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SAFETY AND LICENSING OF MHTGR

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Abstract

The Modular High Temperature Gas Cooled Reactor (MHTGR) design meets stringent top-level regulatory and user safety requirements that require that the normal and off-normal operation of the plant not disturb the public's day-to-day activities. Quantitative, top-level regulatory criteria have been specified from US NRC and EPA sources to guide the design. The user/utility group has further specified that these criteria be met at the plant boundary. The focus of the safety approach has then been centered on retaining the radionuclide inventory within the fuel by removing core heat, controlling chemical attack, and by controlling heat generation. The MHTGR is shown to passively meet the stringent requirements with margin. No operator action is required and the plant is insensitive to operator error.
(Key words: coated fuel particles, emergency planning, integrated approach, licensing, Modular HTGR, passive features, plant and public protection, safety design, top level regulatory and user requirements)

1 Philosophy and Approach

The overall philosophy guiding the design of the Standard MHTGR may be stated, in its simplest terms, as follows:

> Produce a safe, economical plant design which meets regulatory and user requirements by providing defense-in-depth through the pursuit of four goals:
>
> 1. Maintain Plant Operation
>
>   Reliably maintain the functions necessary for normal plant operation, including the plant states of energy production, shutdown, refueling, and startup/shutdown operations.
>
> 2. Maintain Plant Protection
>
>   Assume that despite the care taken to maintain plant operation, failures will occur and provide additional design features or systems to ensure availability and provide investment protection by preventing plant damage.

3. Maintain Control of Radionuclide Release

   Provide additional design features or systems to assure containment of radionuclides in the event normal operating conditions cannot be maintained and/or plant protection is not assured.

4. Maintain Emergency Preparedness

   Maintain adequate emergency preparedness to protect the health and safety of the public in the event control of radionuclide release is not accomplished.

With regard to the achievement of US Nuclear Regulatory Commission (NRC) criteria for the accomplishment of Goal 1 and 2 functions, measures are taken in the design of the Standard MHTGR, to use inherent characteristics and high quality fuel, to minimize release of radionuclides so that normal operational releases or any accidental releases of primary circuit activity are low and worker exposures are minimized. These techniques have been proven to be effective in other gas-cooled reactors as has been demonstrated by measuring releases and worker exposures from operating plants.

The unique aspect of the Standard MHTGR, however, is the approach which has been taken to achieve Goal 3. To accomplish this goal with high assurance, the design of the Standard MHTGR has been guided by the additional philosophy that control of radionuclide releases be accomplished by retention of radionuclides within the fuel particles with minimal reliance on active design features or operator actions. The overall intent is to provide a simple approach to safety that will provide high confidence that the Goal 3 safety criteria are met. There are two key elements to this philosophy which have had a profound impact on the design of the Standard MHTGR (especially in the selection of core size and geometry, power density, and vessel type); the basis for each element is described below.

First, the philosophy requires that control of radionuclides be accomplished with minimal reliance on active systems or operator actions; the approach to safety is to rely primarily on the natural processes of thermal radiation, conduction, and convection, and on the integrity of passive design features. Arguments need not center on an assessment of the reliability of pumps, valves and their associated services or on the probability of an operator taking various actions, given the associated uncertainties involved in such assessments.

Second, the philosophy requires control of releases by the retention of radionuclides primarily within the coated fuel particle rather than reliance on secondary barriers (such as the primary coolant boundary or reactor building). Thus, assuring that the safety criteria are met is the same as assuring that the retention capability of the coated fuel particles is not compromised. Three functions have been identified which, when accomplished, assure that radionuclide retention within the fuel remains acceptable:

1. Control Heat Generation
2. Remove Core Heat
3. Control Chemical Attack

If the proof of containment is dramatically simplified, arguments can center on issues associated with fuel particle coating integrity when these functions are accomplished (Reference 1).

The following section describes the method which has been employed to ensure the consistent incorporation of this safety philosophy into the design.

2. Design Approach

The plant configuration and features are established within the framework of the four Goals defined earlier, in accordance with Utility/User Requirements and Regulatory criteria.

The Utility/User Requirements are established through Gas-Cooled Reactor Associates, the utility/user group that supports the development and commercialization of the HTGR. These requirements (Reference 2) are intended to provide a basis for HTGR development responsive to the needs of the electric generation market as perceived by the end user.

The Top-Level Regulatory Criteria for the MHTGR were developed through a detailed review of current regulatory requirements and policy statements for criteria which are: (1) direct statements of acceptable health and safety consequences or risks to individuals or the public; (2) independent of reactor type and site; and (3) are quantifiable. Table 1 provides a summary of the Top-Level Rgulatory Criteria selected for the MHTGR; the complete listing of requirements may be found in Reference 3.

The technical management methodology used to direct the development of the MHTGR design is called the Integrated Approach. It is a systematic process of developing the functions, requirements, and design selections at increasingly mor detailed design levels to meet the Top-Level Regulatory Criteria and the Utility/User Requirements and attain the objective of safe, economic nuclear power.

An essential feature of the Integrated Approach is functional analysis, which is a process of ordering hierarchical plant functions. The evolving design is guided with other analytical methodologies which include probabilistic risk assessments, reliability evaluations, and economic comparisons. The Standard MHTGR design which resulted from this approach is described in the Preliminary Safety Information Document (PSID) (Reference 4) and summarized in the following section.

3. Safety Features

The safety features of the Standard MHTGR are dominated by the safety characteristics common to all HTGRs as well as features unique to the particular configuration of the Standard MHTGR module. The general safety characteristics of an HTGR design tend to be dominated by the inherent characteristics of the coolant, core materials, and fuel as described below.

TABLE 1
SUMMARY OF TOP-LEVEL REGULATORY CRITERIA

Criteria Type	Criteria
Public Risk	NRC Safety Goals - Risk of prompt fatality less than one-tenth of one percent of the sum of prompt fatality risks resulting from other causes - Risk of cancer fatalities less than one-tenth of one percent of the sum of cancer fatality risks resulting from other causes
Dose Limits	40 CFR 190 limits - Whole body annual dose from entire fuel cycle less than 0.025 rem whole body, - less than 0.075 rem thyroid, or less than 0.025 rem any organ.
	10 CFR 50 Appendix I doses on an expected, annualized basis - dose from gaseous pathway less than 0.005 rem whole body or 0.015 rem any organ, - dose from liquid pathway less than 0.003 rem whole body or 0.010 rem any organ, - dose from particulates less than 0.015 rem any organ.
	10 CFR 20 limits - less than 0.5 rem whole body/year - less than 0.002 rem in any one hour - less than 0.1 rem in any 7 consecutive days.
Annual Dose Limits for Anticipated Operational Occurrences	10 CFR 50 Appendix I doses on an expected, annualized basis
Dose Limits for Accidents	10 CFR 100 Guidelines - less than 25 rem whole body - less than 300 rem thyroid
Dose Limits Triggering Offsite Sheltering or Evacuation	EPA 520 Guidelines - sheltering: one to five rem whole body five to 25 rem thyroid - evacuation: greater than 5 rem whole body greater than 25 rem thyroid

- HELIUM COOLANT -- The inert and single phase helium coolant has several advantages. No flashing or boiling of coolant is possible, pressure measurements are certain, no coolant level measurements are required and pump cavitation cannot occur. Further, there are no reactivity effects associated with the helium and no chemical or energetic reactions between coolant and fuel or cladding is possible.

- GRAPHITE CORE -- The strength of the graphite core and the stability of the ceramic fuel coating at high temperatures result in a wide margin between operating temperatures and temperatures that would result in core damage. Further, the high heat capacity and low power density of the core result in very slow and predictable temperature transients.

- COATED FUEL PARTICLE -- The multiple ceramic coatings surrounding the fuel kernels constitute tiny independent pressure vessels which contain fission products. These coatings are capable of maintaining their integrity to very high temperatures in excess of 2000°C temperature range.

The special features unique to the modular HTGR include the sizing and configuration that take advantage of these inherent HTGR characteristics so that minimal reliance need be placed on active or powered systems or operator actions to accomplish safety functions. Specifically, the geometry and size of the reactor core, its power density, and the uninsulated steel vessel have been selected to allow for decay heat removal from the core to the ultimate heat sink through the natural processes of radiation, conduction and convection. In addition, fuel enrichment values are selected which provide a negative temperature coefficient. This inherently compensates for positive reactivity insertions by self-terminating the nuclear reaction as the core heats up. As a result, the Standard MHTGR can withstand a loss of helium coolant in combination with the loss of all forced circulation from full power without fuel temperatures exceeding a level at which significant incremental fuel particle failure would be observed. Reference 5 provides additional information on the MHTGR core.

Thus, by combining the inherent characteristics common to all HTGRs with special features unique to the MHTGR design, an enhanced level of safety is provided by the Standard MHTGR.

4. Safety Assessment

Licensing Basis Events are selected utilizing a Probabilistic Risk Assessment (PRA) to judge compliance with the Top-Level Regulatory Criteria. Figure 1 presents the frequency regions for each of the limiting criteria. The evaluation of the LBEs in the top two regions, the Anticipated Operational Occurrences and the Design Basis Regions is presented in the PSID. The PRA (Reference 6) evaluates the events in the Emergency Planning Basis Region. The limiting events and their doses examined in the PSID are summarized in the following section.

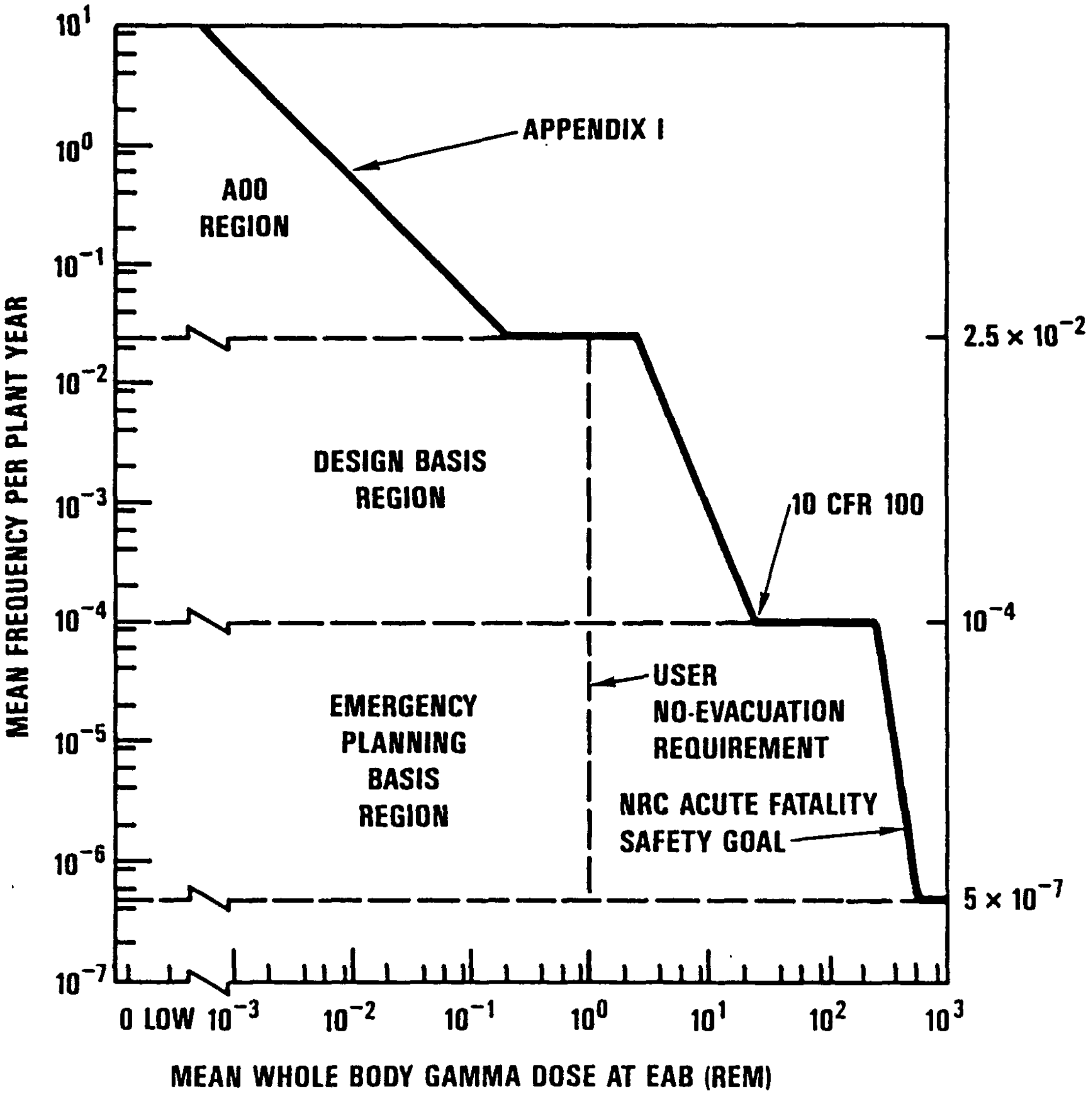

Figure 1 Licensing Basis Regions for the MHTGR

4.1 Accident Doses

The MHTGR design is analyzed in the PSID to assess its adequacy to perform the functions necessary to control accidental radioactivity releases to within the offsite dose limits of 10CFR100.

4.1.1 Mechanistic Design Basis Events

To accomplish this, two classes of analysis are provided. First, a series of design basis events (DBE) is analyzed. As shown in Figure 1, design basis events are those sequences of events which are not expected to occur in the lifetime of a single Standard MHTGR plant, but would be expected to occur in the lifetime of several hundred plants. These design basis events are treated mechanistically, with radiological consequences assessed statistically. Both the median and the upper 95th percentile radiological consequences are provided. Even for the upper 95th percentile consequence there is a large margin between the assessed doses and the guidelines of 10CFR100.

4.1.2 Deterministic Design Conditions

To further ensure with the highest confidence that the design meets the guidelines of 10CFR100, a more bounding analysis is provided. Consistent with the safety philosophy that reliance be placed to the maximum extent on inherent or passive means to protect the public health and safety, design conditions are postulated for the passive features that are relied on to meet 10CFR100. A deterministic assumption is made that those SSCs which are not relied on during the DBEs are unavailable. In essence, the resulting design conditions are even lower probability events than the DBEs. The radiological consequences of these deterministic design conditions are analyzed statistically. The results are summarized in Table 2. It may be seen from the table that even with reliance only on the passive or inherent features of the design, even the upper 95th percentile dose assessments meet the guidelines of 10CFR100 with considerable margin.

The limiting events analyzed in the PSID challenging each function and the resulting consequence is summarized in the following sections.

4.2 Control Heat Generation Events

Two key events are considered in the PSID with regard to the ability to control heat generation. The first involves a loss of the normal cooling path using the Heat Transport System (HTS) without a control rod trip (such an event is commonly referred to as an anticipated transient without scram or ATWS) and the second involves a control rod withdrawal.

The function of controlling core heat is challenged in the first event when, with the reactor initially at 100 percent power, the HTS experiences a transient which causes it to trip. Outer control rod insertion, which normally results automatically from a loss of the HTS, fails. Additionally, for the limiting deterministic design condition, the shutdown cooling system is assumed to fail. Thus the reactor is postulated to undergo a complete loss of forced circulation

TABLE 2
POTENTIAL OFFSITE DOSES FROM THE BOUNDING DESIGN CONDITIONS ANALYZED IN PSID

	DBE	Deterministic Condition	Dose @ 30 Day Exclusion Area Boundary (425 m) Thyroid (rem)	Whole Body (rem)
1.	Loss of HTS and SCS Cooling	Pressurized Conduction Cooldown (PCC)	No release	No release
2.	HTS Transient w/o Control Rod Trip	PCC w/o Control Rod Trip	No release	No release
3.	Control Rod Withdrawal w/o HTS Cooling	PCC with Control Rod Withdrawal	No release	No release
4.	Control Rod Withdrawal w/o HTS and SCS Cooling	PCC with Control Rod Withdrawal	No release	No release
5.	Earthquake	PCC with Earthquake	No release	No release
6.	Moisture Inleakage	Depressurized Conduction Cooldown (DPC) with Moderate Moisture Ingress		
		- Median Dose	.37	.004
		- 95th Percentile Dose	3.8	.045
7.	Moisture Inleakage w/o SCS Cooling	DPC with Moderate Moisture Ingress		
		- Median Dose	.37	.004
		- 95th Percentile Dose	3.8	.045
8.	Moisture Inleakage with Moisture Monitor Failure	DPC with Small Moisture Ingress		
		- Median Dose	<.37	<.004
		- 95th Percentile Dose	<3.8	<.045
9.	Moisture Inleakage with Steam Generator Dump Failure	DPC with Small Moisture Ingress		
		- Median Dose	<.37	<.004
		- 95th Percentile Dose	<3.8	<.045
10.	Primary Coolant Leak	DPC with Moderate Primary Coolant Leak		
		- Median Dose	.03	.0005
		- 95th Percentile Dose	.36	.004
11.	Primary Coolant Leak w/o HTS and SCS Cooling	DPC with Small Primary Coolant Leak		
		- Median Dose	.21	.0009
		- 95th Percentile Dose	2.75	.007

Requirement

	Thyroid (rem)	Whole Body (rem)
10CFR100 Doses	300	25
EPA 520 Lower Limit PAG Doses	5	1

without scram. The loss of forced circulation while the reactor remains at full power results in a core temperature rise which, in turn, causes the reactor to become subcritical owing to its negative temperature coefficient.

At 56 seconds after the initiating event, the Plant Protection and Instrumentation System (PPIS) releases the reserve shutdown control material into the reactor core as a result of core power-to-circulator speed ratio being above its trip point for more than fifty seconds. At that time, core power has already been reduced to 33 percent by the inherent negative temperature feedback. The power would continue to decrease as the core temperatures increase. However the reserve poison material is inserted causing the reactor power to fall rapidly to decay heat levels.

The gradual heatup of the core in this event produces a peak maximum core temperature of 1296°C after some 95 hours. The rapid initial heatup causes the system pressure to peak at 1009 psia which is below the relief setting of 1041 psia. Since the Reactor System remains pressurized with the primary coolant boundary intact and temperatures stay well below the threshold for fuel particle failure, no radioactivity is released from the vessel, and no offsite dose occurs.

The function for controlling core heat is also challenged by the second event. For this event, the spurious uninhibited withdrawal of an outer reflector control rod group (consisting of three rods) at their maximum withdrawal speed without reactor power setback results in a reactivity transient producing excess power above normal levels and increased core temperatures. As core temperatures increase, the reactivity defect due to temperature becomes more negative and thus mitigates the reactivity increase caused by the rod group withdrawal. Heat generation is brought fully under control when at 99 seconds the reactor trips on high core power ratio of 1.5. The peak power level reaches only 147 percent at about 100 seconds. The heatup of the core in this event leads to a peak temperature of only 1394°C, well below the threshold for fuel particle failure. Additionally, the primary system integrity is maintained and thus there is no offsite dose.

4.3 Remove Core Heat Events

The limiting event analyzed in terms of challenging the remove core heat function is a loss of coolant or depressurization accident with failure of all forced cooling systems. For this event, a 12.7 square inch leak is assumed to occur in the pressure relief train, depressurizing the primary system in some minutes. At 20 seconds, the reactor trips on a low-pressure signal. Although the Shutdown Cooling System is designed to cool the core following this event, it is assumed to fail. Thus, heat is removed from the core in this event only by radiation and conduction to the reactor vessel. In turn, heat is removed from the vessel by the passive Reactor Cavity Cooling System.

The thermal transient experienced by the core is shown in Figure 2. The core temperature reaches a peak of slightly over 1600°C in approximately 80 hours after the loss of forced circulation. Beyond this point, the core heat removal rate exceeds the afterheat-generation rate resulting in a slow cooldown of the core.

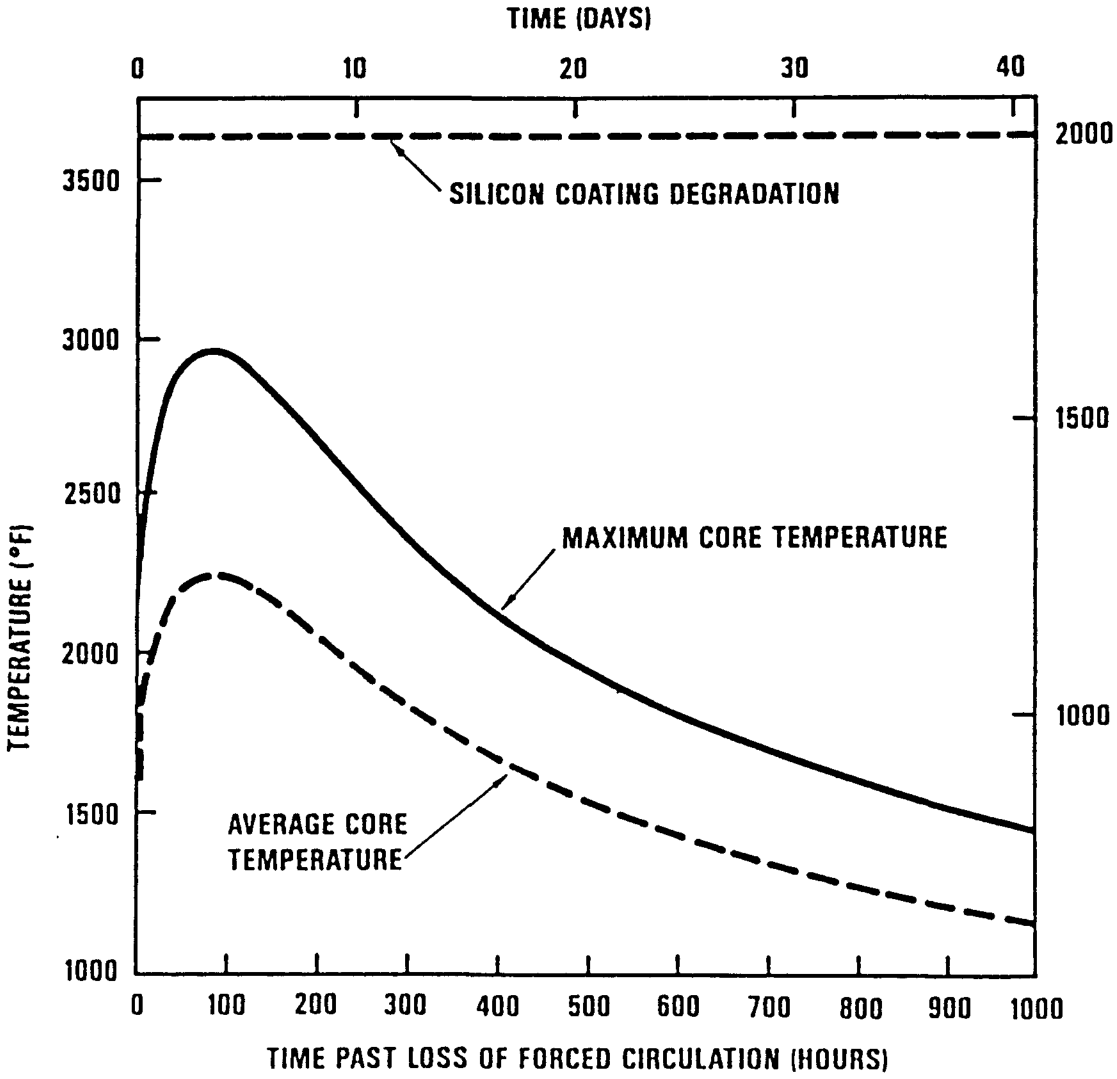

Figure 2
Maximum and Average Core Temperatures During a Depressurized Conduction Cooldown with Primary Coolant Leak

For this event the core does heat up to a point at which some incremental release from the fuel occurs and the event initiator causes a loss of primary boundary integrity, so that there is a release of fission products to the Reactor Building and to the environment. The limiting radionuclide in this release is calculated to be Iodine-131; some 150 curies of this are assessed to be released from the core and approximately 25 curies of this are assessed to be released from the vessel to the environment. The resulting offsite dose to the thyroid is conservatively assessed to be .36 rem, well within the 10CFR100 limit of 300 rem.

4.4 Control Chemical Attack Events

The limiting event analyzed in the PSID with regard to controlling chemical attack is a depressurized conduction cooldown with a moisture ingress. This event is postulated to be initiated by a break in a single steam generator tube resulting in a water ingress to the primary system. Reactor trip is provided by high power-to-flow ratio as a result of power increase due to the positive reactivity effect of the moisture. The detection of the moisture inleakage by the moisture monitor system is deterministically assumed unavailable. As the water ingress continues, primary system pressure increases until the reserve shutdown control material is inserted due to a high pressure signal. At the same time, the main loop and steam generator are automatically isolated on high pressure. However, the steam generator dump system is deterministically assumed to fail to empty the remaining contents of the steam generator and further the Shutdown Cooling System is assumed to fail to start. Thus even after steam generator isolation, steam continues to enter the primary system until the pressure in the steam generator equilibrates with the primary system pressure. As a result, some 9000 lbm of steam enters the primary system.

Due to the overpressure condition, the primary relief valves open some 370 seconds into the accident. The relief valve is assumed to cycle open and closed several more times maintaining primary pressure below the relief setpoint of 1041 psia, until finally the relief valve fails open, depressurizing the primary system. At this point the system enters into a depressurized conduction cooldown such as was described in the previous section.

The moisture in the primary system in this case, however, is available to chemically attack both the graphite and the fuel. Reaction of the graphite occurs mainly in the hotter bottom half of the core; however, not enough oxidation occurs to result in any structural or flammability concerns. Additionally, a fraction of the fuel particles which initially have defective fuel coatings and thus expose the fuel kernel to the steam environment results in fuel hydrolysis and liberation of fission products.

For this event the core does heat up to a point at which some incremental release from the fuel occurs; additionally fission product release occurs due to fuel hydrolysis, and the event initiator causes a loss of primary boundary integrity, so that there is a release of fission products to the Reactor Building and to the enviroment. The limiting radionuclide in this release is calculated to be Iodine-131; some 50 curies of this are assessed to be released from the vessel to the environment. The resulting offsite dose to the thyroid is conservatively assessed to be 3.8 rem, well within the 10CFR100 limit of 300 rem.

4.5 Safety Asessment Summary

The assessment of the Standard MHTGR capability to control accidental radioactivity releases shows that the doses are a small fraction of 10CFR100 even for the bounding analyses which consider only the systems, structures and components that require neither operator action nor other battery power. In fact, the exposures are so low that the Protective Action Guidelines would require no evacuation or sheltering plans for the public. The PSID evaluation confirms that USNRC accident dose criteria can be made with a containment system that places primary emphasis on the retention within the fuel barriers.

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