on the marketability of the MHTGR, and it is recommended that they not be incorporated at this time.

9.2 SOURCE TERM ALTERNATIVES

The reference fuel quality specification and fuel performance, if successfully demonstrated, allows a simple approach to fission product retention. It is clear that the vented, low pressure containment can successfully protect the health and safety of the offsite public if the reference fuel quality is met. If the as-manufactured fuel quality is lower, modest additions to the containment system allow the MHTGR to continue to meet its offsite dose goals for individual events. The lower limit on fuel quality represents approximately a ten-fold increase in the size of accident source terms. Even with this lower limit in fuel quality, 10CFR100 requirements would be met; however, a filter on the relief valve train would be needed in order to meet the utility/user dose goal of 5 Rem thyroid at 5 x 10^{-7} per plant-year.

If it is assumed that the rate of hydrolysis is much higher than expected, and the hydrolysis of exposed kernels occurs in the first few minutes of a water ingress event, but reference fuel quality is assumed, the MHTGR can still meet the PAG limits without credit for a filter on the relief valve exhaust. However, if both lower quality fuel and rapid hydrolysis are assumed, it requires a filter with a decontamination factor of at least 100 for halogens and particulates in order to meet the PAG limit.

If a breakdown in the fuel quality control process is assumed, and it is hypothesized that a batch of "weak fuel" is operated in the reactor, an increased source term has been estimated. The size of the source term is highly dependent on the assumptions regarding the nature and manifestation of the defect. However, if the weak fuel source term assumed in this study is released due to a primary coolant leak (represented by SRDC-11), the estimated dose consequences are well below 10CFR100. If the VLPC is augmented with building filters and either an increase in the site boundary to 805 meters or an elevated release point, the incremental dose due to the weak fuel will be less than the PAG limits. The slow release and passive heat rejection inherent in the MHTGR design both limit the transient and permit a vented building strategy to deal with such a bounding event scenario.

If the prompt source term due to release of plateout activity is assumed to be 100% with reference fuel quality, sensitivity studies show that for both helium depressurizations up to the design basis primary coolant leak (13 sq. in.) and steam generator leaks that a VLPC can limit doses to below the PAG limit. Even if this hypothetical release of plateout is coupled with the characteristics of lower fuel quality, 10CFR100 can be met with a large margin.

9.3 CONCLUSIONS

The VLPC strategy is superior to a conventional, pressure retaining containment for the MHTGR. With the addition of a filter to the relief train, the events which cause the highest doses at the site boundary are all events in which the fission product release is accompanied by a low rate of blowdown energy release. When this is the case, it is more effective to filter the release, rather than contain it. The VLPC has the following additional advantages over a conventional pressure retention containment:

- By releasing the blowdown energy, the VLPC dissipates fission product transport energy.
- The VLPC has fewer fault modes, fewer active components, and requires less operator supervision and vigilance than a conventional containment.
- The VLPC is compatible with a highly reliable, passive residual heat rejection design.
- The VLPC is lower in cost than a conventional containment strategy.

Continuing the design process with a VLPC strategy is acceptable. A decision to incorporate additional features including a filtered reactor building exhaust, increase in the EAB distance, or the addition of a tall stack can be deferred until additional studies are complete and fuel and fission product developments are further advanced.

9.4 RECOMMENDATIONS

This report recommends the following:

- The MHTGR Program should undertake a comprehensive review of the basis and methodology used to calculate the cumulative dose at low frequency. It is likely that the weather model, which dominates the low frequency part of the individual event curves, requires revision. It is also potentially unreasonable to use 5×10^{-7} per plant year as the event frequency for both the thyroid dose and the whole body dose. The calculations and conclusions in this report should be revisited after that review.

- The fuel and fission product testing should continue to be directed toward definitively validating the design basis fuel quality and source terms established during the PSID review.

- The fuel manufacturing process should continue to be directed toward providing fuel of the design basis fuel quality established during the PSID review. The development should also address methods needed to assure exceedingly low probabilities of a breakdown of process or quality control leading to fission product release during normal operation or accident conditions in excess of the design basis.

- The proposed 450 MW(t) MHTGR design should include a filter train connected to the helium relief valve discharge points. This filter should be located outdoors and discharge to the environment. It will improve plant operator exposure and safety and provide additional margin against alternative source terms.

- The need for a filter on the vented containment will be determined as the design and technology programs progress.

- The alternative features related to the site boundary distance and the inclusion of a tall stack must be evaluated in the overall context of MHTGR marketability and user needs. Neither alternative is required to meet any individual event radiological goal, but either would enable the MHTGR to meet the cumulative PAG risk goal.

10. REFERENCES

1. *Preliminary Safety Information Document for the Standard MHTGR*, U.S. Department of Energy report DOE-HTGR-86024, September 1986.

2. *Probabilistic Risk Assessment for the Standard Modular High Temperature Gas-Cooled Reactor*, DOE report DOE-HTGR-86011, Rev. 5, April 1988.

3. Williams, P. M., et al., *Draft Preapplication Safety Evaluation Report for the Modular High-Temperature Gas-Cooled Reactor*, NUREG-1338, March 1989.

4. *Containment Study for MHTGR*, DOE report DOE-HTGR-88311, November 1989.

5. Morris, B. M., Letter to Dr. Sol Rosen, May 9, 1990, included in the Appendix of this report.

6. *Manual of Protective Action Guides and Protective Actions for Nuclear Incidents*, United States Environmental Protection Agency, EPA 400-R-92-001, October 1991.

7. *MHTGR Cost Reduction Study*, DOE report DOE-HTGR-88512, October 1990.

8. *Top Level Regulatory Criteria for the Standard HTGR*, U.S. Department of Energy document HTGR-85-002, Rev. 3, September 1989.

9. *Utility/User Requirements for the Modular High Temperature Gas-Cooled Reactor Plant*, Gas-Cooled Reactor Associates, GCRA 86-002, Rev. 6, October 1990.

10. *Licensing Basis Events for the Modular HTGR*, DOE report DOE-HTGR-86034, Rev. 1, February 1987.

11. *US/FRG Accident Condition Fuel Performance Models*, DOE report DOE-HTGR-85107, Rev. A, March 1989.

12. *Overall Plant Design Specification Modular High Temperature Gas-Cooled Reactor*, DOE report DOE-HTGR-86004, May 1990.

APPENDIX

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

Dr. Sol Rosen, Director
Advanced Reactor Programs
Office of Civilian Reactor
Development
U.S. Department of Energy
Washington, DC 20545

Dear Dr. Rosen:

We are currently reviewing your report entitled "Containment Study for MHTGR" DOE-HTGP-88311 (November 1989) as well as certain earlier documents. An important concern emerging from this review is that the two source terms now used for the containment design basis in your report do not appear to adequately span the range of credible possibilities and, moreover, may not be appropriately conservative. Specifically, the source terms are: (1) a prompt and low-level release of fission products from a rapid depressurization event characterized by lift-off of a fraction of the inventory of radionuclides plated-out in the primary system, and a low-level release of fission products from rapid fuel kernel hydrolysis when moisture ingress occurs and (2) a delayed and somewhat greater release of fission products from fuel kernel hydrolysis caused by core heatup in combination with a moisture ingress event. We believe that both the prompt and delayed source terms should be revised. We believe that the prompt source term should include all those radionuclides that could be released by fuel kernel hydrolysis. Our basis for considering fuel hydrolysis effects in the prompt source term is that the time potentially available prior to reactor depressurization following a steam generator failure might be sufficient to permit essential completion of the hydrolysis chemical reaction. The time estimates you currently use for hydrolysis (hours to days) have not been confirmed experimentally. In addition, such an augmented prompt source term would also encompass the contingency that the liftoff release fraction and plateout inventory of fission products could be greater than currently proposed.

With regard to the delayed source term, we believe that it should provide for the contingency that additional and substantial delayed fuel failures and a significant release of radionuclides during a core heat-up event might result from undetected sub-specification fuel. Although several mechanisms could be considered for additional fuel failure, the most apparent mechanism for sub-specification fuel failure would be from "weak-fuel" as discussed in Section 4.2.5.D of the Draft Preapplication Safety Evaluation Report for the MHTGR (NUREG-1338). We would define sub-specification fuel as defective fuel that is not detected by the manufacturing quality assurance program and remains undetected during normal operation. However, it would be capable of causing an unexpected fission product release during a core heat-up event, which could be aggravated by moisture ingress.

We understand that your current position is that sub-specification fuel need not be considered in the containment study on the grounds of expected success in research for both fuel performance and the quality assurance program for manufacturing. We, of course, encourage the development of a very high quality nuclear fuel and recognize its advantage. However, we believe that recognition and analysis of an augmented delayed source term based on a sub-specification fuel failure model with core heat-up and moisture ingress is necessary at this time.

We are performing an independent assessment to illustrate the consequences of the revised source terms described herein. We request that you perform a similar analysis employing the source terms revised as suggested above. In particular, we believe it would be useful if you were to reinvestigate your design alternatives and evaluate the possible benefits of additional containment design features. I have enclosed background and general guidance for this analysis as well as other requests for information pertaining to the containment adequacy issue. In order to complete our revision of NUREG-1338, "Draft Preapplication Safety Evaluation Report for the Modular High-Temperature Gas-Cooled Reactor," I will need to receive this information to the maximum extent possible before June 30, 1990. If you have any questions on the enclosure, please contact Dr. P. Williams at 301-492-3736.

Sincerely,

Bill M. Morris, Director
Division of Regulatory Applications
Office of Nuclear Regulatory Research

Enclosure:
Request for Additional Information

cc: J.A. Kendall, GCRA

Request for Additional Information

MHTGR Containment Adequacy Issue

1. The building alternative study given in DOE-MHTGR-88311 should be repeated using revised quantities for both prompt and delayed fission product sources. The revised prompt source term should include those radionuclides that could be released by fuel kernel hydrolysis. This revision should recognize recent experiments indicating moisture levels that could accompany a steam generator failure. Because the time estimates you currently use for hydrolysis have not been confirmed experimentally, you should assume that hydrolysis could take place in minutes and during the time interval before the relief valve could open prior to blow down. The revised prompt source term should also include a parametric treatment of the plateout fission product inventory to address the contingency that this inventory could be greater than currently expected.

   The revised delayed source term should consider fission product quantities based on a sub-specification fuel model. This model would address the contingency that additional and substantial release of radionuclides during core heatup events, both dry and wet, might result from sub-specification fuel. Sub-specification fuel is described in NUREG-1338 in Section 4.2.5.D under the term "weak fuel." At this stage of our review, we cannot conclude that all sub-standard fuel will be accounted for by the manufacturing QA/QC program for the operating life of a large number of MHTGRs.

In considering the delayed source term provide a parametric study characterizing the composition and magnitude of the source term at various stages (i.e., at barriers such as the fuel particle surface, fuel rod matrix, graphite block, fuel assembly, primary system boundary, stages of the "containment/confinement system" and to the environs) as a function of the fractions of a full core which has failed as a result of fabrication defects below the specified standard (i.e., 25%, 50%, 75%, etc.). Indicate the factors which would support a selection of one of these as an appropriate "mechanistic source term" with margin to account for uncertainties.

All calculations should be sufficiently detailed to identify the initial availabilities and the significant mechanisms, barriers, assumptions, and credits for the attenuation of fission products as they progress to the environs. Best-estimate models and assumptions should be used with conservative treatment of uncertainties. The outputs of the revised study should include pressure and temperature duties for each essential confinement or containment component as a function of time. The calculations should be realistic so as not to obscure the practical values and limitations of any mechanism or barrier.

2. For Building Alternatives 2 through 5 the quality classification requirements of the equipment to be installed (filters, dampers, etc.) should be identified. It should be established how designing for the revised prompt and delayed source terms would affect the estimates presented and discussed in Tables 4.8-2 and 4.8-3, respectively.

3. Discuss the characteristics of driving pressures for radionuclides (include moisture ingress events) for leakage through sealed and isolated building alternates as a function of time. Would there be an opportunity and would it be worthwhile to delay containment closure, such as provided by Alternate 3, either automatically or manually to reduce the driving pressure?

4. Will the requirements to consider the revised source terms result in changes in the design of any of the building alternates presented? If so, please present a revised conceptual design or designs for our review.

5. When will additional research information become available providing release fractions of fission product inventory in the fuel and primary circuit and supporting your present position that sub-specification fuel or other means of significant fuel failure need not be considered as a containment design basis? When will the research program be fully completed in terms of the start-up date of the first MHTGR module?

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