Containment Versus Confinement for High-Temperature Gas Reactors

Regulatory, Design Basis, Siting, and Cost/Economic Considerations

1011948

Plant Maintenance Support

Reduced Cost

Equipment Reliability

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REPORT SUMMARY

This report provides the results of an investigation pertaining to the use of the confinement that has been proposed for the high temperature and very high temperature gas reactors (HTGR, VHTR). No comprehensive study of this question has been published since 1985. All power reactor designs to go into commercial service in the United States were light water reactors (LWR), except for Fort St. Vrain (FSV) and Peach Bottom Unit 1, which were steam cycle helium gas cooled reactors. All designs use a leak tight containment except FSV, which used a confinement.

Background

For the LWR, the containment is one of the three barriers to fission product release to the environment, the others being the fuel cladding and the reactor coolant pressure boundary. These barriers were part of a larger concept of defense-in-depth that basically required that plant safety not be wholly dependent on any single element of the design. The confinement, on the other hand, refers to a vented low pressure containment. The vent operates to limit positive pressure. During normal operation, the confinement is maintained at slightly negative pressure and during an accident, the vent operates to limit pressure in the confinement. The basis for using a confinement is predicated on the thermal response of the HTGR, in combination with the time-at-temperature effect on fuel fission product retention and helium characteristics. In a depressurisation accident, fission product release occurs long after the depressurization is completed, even for relatively small leaks. If the containment has been vented, there will be no transport force for the fission products. This suggests that the vented confinement system is in fact the more optimum approach to ensure public safety. This is premised on the qualification of the fuel as a containment or barrier to fission product escape.

Objectives

Refine existing assessments and studies to consider the effects of the following constraints to the confinement concept:

- Siting considerations
- Design basis events scenarios
- Regulatory climate
- Costs/economics relating to not only the initial investment but also the ongoing operations

Approach

A survey of literature, reports, presentations, etc., that are related to the use of a confinement as the final barrier to release of radioactivity to the environs was performed. The information presented was peer-reviewed by experienced plant personnel of the EPRI Maintenance Center. The document discusses confinement structure characteristics, current thinking and activities at the NRC, design basis events (accidents) and their mitigation, siting considerations which focuses on source term and fuel performance, and cost and economics associated with a confinement as opposed to a conventional containment. A concluding section integrates the challenges associated with each section.

Results

Many challenges exist to the confinement concept. Confinement designs are being scrutinized for adequacy in defense-in-depth and the ability to deal with "cliff-edge" accidents or air-ingress post-accident. A hybrid design is being proposed that permits initial confinement depressurization followed by closure and a low-leakage requirement. In addition experience with the use of TRISO coated fuel is limited. Fuel integrity is crucial to any assumptions made that permit the use of a confinement. Additional fuel integrity research is essential.

In the final analysis, the use of a confinement for the HTGR will rest on the ability for the industry to provide incontrovertible evidence that the fuel integrity will be robust enough to provide the advertised barrier to fission product release. Without that confidence, provided by low probabilities with high confidence coefficients (meaning lots of realistic testing), it will be difficult to justify the fuel as a barrier and the requirement for a low-leakage containment will exist.

Keywords

Confinement
Regulatory
Design Basis
Siting
Costs

CONTENTS

1
INTRODUCTION

1.1 Scope

This report provides the results of an investigation pertaining to the use of the confinement that has been proposed for the high temperature and very high temperature gas reactors (HTGR, VHTR). HTGR will be used throughout this report. These include the General Atomic designed gas turbine modular helium reactor (GT-MHR) and PBMR (Pty), Ltd designed pebble bed modular reactor (PBMR). No comprehensive study of this question has been published since 1985 [1].

1.2 Background

The last of the current power reactor designs to go into commercial service was Watts Bar I in 1996. All were light water reactors (LWR), except for Fort St. Vrain and Peach Bottom Unit 1, which were steam cycle helium gas cooled reactors that operated during the period 1967 through 1989. All designs use a leak tight containment (a requirement that was finally codified in 10CFR50 Appendix A, Criterion 16), except Fort St. Vrain (FSV) that used a confinement. The reason for this probably stems from the fact that FSV was originally operated by the Department of Energy, which typically uses confinement structures for its production reactors [2].

For the LWR, the containment is one of the three barriers to fission product release to the environment, the others being the fuel cladding and the reactor coolant pressure boundary. While a great deal of effort is expended to ensure the integrity of the latter two, no containment performance is considered the most important of the three. The mitigating effects of the containment on the outcome of the Three Mile Island accident in 1979 cannot but reinforce this. These barriers are part of a larger concept of defense-in-depth that basically required that plant safety not be wholly dependent on any single element of the design [3].

The confinement, on the other hand, refers to a vented low pressure containment. The vent operates to limit positive pressure. During normal operation, the confinement is maintained at slightly negative pressure using a filtered exhaust. During a depressurization accident, the vent operates to limit pressure in the confinement to less than 7 kPa (1 psi) differential. The typical accident leakage rate is one confinement volume per day at the release pressure of 7 kPa (1 psi) differential. The vent may or may not be filtered depending on design and accident scenario. [4] [5]

1.3 Discussion

The proposed use of a confinement versus the very low leakage containment used for today's LWRs is based on the following discussion found in Reference [6] issued by the International Atomic Energy Agency.

> "The thermal response, in combination with the time-at-temperature effect on fuel fission product retention and the helium characteristics noted earlier, fundamentally alters the effectiveness of strategies for fission product containment. For example, in a depressurisation accident, the predicted small fission product release from the fuel occurs long after the depressurization is completed, even for relatively small leaks. At this time, there would be no driving force to transport the fission products. In fact, once the maximum temperature is reached and the system begins to cool, the net flow is inward. However, if the released gas is contained, with a small (e.g. 1%/day) leakage rate, the leakage flow and slowly decreasing pressure would provide a mechanism for fission product transport. Thus attempting to contain the leaking helium can result in a higher fission product release rate for some of the most limiting events.

For the United States, what is necessary to allow a confinement is an approach and associated criteria that leads to the selection of events to be considered in the design and for emergency planning purposes and the accident source terms used in the analysis. As discussed in Section 3, NRC activities are on-going that address these matters.

1.4 Approach

This report is the result of a survey of literature, reports, presentations, etc., that are related to the use of a confinement as the final barrier to release of radioactivity to the environs. It presents relevant information and challenges for the confinement approach.

The document is divided into five sections. It begins with identifying what the confinement structure means in a plant context. Next it reviews current thinking and activities at the NRC. This is followed by design basis events (accidents) and their mitigation and siting considerations which focuses on source term and fuel performance. The final informational chapter reviews cost and economics associated with a confinement as opposed to a conventional containment. A concluding section integrates the challenges associated with each section.

1.5 Acronyms

The acronyms used in this report are found in Appendix A.

2
CONFINEMENT DESCRIPTION

2.1 Basis for Confinement

The basis for a HTGR confinement can be found in the International Atomic Energy Agency publication *Considerations in the Development of Safety Requirements for Innovative Reactors: Application to Modular High Temperature Gas Cooled Reactors*, [6]. Paragraph 2.5:

> The [slow] thermal response, in combination with the time-at-temperature effect on fuel fission product retention …fundamentally alters the effectiveness of strategies for fission product containment. For example, in a depressurisation accident, the predicted small fission product release from the fuel occurs long after the depressurization is completed, even for relatively small leaks. At this time, there would be no driving force to transport the fission products. In fact, once the maximum temperature is reached and the system begins to cool, the net flow is inward. However, if the released gas is contained, with a small (e.g. 1%/day) leakage rate, the leakage flow and slowly decreasing pressure would provide a mechanism for fission product transport. *Thus attempting to contain the leaking helium can result in a higher fission product release rate for some of the most limiting events.* [Italics added]

In other words, when coolant is no longer removing heat, the temperature build-up is slow and significant fission product release is delayed. As long as the coolant activity is kept low during operation (this can be monitored), then venting the pressure caused by the break is acceptable at the beginning of the accident scenario since the fission product content is at operational levels. When expect fission product release does become significant, the only source of pressurization is reactor decay heating the remaining gas. This is mitigated by the passive reactor cavity cooling system.

2.2 Building Description

Confinement refers to a space that surrounds the reactor and associated systems that contain the helium coolant. Other terminology used includes vented containment or vented low pressure containment (VLPC). The vent flow path uses dampers to limit positive pressure and allow for maintaining a negative pressure to prevent radioactive product release during operation using a filtered exhaust. The reactor building for the proposed PBMR is shown in Figure 2-1. The shaded area is the confinement. This structure is presently designed to withstand the Safe Shutdown Earthquake and well as tornados, flooding, non-commercial aircraft crash, etc. [7] However, in today's terrorist threat environment, the need for designing for a commercial airplane crash may be needed.

The single most significant difference between the present day containment and the proposed confinement is, of course, the accident leakage rate. As explained in Section 5, this is essential for any off site dose calculation. Typical accident leakage rate is one confinement volume per day at the release pressure of 7 kPa (1 psi) differential. The confinement volume is on the order of 7400 cubic meters (260,000 cubic feet) for the GT-MHR [4] and 16,000 cubic meters (565,000 cubic feet) for the PBMR [8].

As a comparison, the typical allowable leakage for the light water reactors containment building is 0.1% of the containment volume (technically the mass) per day at accident pressure. The volume is typically on the order of 57,000 cubic meters, (2,000,000 cubic feet). Accidents pressures are typically can vary from about 70-350 kPa (≈10-50 psi) differential depending on containment design.

2.3 Limiting Confinement Heat Up

There are two cooling systems that remove heat from the core and hence limit confinement heat-up when generation is zero. These are the:

- Shutdown Cooling System
- Reactor Cavity Cooling System

Only the Reactor Cavity Cooling System (RCCS) is safety grade. It is passive, i.e., it does not rely on moving parts, any source of external power, instrumentation, etc. It is capable of removing all core residual heat without allowing the fuel to exceed its design temperature. By removing core heat it also mitigates the thermally driven accident leakage. It also prevents the vessel from exceeding its design limits and the concrete walls from exceeding design temperature limitations. The RCCS based on the General Atomic GT-MHR design is shown in Figure 2-2 [4] and is representative of a typical passive cooling system.

The RCCS relies on the natural circulation of air due to temperature differences between the atmosphere and cooling panels. Cold air flows down from the intake/exhaust structure and into the outer cold downcomer section of the cooling panel, turns in the bottom cold plenum and ascends in inner riser section where it receives heat from the annulus space sorrounding the vessel. Heating lowers the density of the air and results in bouyant force that causes the air to flow through the hot duct to the intake/exhaust structure to the atmosphere. The annulus space is heated by the core heat that has transferred through the vessel wall. The cooling panel surrounds the vessel over the full length and circumference. Most of the outside of the cooling panel is in contact with the concrete in order to perform its temperature limiting function.

2.4 Challenges

The confinement designs, like their confinement predecessor, are not designed for a commercial airplane crash. However, in today's terrorist threat environment, the need for designing for a commercial airplane crash may be needed.

To limit pressure to 7 kPa (1 psi) differential, dampers will be needed. Dampers are like check valves and are required to change position to perform their safety function. Damper action will need to be tested periodically to confirm its reliability. These tests will probably be conducted in conjunction with the LLRT tests described in Section 6.

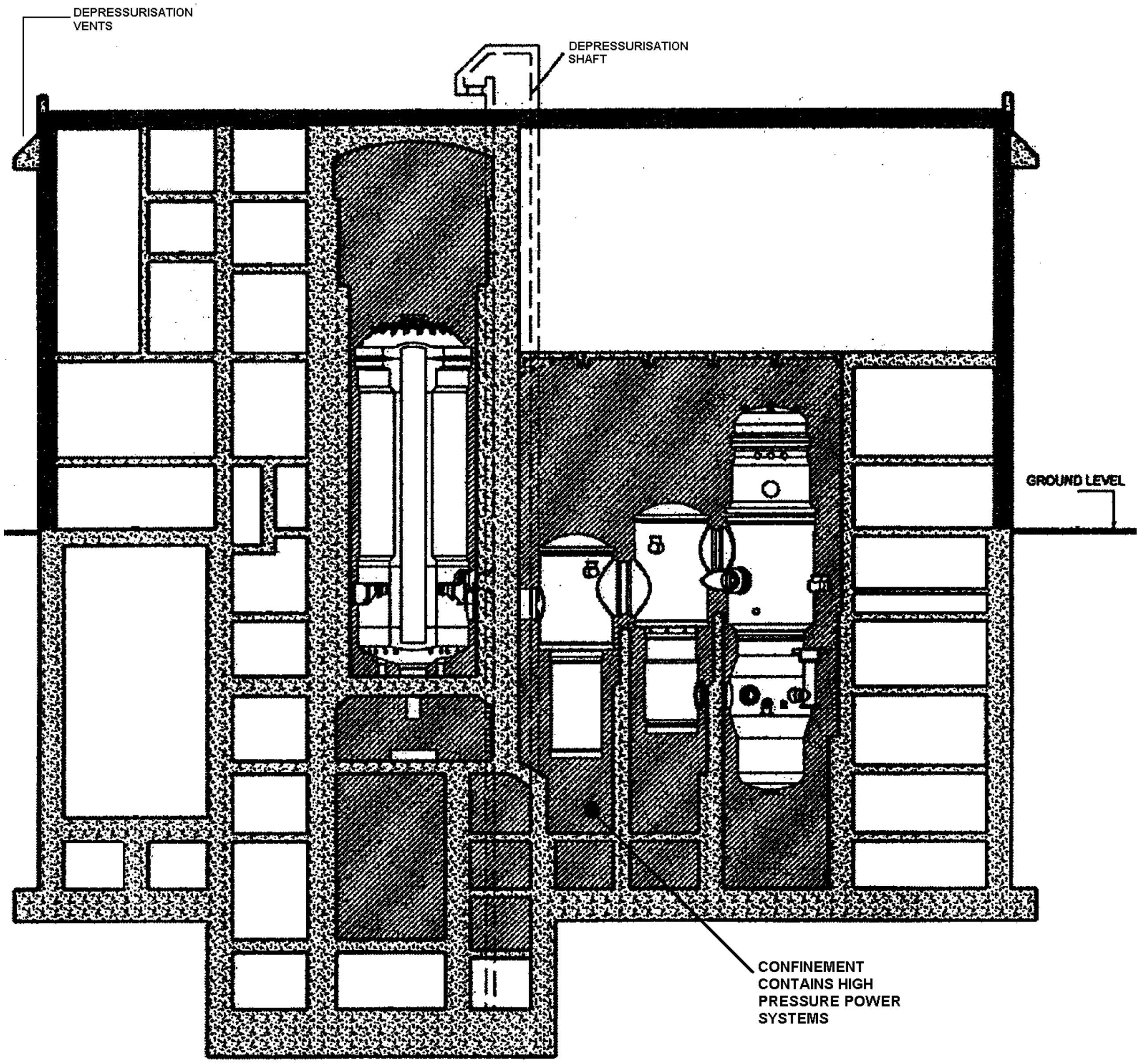

Figure 2-1
PBMR Confinement [7]

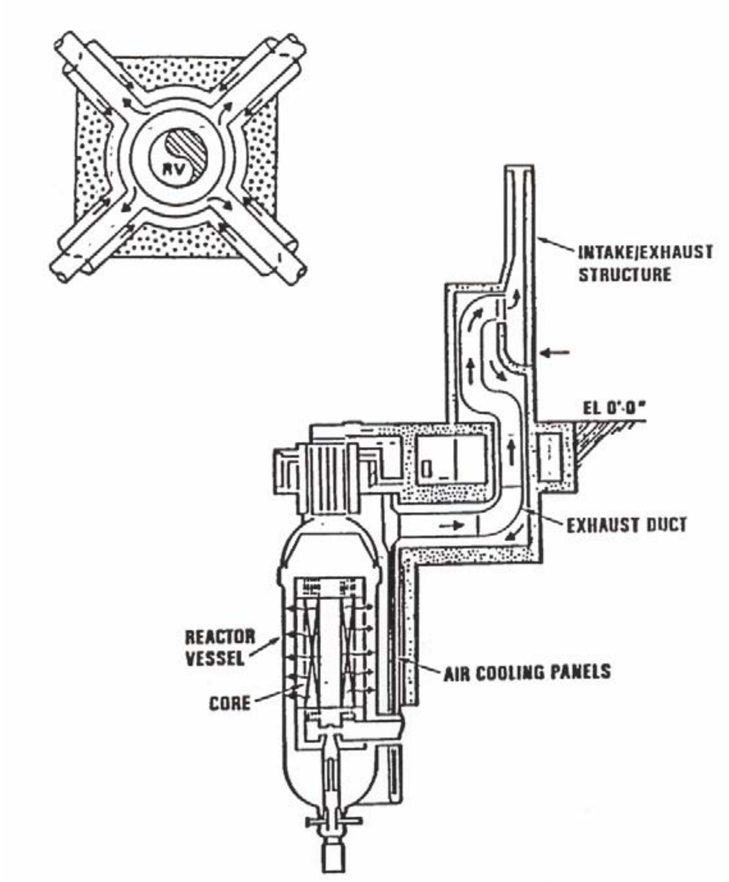

Figure 2-2
GT-MHR Reactor Cavity Cooling System [4]

3 REGULATORY CONSIDERATIONS

3.1 Introduction

In the final analysis, the U. S. Nuclear Regulatory Commission (NRC) will play an important, if not the deciding factor, as to whether the proposed confinement is used. They ultimately will reflect public opinion through policies and the rule making and licensing process. This section will review the current policy making process in the NRC. This is a dynamic process, accelerating a great deal in the past three years.

3.2 Background

Considerable activity is underway to establish the criteria that will be used to resolve the question of use of a confinement. This has been a long-term activity going back to 1986 with the issuance of the the Statement of Policy for the Regulation of Advance Nuclear Power Plants [9].

In this policy statement the Commission expected that advanced reactors would provide enhanced margins of safety and/or utilize simplified, inherent, passive, or other innovative means to accomplish their safety function. Among attributes considered important for acceptability were designs that minimized the potential for severe accidents and their consequences by providing sufficient inherent safety, reliability, redundancy, diversity and independence in safety systems. Also important were designs that incorporated a *defense-in-depth* philosophy by maintaining multiple barriers against radiation release and by reducing the potential for and consequences of severe accidents.

3.3 Current Activity

Since 2002 activity at the NRC has increased considerably. At present only PBMR (Pty.) Ltd. is in early stages of the licensing process. General Atomics (GA), designers of the gas turbine modular helium reactor (GT-MHR) informed the NRC last year that it plans to terminate GT-MHR pre-application review interactions with the NRC and refocus its gas reactor design and development effort on the VHTR project. [10]

In July 2002 the NRC Staff layed out their plan to resolve licensing issues relating to non-light water reactor designs that were being proposed for future nuclear power plants. [3] The following issues were identified: enhanced safety, defense-in-depth, international safety standards and requirements, probabilistic approach, scenario-specific accident source terms, licensing without a pressure-retaining containment building, and reduction of the site exclusion boundary. These issues have been further refined and in January 2005 resulting in the following current nine issues [11]:

1. Integrated risk: Not applicable to this report.
2. Containment functional performance requirements and criteria: Discussed in Section 3.4.
3. Level of safety: Not applicable to this report.
4. Definition of defense-in-depth: Discussed in Section 3.5.
5. Use of a probabilistic approach to establish the licensing basis: Discussed in Section 3.5.
6. Use of scenario-specific source terms for licensing decisions: Discussed in Section 3.6.
7. Possible modifications of emergency preparedness requirements: Discussed in Section 3.7.
8. Physical protection: Not applicable to this report.
9. Selective implementation: Not applicable to this report.

3.4 Containment Functional Performance Requirements and Criteria:

3.4.1 Background

On July 8, 1986 the NRC issued its statement of policy for the Regulation of Advanced Nuclear Power Plants. Advanced reactors were those that were significantly different from the existing generation LWRs then under construction or operation. The Commission expected that advanced reactors would provide enhanced margins of safety and/or utilize simplified, inherent, passive, or other innovative means to accomplish their safety function.

Among attributes considered important for acceptability were designs that minimized the potential for severe accidents and their consequences by providing sufficient inherent safety, reliability, redundancy, diversity and independence in safety systems. Also important were designs that incorporate *defense-in-depth* philosophy by maintaining multiple barriers against radiation release, and by reducing the potential for and consequences of severe accidents.

In 1993 the staff began to address the confinement concept for modular HTGRs by recommending that the acceptability of proposed containment designs be evaluated against a functional performance standard rather than a prescriptive criterion. Specifically, the containment designs were to be adequate to meet the specified onsite and offsite radionuclide release limits for the event categories within their design envelope. The Commission responded to this recommendation by adding that the staff should also address the loss of primary coolant pressure boundary integrity whereby air ingress could occur (from the “chimney effect”) resulting in core graphite oxidation and loss of fuel particle integrity. [12], [13]

In March 2003 the staff recommended that the Commission approve the use of functional performance requirements to establish the acceptability of a containment (i.e., a non-pressure retaining building may be acceptable provided the performance requirements can be met). In June 2003 the Commission responded by stating that there was insufficient information to prejudge the best options and to make a decision on the viability of a confinement. The Commission tasked the staff to develop containment functional performance requirements and criteria working closely with industry experts (e.g., designers, Electric Power Research Institute, etc.) and other stakeholders regarding options in this area, taking into account such features as core, fuel, and cooling systems design for new plants. The staff was requested to pursue the development of containment functional performance standards and then submit options and recommendations to the Commission on this important policy decision. [14], The staff has recently developed options for the functional performance requirements and criteria for containment utilizing applicable Commission technical policies, NRC and industry documents, foreign and domestic technical information, and stakeholder input [11]. Stakeholder input includes feedback and comments from industry experts and other stakeholders received at public meetings and letters (Nuclear Energy Institute, Westinghouse, PBMR (Pty) Ltd., Framatome). A summary of those meetings is found in Appendix B. The staff also met with the Advisory Committee for Reactor Safety (ACRS).

3.4.2 Functional Requirements

The NRC determined that the function of containment designs includes a direct or support functional role for the following accident prevention and accident mitigation safety functions:

1. Protect risk-significant structure, systems and components (SSC) from internal and external events
2. Physically support risk-significant SSCs
3. Protect onsite workers from radiation
4. Remove heat to prevent risk-significant SSCs from exceeding design or safety limits
5. Provide physical protection (i.e., security) for risk-significant SSCs
6. Reduce radionuclide releases to the environs (including limiting core damage)

Functions 1 thru 5, of course, are not associated with the policy issue raised to the Commission in Issue 2 (Section 3.3). The NRC is therefore presently focusing only on Function 6.

3.4.3 Performance Requirement

For Function 6, the following technology-neutral *performance requirement* is proposed:

> The containment must be adequate to reduce radionuclide releases to the environs to ensure that doses do not exceed the dose criteria for the selected events in the event categories.

Four *criteria* options for this containment performance requirement options were developed to address “reducing radioactive releases to the environs” for use in the proposed regulatory structure for new plant licensing:

Option 1: The containment must adequately reduce radionuclide releases to the environs to meet the onsite and offsite radionuclide dose acceptance criteria for the events selected for the event categories.

Comment: This is basically the criteria used for proposed confinement designs.

Option 2: The containment must adequately reduce radionuclide releases to the environs to meet the onsite and offsite radionuclide dose acceptance criteria for the events selected for the event categories (including within the design-basis category, selected credible events having the potential for high consequence source terms).

Comment: This adds the requirement to Option 1 for so-called “cliff-edge” events. These events have the potential for a steeply increased source term. As implied, these events would need to be considered even though their frequency including uncertainties) fell below the lower cutoff for the *design-basis event* category, or potentially even below the lower cutoff frequency for the *beyond the design-basis event* category.

Option 3: The containment must adequately reduce radionuclide releases to the environs to meet the onsite and offsite radionuclide dose acceptance criteria for the events selected for the event categories (including within the design-basis category, selected credible events having the potential for high consequence source terms) and have the capability to establish controlled leakage and controlled release of delayed accident source term radionuclides.

Comment: This is Option 2 with the additional capability to control leakage. Option 3 was the option chosen and the consequences of this choice are discussed below.

Option 4: The containment must adequately reduce radionuclide releases to the environs to meet the onsite and offsite radionuclide dose acceptance criteria for the events selected for the event categories (including within the design-basis category, selected credible events having the potential for high consequence source terms) by being essentially leak tight against the release of prompt and delayed accident source term radionuclides.

Comment: This is the present day LWR criteria. It is not consistent with current Commission guidance and was considered to show why this is so.

3.4.4 Functional Performance Criteria

Option 3 was chosen by the NRC Staff to be the proposed containment *functional performance criteria* for reducing radionuclide releases to the environment.

Option 3 contains a prescriptive requirement that the containment have the capability to establish a controlled leakage and a controlled radionuclide release capability. The NRC feels that this capability ensures that the containment will provide a significant deterministic element of *defense-in-depth* to controlling radioactive releases, should the other mechanistic barriers and obstacles to fission product transport provided by the fuel, core materials and reactor coolant system not perform as expected or should unanticipated events involving a larger than expected accident source term occur. This element is independent of the performance of the other mechanistic barriers and, for some designs, also has the potential to prevent or mitigate certain kinds of accidents (e.g., HTGR air ingress).

This option would reduce concerns related to maintaining fuel quality and fuel performance during normal operation and accidents over the life of the plant. However, by requiring the containment to have an additional capability to reduce releases, it might reduce the incentive to emphasize accident prevention in designs, thereby potentially having an adverse affect on plant safety. Since this criterion involves a prescriptive element, it is not totally performance-based. This option also goes beyond the containment functional performance criteria that is being proposed for selected new plant designs (i.e., HTGRs)

Compared to Option 1, this option could add to the cost of the containment. It would also differ from the prior Commission decision on containment performance requirements documented in Reference [12] by requiring additional mitigation capability regardless of meeting onsite and offsite dose performance criteria. However, It would likely further increase public confidence compared to Options 1 or 2.

The inclusion of “cliff-edge” events in the design-basis could require additional technology development to support the source term calculations for these events. Also, depending on the analysis results, this option might require design-related enhancements involving incremental costs. Including more challenging and lower probability events in the containment design-basis would likely increase public confidence relative to Option 1.

3.4.5 NRC Perspective of HTGR Design Implications

For modular HTGRs, the NRC believes that Option 3 would still allow a confinement structure, which they designate as a vented low pressure containment (VLPC). Since it will have a prescriptive requirement to have the capability to establish controlled leakage, this design has been referred to as a “hybrid containment” because it would allow the initial RCS depressurization to vent directly to the environs for loss of reactor coolant pressure boundary events, but would require that it have the capability to establish a controlled, low leakage, thereafter. This option would limit the volume of air ingress available for core oxidation and would limit the volume of air out-leakage available for radionuclide transport to the environs of the delayed radiological source term.

The NRC’s current view is that not all HTGR designs presently require the VLPC to have the capability the limit the inleakage rate and out-leakage rate from the VLPC to a controlled limited value after a severe depressurization event. It is likely that design changes would be required. For such plants, this option would also likely require structural changes to the VLPC design in order to meet structural stress (i.e., ASME) limits and upgrades to the vent system in order to assure a reliable vent path reclosure capability. This would add to the design and construction costs of the VLPC. It would also have a impact on testing and confinement maintenance.

3.5 Defense-in-Depth

3.5.1 Background

Defense-in-depth (DID) appears to be the singular most important consideration for any containment versus confinement question. It was first mentioned in regulations in the Advanced Reactor Policy (See section 3.4.1). The only other mention is 10CFR50 Appendix R.

DID is a concept deep-rooted in the approach taken early in the design of nuclear power plants. As will be discussed below DID and probabilistic risk assessment (PRA) are strongly interlinked. In fact the one definition that was put forward for DID is in the NRC’s White Paper on Risk-Informed and Performance-Based Regulation [15]:

> "Defense-in-Depth is an element of *NRC's Safety Philosophy* that employs successive compensatory measures to prevent accidents or mitigate damage if a malfunction, accident, or naturally caused event occurs at a nuclear facility.

As a consequence, based on the Reference [14], the staff is currently developing a description of defense-in-depth for incorporation into the policy statement on the use of probabilistic risk assessment (PRA).

3.5.2 Probabilistic Risk Assessment

Issue 4 (Section 3.3) deals with PRA. The Commission has already agreed to the use of PRA (in the form of risk information) in the establishment of a licensing basis. The outcome of Issue 4, per se, will not affect the confinement decision. How DID will affect the decision making of PRA may result in prescriptive requirements, however.

The reason for this is that PRA and DID are both complementary and opposing concepts, and there has been, in the past, a reluctance to accept PRA results in regulatory decision making. For example, instances were identified by the ACRS in their letter to the Commission of May 19, 1999 [16]. The ACRS reported seemingly arbitrary appeals to DID to avoid making changes in regulations or regulatory practices that seemed appropriate in the light of results of quantitative risk analyses. An example given was a reluctance to develop new, risk-informed limits on leakage from steam generator tubes because the tubes were part of the DID barriers. (Read here the acceptance of risk-informed limits on leakage from the containment.)

The ACRS presented reasons for these why they felt this was happening. There were two views: a structualist view in which DID is considered to be the application of multiple and redundant measures to identify, prevent, or mitigate accidents to such a degree that the design meets the safety objectives. This is the general view taken by the plant designers. The other view (the "rationalist"), sees the proper role of DID in a risk-informed regulatory scheme (PRA based) as compensation for inadequacies, incompleteness, and omissions of risk analyses. A key missing ingredient was to place quantitative limits on DID measures by assigning acceptance values on the level of uncertainty for each safety objective. In other words the rationalist view. The ACRS acknowledged, however, that considerable judgement would have to be exercised to set these limits on uncertainty, expecially uncertainties not quantified by PRA.

Another reason for this reluctance is that the structuralist view is deterministic. The application of "well-understood" rules that have withstood the test of time is a powerful argument. In most cases the safety vulnerability could be handled with various strategies including well-defined safety classifications, construction codes, single-failure criterion, etc. But these strategies are really based on a sense of vulnerability for the structure, system or component to the design basis event. These "arbitrary" safety classifications, etc., can be replaced with PRA based classifications, etc. That is exactly the intended result of Issue 4.

3.5.3 Current Actions

It is clear, therefore, that a better understanding of DID and "rationalists" viewpoint is necessary if the confinement is to be the final choice. The NRC is proceeding along a path to bring understanding and reason and limitation to the use of DID. The basis for this path are three objectives for DID. These are the ability to:

- compensate for potential adverse human actions (this includes commission as well as omission) and component failures,
- maintain the effectiveness of barriers by averting damage to the plant and the barriers themselves, and
- protect the public and environment from harm in the event that these barriers are not fully effective.

The path includes developing a technology-neutral (applies to all reactor designs) framework for specifying DID. This framework has three key elements:

1. development of defense-in-depth principles
2. development of a defense-in-depth model for application
3. guidance on the implementation of the defense-in-depth model.

Defense-in-Depth Principles

To achieve the defense-in-depth objectives, and therefore assure public safety despite uncertainties, the staff is proposing the following fundamental principles.

First measures against intentional as well as inadvertent events must be provided. This is intended to ensure that in the application of defense-in-depth human initiated (e.g., security), as well as random events and natural phenomena, are considered.

Second the design should provide accident prevention and mitigation capability. Accident prevention and mitigation capability should be provided such that there is no undue emphasis on either' at the expense of the other, for maintaining the plant in a safe condition given various challenges.

Third accomplishment of key safety functions should not be dependent upon a single element of design, construction, maintenance or operation. Redundancy, diversity, and independence in structures, systems, and components (SSCs) and actions will ensure that no key safety functions will be dependent on a single element (i.e., SSC or action) of design, construction, maintenance or operation.

Fourth, uncertainties in SSCs and human performance should be accounted for such that reliability and risk goals can be met. Allocation of risk goals for a new design must include uncertainty. The setting of success criteria for the achievement of safety functions should be set, and the calculations that show they have been met should be performed, in such a way that uncertainties are accounted for with a high level of confidence.

Fifth, plants should be sited in areas that meet the intent of Part 100 and are consistent with the principles for siting established in Regulatory Guide 4.7 ("General Site Suitability Criteria for Nuclear Power Plants). The location of regulated facilities should be chosen so as to serve the protection of public health and safety.

Defense-in-Depth Model

The model of defense-in-depth which the staff is recommending for application to new reactors incorporates both deterministic and probabilistic elements. The deterministic part of the model mainly addresses completeness uncertainties by asking the question, "What if this barrier or safety feature fails?" without relying on a quantitative estimate of the likelihood of such a failure. As a result, the deterministic element is defined by protective strategies that are successive measures designed to protect public health and safety even if some of the strategies fail. The protective strategies of the technology-neutral framework are to ensure Physical Protection, maintain Barrier Integrity, limit Initiating Event Frequencies, assure adequate reliability of Protective Systems, and provide Accident Management. In addition, the deterministic element imposes specific qualitative requirements to be included in the regulations to ensure that the accomplishment of key safety functions are not dependent upon a single element of plant design construction, maintenance or operation.

The probabilistic part of the model seeks to evaluate the uncertainties in the analysis and to determine what steps should be taken to compensate for those uncertainties. The probabilistic elements address primarily modeling and parameter uncertainties, and establish specific quantitative performance goals, such as equipment reliability goals, that compensate for the calculated uncertainty.

The staff's defense-in-depth model uses a deterministic approach at a high level by requiring that all the protective strategies are included. Within each protective strategy a probabilistic approach is used to determine how much defense-in-depth is needed to achieve the desired quantitative goals on initiating event frequency and safety system reliability, including uncertainty.

Implementation of the Defense-in-Depth

The staff's approach for implementation of the above model will rely on the application of the defense-in-depth principles as qualitative criteria to be adhered to, and the use of a PRA for achieving quantitative risk goals. Inclusion of all the protective strategies assures some protection against completeness uncertainty.

The staff envisions process of applying defense-in-depth as iterative. They are expected to be used initially by the designer and ultimately by the designer and regulator to develop the emerging design. As the design evolves, the PRA will also be able to be developed to greater detail.

3.6 Use of Scenario-Specific Source Terms for Licensing Decisions

The NRC has developed draft guidance for implementation of a scenario specific source term approach to be used as NRC staff guidance. Summarized below are the key elements of the draft guidance:

- The scenarios to be used for the source term evaluation are to be selected from a design specific probabilistic risk assessment, with due consideration of uncertainties, as discussed under the issue addressing the use of a probabilistic approach for establishing the licensing basis.
- The source term calculation, using the selected scenarios, should be based upon analytical tools that have been verified with sufficient experimental data to cover the range of conditions expected and to determine uncertainties.

- The source terms used for assessing compliance with dose related siting requirements should be 95% confidence level values based upon best estimate calculations with quantified uncertainties. Where uncertainties cannot be quantified, engineering judgement shall be used.
- The source terms used in assessing emergency preparedness should be mean values based upon best estimate calculations with quantified uncertainties.
- The source terms used for licensing decisions should reflect the scenario specific timing, form and magnitude of radioactive material released from the fuel and coolant. Credit may be taken for natural and/or engineered attenuation mechanisms in estimating the release to the environment, provided there is adequate technical basis to support their use.

The guidance is intended to provide a flexible, performance-based, approach for establishing scenario specific licensing source terms. However, it also puts the burden on the applicant to develop the technical bases (including experimental data) to support their proposed source terms. Applicants could, however, propose to use a conservative source term for licensing purposes (in order to reduce research and development costs and schedule), provided the use of such a source term does not result in design features or operational limits that could detract from safety.

3.7 Emergency Preparedness Requirements

The Commission has approved the NRC recommendation that no change to emergency preparedness requirements be made at this time. Provision already exists in 10 CFR 50.47 (“Emergency Plans”) for accommodating the unique aspects of high-temperature gas reactors. In the near term, new plants are likely to be built on an existing site which conforms to current requirements. In the longer term, the staff also recommended that the role of emergency preparedness in defense-in-depth would be addressed as part of the staff’s work to develop a policy or description of defense-in-depth which is part of the framework development, as recommended under the defense-in-depth issue.

10 CFR 50.47 does recognize that for gas-cooled nuclear reactors and for reactors with authorized power level less than 250 Mwt, the size of the emergency planning zones (EPZs) may be determined on a case-by-case basis. This situation was the case for the Fort Saint Vrain reactor which had a 5-mile EPZ, instead of the 10-mile EPZ, that is applied to currently operating LWRs. In the past, there have been proposals to modify current emergency preparedness requirements to give credit for designs with enhanced safety characteristics. The NRC Staff has indicated that for new reactor designs, it is too early to identify specific conditions that would allow a reduction in the 10-mile plume exposure pathway EPZ. Until sufficient experience is gained on any prototype reactor, a case-by-case basis should be used to evaluate whether a requested reduction in the size of the EPZ can be allowed. This criterion would also apply to the 50-mile ingestion control pathway EPZ.

Some conditions that would have particular importance would include, but would not be limited to, the following:

- consideration of the full range of accidents
- use of the defense-in-depth philosophy
- prototype operating experience is gained
- acceptance by federal, state, and local agencies
- acceptance by the public.

The staff plans to obtain stakeholder feedback on the above emergency preparedness considerations, as they relate to modifying emergency preparedness requirements to give credit for reactor designs with enhanced safety characteristics. Based upon feedback and further technical considerations, the staff intends to provide a recommendation to the Commission in late 2005.

3.8 Challenges

The NRC is currently developing policy related to licensing non-light-water reactor designs. The NRC staff has been working at the direction of a Commission Staff Requirements Memorandum issued in July of 2003. Seven issues were identified and two directly affect the decision on containment design; defense-in-depth and containment functional requirements and criteria.

The most recent document, issued in January 2005, has presented the staff's latest thinking and is based on stakeholders input and NRC directives regarding the licensing of advance nuclear reactors. The functional requirement criteria for the containment is as follows:

> "The containment must adequately reduce radionuclide releases to the environs to meet the onsite and offsite radionuclide dose acceptance criteria for the events selected for the event categories (including within the design-basis category, selected credible events having the potential for high consequence source terms) and have the capability to establish controlled leakage and controlled release of delayed accident source term radionuclides.

This functional requirement includes credible events with potential for high consequence source terms (so-called "cliff-edge") events in the design-basis that are not presently being considered. This could require additional technology development to support the source term calculations for these events. Also, depending on the analysis results, this option might require design-related enhancements involving incremental costs.

This functional criteria also contains a prescriptive requirement that the containment have the capability to establish a controlled leakage and a controlled radionuclide release capability. The NRC feels that this capability ensures that the containment will provide a significant deterministic element of *defense-in-depth* to controlling radioactive releases, should the other mechanistic barriers and obstacles to fission product transport provided by the fuel, core materials and reactor coolant system not perform as expected or should unanticipated events involving a larger than expected accident source term occur. This element is independent of the performance of the other mechanistic barriers and, for some designs, also has the potential to prevent or mitigate certain kinds of accidents (e.g., HTGR air ingress).

Since it will have a prescriptive requirement to have the capability to establish controlled leakage, this design has been referred to as a “hybrid containment” because it would allow the initial RCS depressurization to vent directly to the environs for loss of reactor coolant pressure boundary events, but would require that it have the capability to establish a controlled, low leakage, thereafter. This option would limit the volume of air ingress available for core oxidation and would limit the volume of air out-leakage available for radionuclide transport to the environs of the delayed radiological source term.

The NRC’s current view is that not all HTGR designs presently meet this functional requirement and that design changes would likely be required. For such plants, this option would also likely require structural changes to the confinement design in order to meet structural stress (i.e., ASME) limits and upgrades to the vent system in order to assure a reliable vent path reclosure capability. This would add to the design and construction costs of the VLPC. It would also have a impact on testing and confinement maintenance.

This option would reduce concerns related to maintaining fuel quality and fuel performance during normal operation and accidents over the life of the plant. However, by requiring the containment to have an additional capability to reduce releases, it might reduce the incentive to emphasize accident prevention in designs, thereby potentially having an adverse affect on plant safety.

While not a challenge, it should be noted that the EPZ sizes will not change with the intial installations. The NRC Staff has indicated that for new reactor designs, it is too early to identify specific conditions that would allow a reduction in the 10-mile plume exposure pathway EPZ. Until sufficient experience is gained on any prototype reactor, a case-by-case basis will probably be used to evaluate whether a requested reduction in the size of the EPZ can be allowed. This criterion would also apply to the 50-mile ingestion control pathway EPZ.

4
DESIGN BASIS EVENTS

4.1 Introduction

As previously discussed, the basis for a HTGR not requiring a conventional pressure-retaining low-leakage reactor containment structure (such as is provided on current LWRs) is that the radionuclides are retained within the fuel during the accident long enough for the effects of depressurization of the reactor vessel to minimize the transport force for delayed fission product escape. This would result in a smaller source term and thus smaller emergency planning zones around the plant compared to today's LWRs. To understand the consequences of the vented containment, an understanding of the events that play a key role in its design requirement is required.

4.2 Fission Transport Phenomena for the HTGR

4.2.1 Fission Products

In the events analyzed, two types of fission products are considered. Immediately available fission products consist of a certain amount of heavy metal contamination (from the manufacturing process), operational fuel failures (defective fuel) and local lift-off of radionuclides that plated out on the primary coolant pressure boundary (PCPB) during operation. Delayed fission products are those that come from the fuel kernel and diffuse through the fuel coating (See Section 5.3) as well as those that result from fuel failure that occurs due to higher temperatures. The availability of these fission products is a function of the chemical element, isotope half-life, fuel burnup and temperature. Figure 4-1 shows the effect of temperature on fission gas release [17]. As would be expected, higher temperatures result in higher releases. An additional discussion on fuel performance is found in Section 5.

4.2.2 Transport Force

Transport force is a result of the pressure difference between the reactor and the containment and the containment and the environment. The operating pressure determines the magnitude of this force, but the effects of temperature must also be taken into account as the accident progresses. In addition since the containment is designed to vent at a very low pressure, (≈ 7kPa (1 psi) differential), and the reactor operating pressure very high, (≈ 7 MPa (1015 psig)), the time availability of the transport force depends largely on the rate of depressurization, with large rates obviously meaning shorter times. Once depressurization has taken place, the decay heat increases the temperature of the gas and this will result in additional release. This temperature buildup is mitigated by the Reactor Cavity Cooling System described in Section 2.

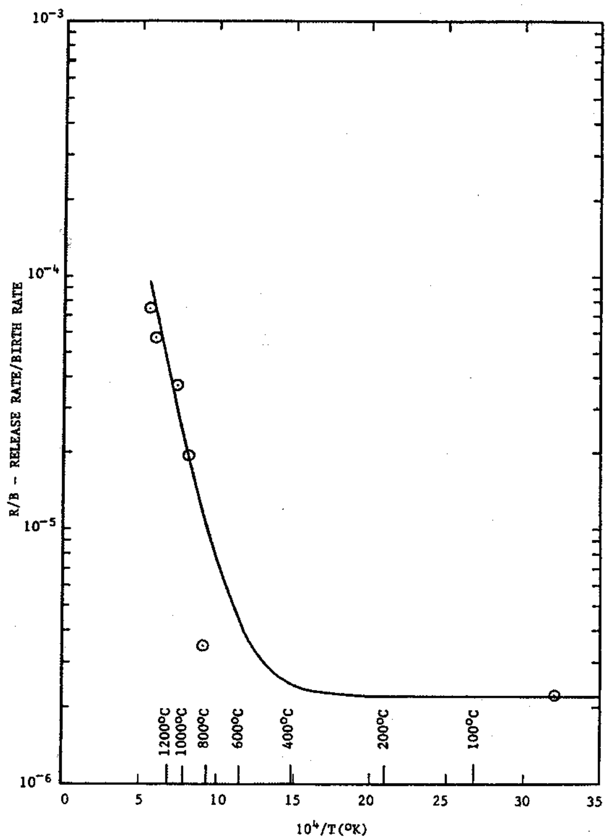

Figure 4-1
Temperature Dependence on Fission Gas Release [17]

4.2.3 Break Size

The consequences of these events are are sensitive to break size. Smaller breaks will mean that depressurization of the reactor vessel will be slow. Therefore the transport force will be available longer. Since fuel degradation increases immediately after the temperature begins to rise, fission products will soon be available for transport. When the containment pressure reaches its maximum value, releases to the environment soon begin. On the other hand with larger break sizes, reactor depressurization is faster and transport forces are not available as long. This also means that containment pressurization is faster and releases to the environment sooner, but with fewer fission products available for release.

4.2.4 Air Ingress

Even after the postulated break and depressurization has occurred, air ingress can occur due to a reversal of flow through the break. These air ingress events can then result in severe oxidation of the graphite core and possibly the fuel coating itself. The exothermic oxidation process itself can then result in the creation of a transport force that carries fission products out of the vessel and thus the containment. Since the fission product flow would be opposite the air flow, these events require larger break sizes, and therefore are less likely, but credible. The NRC considers air ingress events significant enough to consider them when recommending the containment design. (See Sections 3.4.1, 3.4.5) See Section 5.8 for additional discussion on air ingress.

4.2.5 Water Ingress

Water ingress is a separate class of events in which the break occurs at a pressure boundary at a location that might result in the admittance of water, e.g., precooler or compressor intercooler. Water ingress into the coolant system and subsequent transport to the core can result in positive reactivity insertion and graphite corrosion [18], [19]. This was a fundamental concern with the steam cycle HTGR, since the pressure in on the secondary side was sufficiently high to make this transient credible. With the GT-MHR and the PBMR, this event is not very likely since the source of water are the pre-cooler and intercooler heat exchanges are at significantly lower pressures. A precooler tube rupture was analyzed for the GT-MHR and shows that the helium would flow into the precooler system resulting in a precooler isolation due to overpressure. [18]

4.3 Events of Interest

The events of interest (also termed accidents) are those that can result in a release of radioactivity and therefore involve a breech of the reactor coolant pressure boundary. The events described illustrate the above discussion regarding break size effect. These events are:

- Single Bypass Valve Pipe Line Failure
- Depressurized Conduction Cooldown

4.3.1 Single Bypass Valve Pipe Line Failure

The bypass pipe line failure described for the GA GT-MHR [18] has an effective leak area of 0.05 m^2 (0.55 ft^2) or a hole size of about 25 cm (10 inches). With a hole size that large and an initial 7 MPa (1000 psi) driving force, the depressurization is rapid and is over within 60 seconds. Pressure buildup results in opening of dampers and release of helium gas with immediately available radioactive nuclides to the environment. As discussed above, further transport of fission products is dictated by thermal expansion of the gas. Maximum temperature of the core is not expected to significantly differ from the normal operating temperature of 850ºC. It should be noted that this analysis assumed that in addition to the safety grade RCCS, the Shutdown Cooling System, which is a nonsafety grade system, was available. The cummulative I_{131} release to environment is 0.65 Curie.

4.3.2 Depressurized Conduction Cooldown

This event involves a very small unisolable leak of 0.48 cm^2 (0.075 in^2), which is slightly larger than the diameter of a pencil. [18] All heat removal by is through the RCCS. Release of helium is into the confinement followed by venting to environment through dampers that open at approximately 7 kPa (1 psi) differential. Depressurization takes place in 31 hours. Gas continues to escape from the vessel due to the resulting thermal expansion resulting from core heatup and will last until the average core temperature has peaked. The temperature profile is shown in Figure 4-2. Note that the temperature peak occurs at about 70 hours. At that point decay heat production becomes less than the RCCS removal rate. Cummulative I_{131} release to environment is 3.35 Curie.

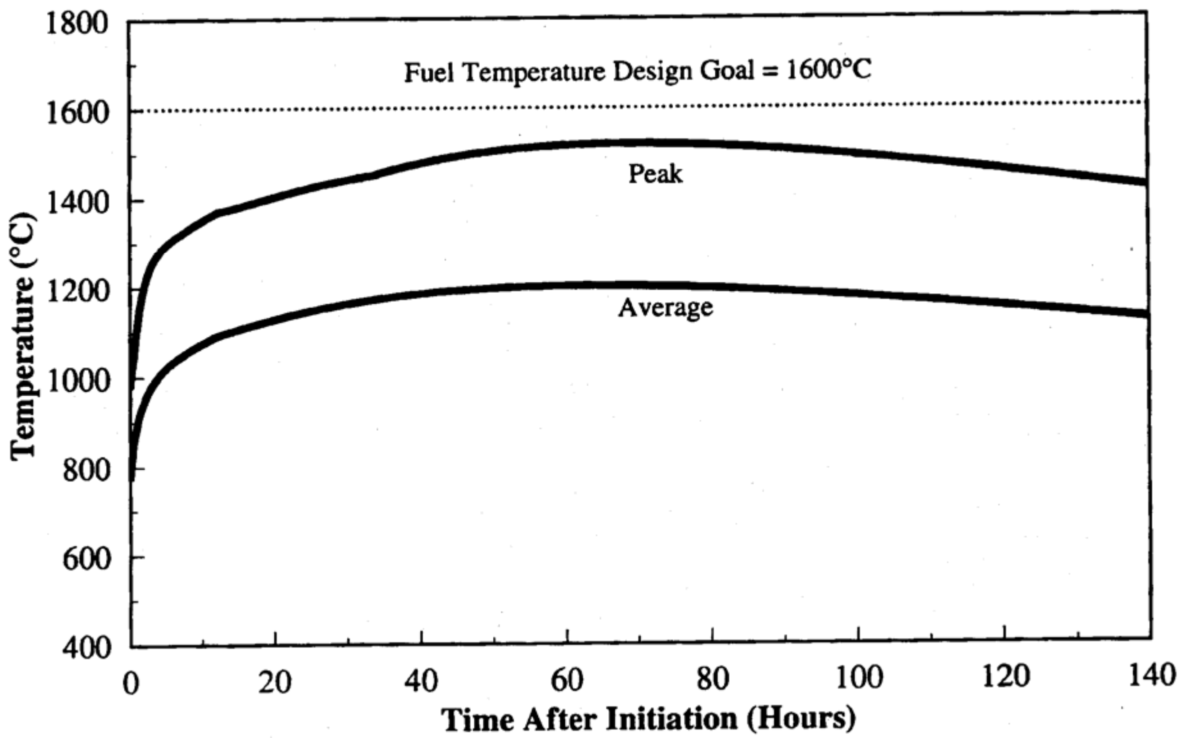

Figure 4-2
Depressurization Conduction Fuel Temperature [18]

4.3.3 Comparison of Events

As expected, smaller breaks result in larger releases. Figure 4-3 [18] provides graphic comparison of the dynamics of the transport mechanisms. Figure 4-3 shows the movement of I_{131} from the fuel to the primary coolant to the reactor building (containment) and finally the environment. As would be expected, the amount of activity becomes smaller as the helium goes from location to location. This is due to different phenomena, e.g., plate out on the confinement walls.

Note that in both cases I_{131} is shown to be available at the same rate. This is probably a conservative assumption, since the fuel temperature remains essentially at the normal operating temperature during the entire event. The dynamics of the break size then dictate the quantity and rate of transport.

For the bypass line break, release to the environment is immediate and continues throughout the event at the same rate, until about 7 hours when the effects of expected delayed accident source terms are apparent.

For the small break leak, release begins about 4 minutes and continues to increase, leveling off at about 30 hours. This reflects the effects of the RCCS to cool the reactor cavity.

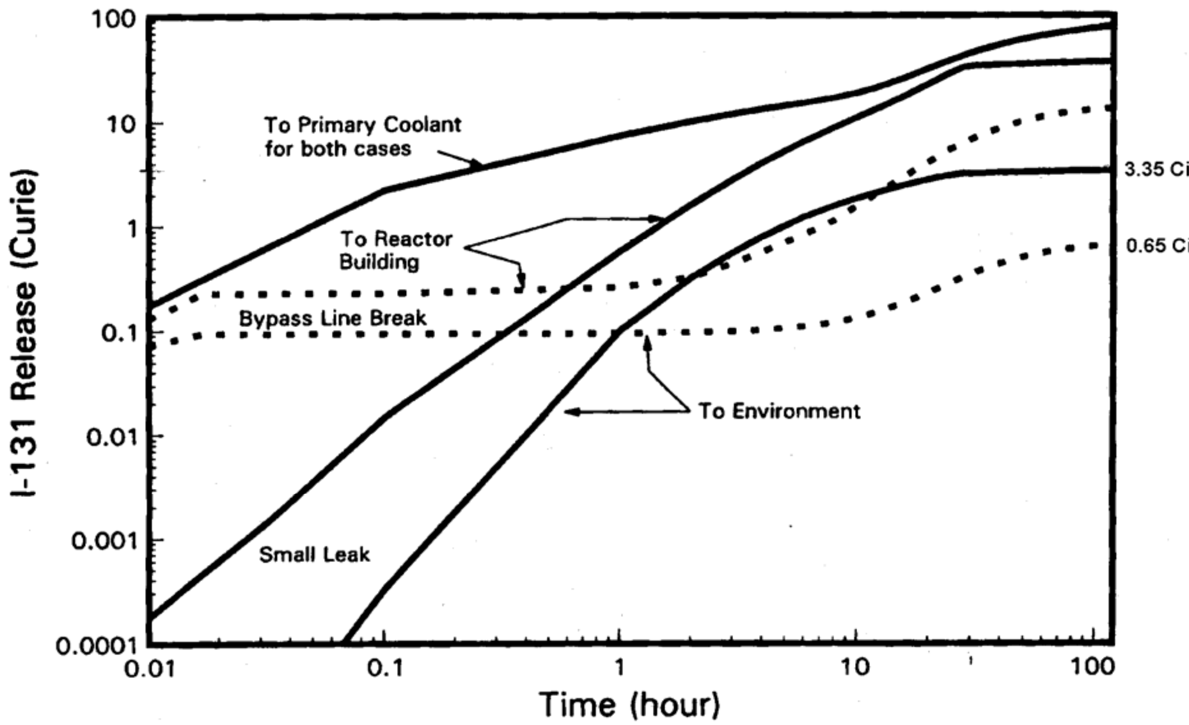

Figure 4-3
I-131 Cummulative Release to Environment During Bypass Line Break and Depressurization Conduction Accident [18]

4.4 Challenges

A significant challenge to accident analysis is the additional knowledge needed to understand better the effects of air ingress. This is described in Sections 5.8 and 5.11.

5
SITING CONSIDERATIONS

5.1 Introduction

Siting considerations are important in assuring that radiological doses from normal operation and postulated accidents will be acceptably low, that natural phenomena and potential man-made hazards will appropriately accounted for in the design of the plant, that site characteristics are such that adequate security measures to protect the plant can be developed, and that physical characteristics unique to the proposed site that could pose a significant impediment to the development of emergency plans are identified. The commission has also had a long standing policy of siting reactors away from densely populated centers. [20]

Of the above siting considerations, the following are relevant to current containment designs:

- Radiological doses from normal operation and postulated accidents will be acceptably lows.
- Development of emergency plans.

5.2 Radiological Dose

The determination of radiological dose includes the following steps:

1. calculation of source terms
2. applying containment leakage rate and any fission product cleanup systems intended to mitigate the consequences of the accidents to determine the amount of the source terms released to the atmosphere
3. calculation of atmospheric dispersion using applicable site characteristics, including site meteorology
4. calculation of dose to persons within certain zones in the vicinity of the facility

5.3 Source Terms Calculation

Source terms characterize the composition and magnitude of the radioactive material, the chemical and physical properties of the material, and the timing of the release to the containment following an accident. [21]

5.3.1 Calculation Methodology

As discussed in Section 3.6, the NRC will allow the used of scenario-specific source terms for future reactors. Designers, however, will be required to develop the technical bases (including experimental data) to support their proposed source terms. The use a conservative source term, as was the practice using TID 14844 [22] will also be permitted, provided the use of such a source term does not result in design features or operational limits that could detract from safety.

5.3.2 Barriers

The three barriers to source terms are the PCPB, the fuel and a combination of the fuel-compact matrix/graphite fuel block (or the the pebble matrix for the PBMR). Accident scenarios described in Section 4 assume that the PCPB is breached. This is highly unlikely as described below. The remaining barriers to release of radioactive material are not the leak-tight barriers that characterize the fuel cladding found in LWRs. The barriers function to delay escape of the material. Their effectiveness is highly dependent on fuel temperature.

5.3.3 Primary Coolant Pressure Boundary

As was discussed in Section 4, the break size in the PCPB plays a key roll in determining the source term quantity, largely due to the fuel temperature achieved during the event. The reliability of the PCPB will largely depend on the use of appropriate materials and construction methodologies. Overall construction will be in accordance with the ASME Code Section III and meet Class 1 requirements. The use of this Code will ensure conservatisms are applied that will result in a very low probability of PCPB failure.

5.3.4 Fuel

The reliability of the fuel will provide challenges. The delayed accident source term radionuclides assumed in justifying a confinement requires the fuel to have intrinsically reliable and predictable barriers to fission product release. These barriers function primarily to delay, rather than to stop, the fission products. Hence the barrier performance is different than the LWR, which uses a metal leak-tight cladding to prevent further migration of the fission products. A discussion of fuel reliability is found in Section 5.7.

The barrier function is accomplished through the use of a coated fuel particle. [17] The coatings consist of three isotropic (tri-isotropic or TRISO) material layers. An electron microscopic view of a TRISO coated fuel particle (TCFP) is shown in Figure 5-1. The diameter of the particle is either 880 μm (0.035 inches) or 800 μm (0.032 inches) depending on the type. The TRISO coatings consist of the inner and outer pyrolytic carbon coatings (I-PyC, O-PyC) and the silicon carbide coating (SiC). Also shown is the fuel particle itself (the kernel) and a buffer layer that permits the swelling of the fuel kernel due to irradiation.

While not directly a barrier, the fuel kernel does assist the process by retaining much of the fission products. Generally speaking, the efficiency of retention decreases with temperature and/or fuel burnup (energy produced or MWd/MTU).

As discussed above, the TRISO layers are not leak-tight barriers in the sense of LWR fuel cladding, but rather are characterized by low solubilities and diffusion coefficients for fission metals. The SiC coating acts as the primary barrier to the release of these metallic fission products. The PyC coatings are partially retentive of cesium (Cs) at lower temperatures, but provide little holdup of silver (Ag) or strontium (Sr) which tend to plate out during plant operations. This is why the Ag and Sr make up a substantial portion of any plate out release during an accident.

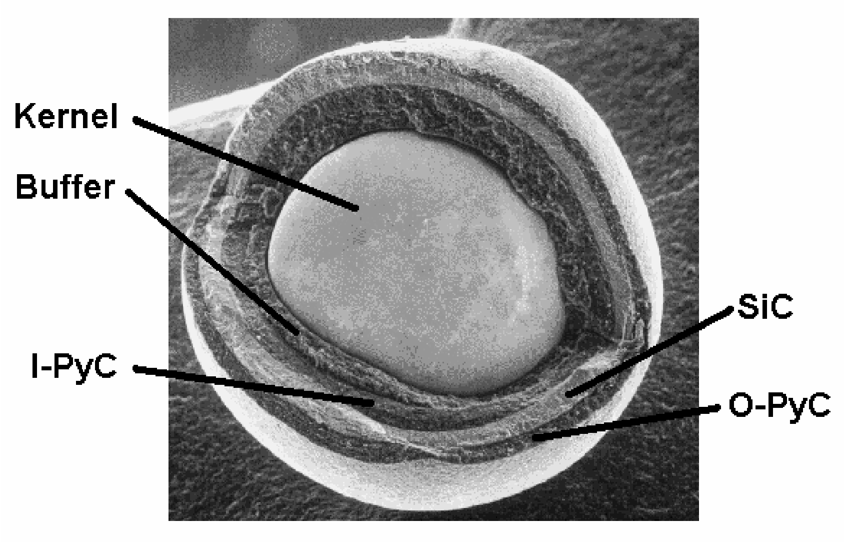

Figure 5-1
TRISO-Coated Fuel Particle [23]

5.3.5 Fuel Compact/Fuel Element Graphite/Pebble Matrix

With a prismatic core, the fuel-compact matrix and the graphite fuel block collectively are the next barrier. With a pebble-bed core, the analog is the pebble matrix, including the unfueled outer shell. The matrix is relatively porous and provides little holdup of the fission gases which are released from the fuel particles. However, the matrix is a composite material which has a high content of amorphous carbon, and this constituent of the matrix is highly sorptive of metallic fission products, especially Sr. While the matrix is highly sorptive of metals, it provides little diffusional resistance to the release of fission metals because of its high interconnected porosity .

The fuel element graphite, which is denser and has a more ordered structure than the fuel-compact matrix, is somewhat less sorptive of the fission metals than the matrix, but it is much more effective as a diffusion barrier than the latter. The effectiveness of the graphite as a release barrier decreases as the temperature increases.

5.4 Containment Leakage Rate and Fission Product Cleanup Systems Effect

Applying containment leakage rate and any fission product cleanup systems effects intended to mitigate the consequences of the accidents determine the amount of the source term released to the atmosphere. For the confinement, the leakage rate has been estimated at 1% of the confinement volume per day. Some designs assume some cleanup for small breaks and that effect would be factored in to the final source term available for atmospheric dispersion.

5.5 Atmospheric Dispersion

Atmospheric dispersion of the source term calculated above would be determined. This would use applicable site characteristics as well as meteorology. Generally speaking much of this determination is in accordance with Reference [24].

5.6 Zone Dose Calculation

Once the effect of dispersion are calculated then the amount of dose to persons within certain zones in the vicinity of the facility is determined. These zones are the exclusion area and low population zone. The criteria are [25]:

- An individual located at any point on the boundary of the exclusion area, for any 2 hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 25 rem total effective dose equivalent (TEDE).
- An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage) would not receive a radiation dose in excess of 25 rem TEDE.

Currently the determination of zones is controlled by NRC Regulations. See Section 3.7.

5.7 Fuel Reliability

5.7.1 Experience

The HTGR fuel draws upon fuel fabrication experience demonstrated in the U.S., the U.K., Germany, and Japan over the past 35 years. In Germany, substantial quantities of coated particle fuels have been fabricated by NUKEM since 1965. More than 10,000 kg of coated particles were made for the AVR and Thorium High-Temperature Reactor before the NUKEM fuel manufacturing facility was decommissioned in 1990. In the U.S., fuel assemblies for the Peach Bottom Unit 1 and the Fort St. Vrain initial cores and reloads were manufactured at the GA fuel production facility in San Diego. For two Peach Bottom cores, about 3,500 kg of pyrocarbon-coated fuel particles were fabricated into more than 1,600 fuel element assemblies. For the Fort St. Vrain initial core and three reload segments, about 30,000 kg of TRISO-coated uranium and thorium carbide particles were fabricated into 2,250 fuel element assemblies.

5.7.2 New Demands

While fuel performance was satisfactory, new demands such as higher burnups up to 120,000 Mwd/MTU, require that the fuel manufacturing processes for the must be capable of producing fuel of higher quality than Fort St. Vrain.This means reducing the defective particle fraction in as-manufactured fuel. The results of the extensive fuel qualification program carried out in the German GCR program have demonstrated the feasibility of fabricating high-quality fuel capable of meeting the needed performance requirements. Coated fuel particles tested in support of the Japanese HTTR program have also shown good performance for fuel with low defect levels at low burnups and moderate fast fluence (See Section 5.7.5).

5.7.3 Required Tests

Much of the German irradiation data is for uranium dioxide ($U0_{2)}$ fuel irradiated to burnups that are substantially lower than the design burnup for HTGR fissile particles; however, there is compelling theoretical and experimental evidence to support the superiority of uranium oxycarbide (UCO) fuel for reactor service to higher burnup. This is also substantiated by the excellent performance of UCO fuel in German capsule tests (FRJ2-P24) using German coating technology. Nevertheless, because irradiation performance data for TRISO-coated UCO in fuel compacts is limited and because there is essentially no post-irradiation heating test data for high-quality, high-performance UCO fuel, irradiation testing and post-irradiation accident simulation tests will be required to demonstrate and qualify UCO fuel for HTGR service conditions and accidents. These tests, as outlined in Reference [26], contain the following objectives that must be satisfied to meet the overall objective of qualifying the fuel and the fuel fabrication processes for the HTGR prototype plant:

- Fabricate, irradiate, and perform post-irradiation simulated accident conditions testing of *demonstration test* fuel to demonstrate that TRISO-coated UCO fuel manufactured under reference process conditions and meeting Fuel Product Specification requirements perfonns satisfactorily under HTGR normal operating conditions and accident conditions.
- Fabricate, irradiate, and perform post-irradiation simulated accident conditions testing of *qualification test* fuel to verify that TRISO-coated UCO fuel manufactured under reference process conditions and meeting Fuel Product Specification requirements performs satisfactorily under the full range of HTGR normal operating conditions and accident conditions.
- Conduct fuel testing and fission product transport technology development, to the extent necessary, to generate the data needed to validate fuel performance and fission product transport models in support of licensing of an HTGR prototype plant.
- Design and construct a fuel fabrication pilot plant consisting of production scale process equipment.
- Demonstrate that fuel meeting HTGR fuel product specification requirements can be manufactured in the pilot plant.
- Fabricate, irradiate, and perform post-irradiation accident conditions testing of *proof test* fuel to verify that the pilot plant fuel performs satisfactorily under HTGR normal operating conditions and accident conditions.

5.7.4 Quality Control

Crucial to fuel reliability will also be quality control techniques that demonstrate that the HTGR fuel meets the new demands with high confidence. These techniques include characterization of fuel, including the detection of SiC defects (the primary fission product barrier), characterization of I-PyC and O-PyC layer microstructure, direct measurement of I-PyC coating layer and stoichiometry of UCO kernels. These measures should be nondestructive, with high throughput rates (potentially high enough to make 100% inspection feasible).

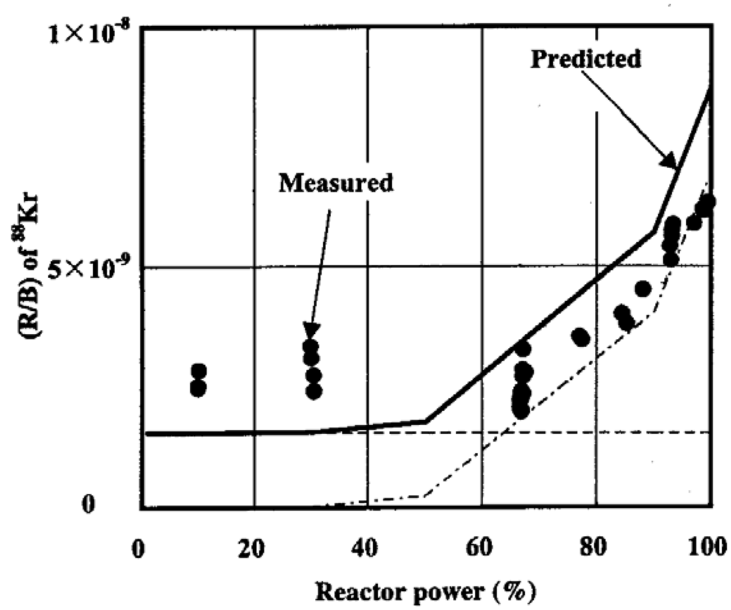

Figure 5-3
R/B Ratio for Kr_{88} in HTTR Testing [27]

5.7.5 *Fuel Test Results at the HTTR in Japan*

The High Temperature Engineering Test Reactor (HTTR) in Japan went critical in 1998. Rise-to-power tests began in September 1999 with a final power of 30 Mw and core outlet coolant temperature of 850°C being attained in December 2001. After carrying out testing at 850°C, the coolant temperature was raised again reaching 950°C in April 2004.

As part of the rise-to-power tests, measurements were taken that were used to evaluate the release rate of fission product gases. The results revealed that the detected gases in the primary coolant were Kr_{85m}, Kr_{87}, Kr_{88}, Xe_{133}, Xe_{135}, Xe_{135m}, and Xe_{138}. Of these Kr_{88} was further analyzed to determine the Release Rate to Birth Rate Ratio (R/B). The results are shown in Figure 5-3. [27]

The predictions are well within an order of magnitude of actuals showing that the prediction approach is sound. The small differences were explained and will be factored in the next version of the model. R/B ratios on the order of 10^{-8} are considered acceptable [23],[28].

Results of tests to be conducted at 950°C are not available at this time.

5.7.6 *NRC Activities*

In July 2004, the NRC issued NUREG/CR-6844 [29] to deal with the reliability questions regarding TRISO-coated particle fuel to be used in the next generation of gas-cooled reactors. This publication describes the TRISO Phenomena Identification and Ranking Table (PIRT) program. The PIRT will be used to (1) identify key attributes of gas-cooled reactor fuel manufacture which may require regulatory oversight, (2) provide a valuable reference for the review of vendor fuel qualification plans, (3) provide insights for developing plans for fuel safety margin testing, (4) assist in defining test data needs for the development of fuel performance and fission product transport models, (5) inform decisions regarding the development of NRC's independent reactor fuel performance code and fission product transport models, (6) support the development of NRC's independent models for source term calculations, and (7) provide insights for the review of vendor fuel safety analyses. To support these objectives, the NRC commissioned a PIRT panel to identify and rank the factors, characteristics, and phenomena associated with TRISO-coated particle fuel. PIRTs were developed for (1) Manufacturing, (2) Operations, (3) a Depressurized Heatup Accident, (4) a Reactivity Accident, (5) a Depressurization Accident with Water Ingress, and (6) a Depressurization Accident with Air Ingress.

5.8 Air Ingress Affect on Source Terms

Air ingress can pose challenges to core integrity and hence affect source terms. However there is controversy over the credibility of this event [30]. Because it is controversial, the NRC has determined that it must be considered as one of the selected credible events having the potential for high consequence source terms which must be considered in the containment design. (See Section 3.5.6) The problem with air ingress is graphite oxidation and the creation of a transport force to carry fission products away from the reactor. Since the heating would pressurize the confinement, it is possible that the source term would be higher than predicted for the depressurized conduction accident. There has been some controversy over the use of the term *burning* and some who claim it does not burn. This is probably a question of semantics. Carbon does combine with oxygen and is exothermic, i.e., heat is released. How it oxidizes is conditioned on both graphite temperature and air flow conditions. This oxidation can result in both CO and CO_2.

Air ingress scenarios require that the helium escape from the coolant system to be replaced with air. This could not occur until the reactor was completely depressurized at which time the effects of bouyancy would become the driving force. It is obvious that both the helium escape and air ingress flow is a function of break size which must be adequate to permit flows in both directions. If there are two breaks, then of course the size requirements would be smaller. Figure 5-4 shows this effect as analyzed by GA for the GT-MHR. [31]

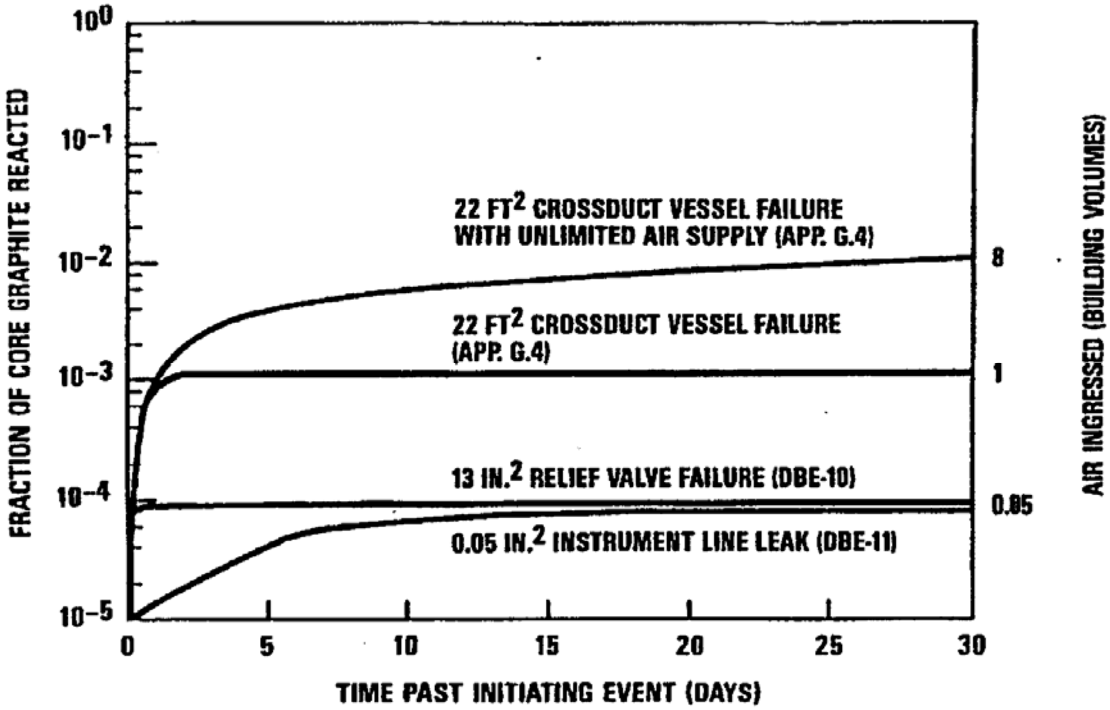

Figure 5-4
Graphite Degradation in Air Ingress Scenario [31]

Therefore, the question of the graphite burning effects requires investigation. At the High-Temperature Gas-Cooled Reactor Safety and Research Issues Workshop [30] the following were identified as research needs associated with air ingress accidents:

- kinetics of high temperature oxidation of graphite
- applicability and adequacy of codes
- termination of sequence
- fuel and fission product release
    - in helium
    - following air ingress
- applicability of current data base to recent graphite forms
- determination of event initiators
    - thermally induced fatigue
    - vibration induced fatigue
    - seismic events initiating accidents
    - embrittlement
    - corrosion
    - failure of turbo-machinery within the primary circuit

- testing recent graphite forms
- probabilistic risk assessment
- fuel particle testing

5.9 Water Ingress

The problem with water ingress is positive reactivity insertion and graphite corrosion [18], [19]. This was a fundamental concern with the steam cycle HTGR, since the pressure in on the secondary side was sufficiently high to make this transient credible. With the GT-MHR and the PBMR, this event is not very likely since the source of water are the pre-cooler and intercooler heat exchanges are at significantly lower pressures.

A precooler tube rupture was analyzed for the GT-MHR and shows that the helium would flow into the precooler system resulting in a precooler isolation due to overpressure. [18]

5.10 Plate Out

Another source of radioactivity are the elements that have plated out and are assumed to be available depending on the dynamics of the pressure boundary stresses during depressurization.

Typically these releases are small or non-existant for smaller breaks and increase with the sized and location of the break. [18] reports that while earlier SC-MHTGR depressurization transients involving relief valve pipe line failures could result in plate-out release, the GT-MHR did not experience it in its most severe line break scenario (Single Bypass Valve Pipe Line Failure).

5.11 Challenges

The determining factor for siting will be the source term. Crucial to source term is fuel reliability. Because irradiation performance data for TRISO-coated UCO in fuel compacts is limited and because there is essentially no post-irradiation heating test data for high-quality, high-performance uranium oxide fuel (UO_2 and UCO), irradiation testing and post-irradiation accident simulation tests will be required to demonstrate and qualify this fuel for HTGR service conditions and accidents. These tests are as outlined in Reference [26].

Also essential will be quality control techniques that will demonstrate that the HTGR fuel meets the new demands with high confidence. These measures should be nondestructive, with high throughput rates (potentially high enough to make 100% inspection feasible).

The effect of air ingress and subsequent graphite oxidation must be investigated further. This is a highly controversial issue and is an explicit consideration in the NRC's thinking about the design of the containment. Many issues have been identified including kinetics of high temperature oxidation of graphite, applicability and adequacy of codes, termination of sequence fuel and fission product release, applicability of current data base to recent graphite forms, determination of event initiators, testing recent graphite forms, probabilistic risk assessment and fuel particle testing.

In the final analysis, the use of a confinement for the HTGR will rest on the ability for the industry to provide incontrovertible evidence that the fuel integrity will be robust enough to provide the advertised barrier to fission product release. Without that confidence, provided by low probabilities with high confidence coefficients (meaning lots of realistic testing), it will be difficult to justify the fuel as a barrier and the requirement for a low-leakage containment will exist.

6
COST/ECONOMICS: INITIAL AND ONGOING

6.1 Introduction

This chapter outlines those differences that could be considered when comparing the confinement with the containment. While theoretically confinements should be less costly, the cost differential is not easily determined without knowing the final design requirements. This would include how many containment/confinement penetrations are involved, allowable leakage rates, other test requirements, etc. Similarly it is premature to determine operating costs. Again the expectation is that they should be less, because the test requirements should be less demanding. However, until the final design requirements are known operating costs cannot be known.

6.2 Construction Costs

A construction cost comparison was performed in 1985 [1]. Four reactor designs were analyzed: two with a confinement and two with a containment. The difference in cost between a 1170 Mwt reference design using a containment and a 1260 Mwt design using a confinement was less than 1%. While the confinement was to not be leak tight, it still needed the robustness of the containment, in order to offer barriers to missile-type events. These missile-type events did not include commercial aircraft crashes since they were not considered credible events at that time. As discussed in Section 2, this may not be valid for future designs and additional costs may be incurred.

6.3 Operating Costs

6.3.1 Leakage Testing

If a savings is expected with the confinement, operating costs holds the greatest potential. Present day containment leakage testing is governed by 10CFR50 Appendix J. It is expected that the same types of tests will be performed for the HTGR confinement. The reduction in costs is expected to involve the number (fewer for the confinement) and acceptance criteria (not as severe). Appendix J requires three types of tests: A, B and C. The Type A is an overall integrated leak rate test (ILRT) for the entire containment. Type A testing normally occurs every ten years and is performance-based. It is always performed during an outage. Type B and C tests are penetration specific and are referred to as local leak rate tests (LLRT). Type B for penetrations involve elastomeric sealing and Type C for all others, e.g. valve seats. LLRT testing is performed at least every two years, except for personnel hatches which are performed every six months (full test) or within 72 hours of use and every 72 hours while in use (seal test only). LLRTs are performed both during operation or outage depending on the nature of the system involved. [32]

Costs for Type A Testing

The costs for Type A test costs involve test procedure preparation and maintenance, pressurization equipment rental and set up, penetration line-ups (to simulate accident conditions and verify alignment), buying or renting the equipment needed to accurately measure the leakage rate (pressure, temperature, humidity), equipment to correlate this information and produce the leakage rate, penetration line-ups (for normal conditions), equipment removal, data analysis and report preparation.

For the LWR containment building leakage is typically on the order of 0.1% of the containment volume (mass) per day at accident pressure. The volume is typically on the order of 57,000 cubic meters, (2,000,000 cubic feet). Accidents pressures are typically can vary from about 70-350 kPa (≈10-50 psid) depending on containment design. It is obvious that the leakage pathway is considerably larger for the confinement.

Testing is always on the outage schedule critical path. Maintenance in the containment is not possible. It takes about 3-5 days. Therefore there will be revenue losses due to extended outage times as well as considerable amount of resources involve. At peak activity (line-ups) there could be 10 or more personnel per 12 hour shift working for two days. In order to test to the peak accident pressure stated above requires several large air compressors to attain the required pressure in a timely manner. Because this test is only conducted every ten years, staffing would also include hiring an experienced contractor who specializes in Type A tests. Reference [32] reported the costs of Type A tests were estimated to be about $1.89 million per test, and included equipment cost rentals, labor costs, and replacement power costs.

For the HTGR confinement, equipment costs should be much less. For the confinement described in Section 2, the quantity of air required would be about 1/2000 of that needed for the LWR containment. A single small compressor would be adequate. In addition since there would be fewer penetrations and a smaller volume, the preparation time for line-ups and test duration will be shorter. Critical path should be on the order of one day. The actual impact on cost, however, is difficult to gauge until final designs are established.

Type B and C Testing

Type B and C test costs involve test procedure maintenance, equipment line-ups (to simulate accident conditions and connect test equipment), testing and equipment restoration (normal conditions). At a typical PWR each penetration will take approximate 35-40 person-hours to perform a single LLRT. Type B testing for personnel hatches usually involves about 10 person-hours for door seals and 20 person-hours for the six month full test. From a staffing perspective, every plant has a LLRT coordinator who maintains procedures, conducts all the tests (outage and on-line), analyzes tests, maintains test equipment, etc. Usually there are a total of four persons per shift conducting the tests. Reference [32] reported that the costs for a full battery of Type B and C tests for a typical LWR was estimated to be about $165,000, and included LLRT crew labor and support from plant staff. This was based on an average of 110 penetrations (90 PWR, 170 BWR). This does not count the maintenance cost for repair of failed penetrations. The number of penetrations should be significantly less for the confinement and coupled with the lower test pressure should result result in fewer failures and less repair costs.

6.3.2 Inspections

Inspections of the general condition of the containment are presently required in accordance with subsections IWE and IWL of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code). While confinements should not have as stringent requirements as containments, these costs may not differ significantly.

6.4 Challenges

An early study showed that the construction costs of a confinement may not be all that different than for a containment. One of the causal factors was the requirement to withstand externally generated missile-type events. These missile-type events did not include commercial aircraft crashes since they were not considered credible events at that time. As discussed in Section 2, this may not be valid for future designs and additional costs may be incurred.

However, operating costs appear to have a potential for savings. It is reasonable to assume that the number of penetrations that require testing will be fewer. Because the confinement volume is about 20% of the typical containment volume and test pressures significantly lower, the air volume requirements, test pressures and leakage rates will be lower making testing less challenging. These factors and the reduced need for maintenance and large numbers of test equipment, reduced critical path should be positive in this outlook. However, it is not possible to know these savings until the final designs are known.

7
CHALLENGES

7.1 Confinement Structure

The confinement designs, like their containment predecessor, are not designed for a commercial airplane crash. However, in today's terrorist threat environment, the need for designing for a commercial airplane crash may be needed.

To limit pressure to 7 kPa (1 psi) differential, dampers will be needed. Dampers are like check valves and are required to change position to perform their safety function. Damper action will need to be tested periodically to confirm its reliability. These tests will probably be conducted in conjunction with the LLRT tests described in Section 6.

7.2 Regulatory Considerations

The NRC is currently developing policy related to licensing non-light-water reactor designs. The NRC staff has been working at the direction of a Commission Staff Requirements Memorandum issued in July of 2003. Seven issues were identified and two directly affect the decision on containment design; defense-in-depth and containment functional requirements and criteria.

The most recent document, issued in January 2005, has presented the staff's latest thinking and is based on stakeholders input and NRC directives regarding the licensing of advance nuclear reactors. The functional requirement criterion for the containment is as follows:

> "The containment must adequately reduce radionuclide releases to the environs to meet the onsite and offsite radionuclide dose acceptance criteria for the events selected for the event categories (including within the design-basis category, selected credible events having the potential for high consequence source terms) and have the capability to establish controlled leakage and controlled release of delayed accident source term radionuclides.

This functional requirement includes credible events with potential for high consequence source terms (so-called "cliff-edge") events in the design-basis that are not presently being considered. This could require additional technology development to support the source term calculations for these events. Also, depending on the analysis results, this option might require design-related enhancements involving incremental costs.

This functional criterion also contains a prescriptive requirement that the containment have the capability to establish a controlled leakage and a controlled radionuclide release capability. The NRC feels that this capability ensures that the containment will provide a significant deterministic element of *defense-in-depth* to controlling radioactive releases, should the other mechanistic barriers and obstacles to fission product transport provided by the fuel, core materials and reactor coolant system not perform as expected or should unanticipated events involving a larger than expected accident source term occur. This element is independent of the performance of the other mechanistic barriers and, for some designs, also has the potential to prevent or mitigate certain kinds of accidents (e.g., HTGR air ingress).

Since it will have a prescriptive requirement to have the capability to establish controlled leakage, this design has been referred to as a “hybrid containment” because it would allow the initial RCS depressurization to vent directly to the environs for loss of reactor coolant pressure boundary events, but would require that it have the capability to establish a controlled, low leakage, thereafter. This option would limit the volume of air ingress available for core oxidation and would limit the volume of air out-leakage available for radionuclide transport to the environs of the delayed radiological source term.

The NRC’s current view is that not all HTGR designs presently meet this functional requirement and that design changes would likely be required. For such plants, this option would also likely require structural changes to the confinement design in order to meet structural stress (i.e., ASME) limits and upgrades to the vent system in order to assure a reliable vent path reclosure capability. This would add to the design and construction costs of the VLPC. It would also have a impact on testing and confinement maintenance.

This option would reduce concerns related to maintaining fuel quality and fuel performance during normal operation and accidents over the life of the plant. However, by requiring the containment to have an additional capability to reduce releases, it might reduce the incentive to emphasize accident prevention in designs, thereby potentially having an adverse affect on plant safety.

While not a challenge, it should be noted that the EPZ sizes will not change with the intial installations. The NRC Staff has indicated that for new reactor designs, it is too early to identify specific conditions that would allow a reduction in the 10-mile plume exposure pathway EPZ. Until sufficient experience is gained on any prototype reactor, a case-by-case basis will probably be used to evaluate whether a requested reduction in the size of the EPZ can be allowed. This criterion would also apply to the 50-mile ingestion control pathway EPZ.

7.3 Design Basis Events

A significant challenge to accident analysis is the additional knowledge needed to understand better the effects of air ingress. This is described in Sections 5.8 and 5.11.

7.4 Siting Considerations

The determining factor for siting will be the source term. Crucial to source term is fuel reliability. Because irradiation performance data for TRISO-coated UCO in fuel compacts is limited and because there is essentially no post-irradiation heating test data for high-quality, high-performance uranium oxide fuel (UO_2 and UCO), irradiation testing and post-irradiation accident simulation tests will be required to demonstrate and qualify this fuel for HTGR service conditions and accidents. These tests are as outlined in Reference [26].

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A
ACRONYMS

Acronym	Meaning
ACRS	Advisory Committee for Reactor Safety
Ag	Silver
Cs	Cesium
DID	Defense-in-Depth
EPZ	Emergency Planning Zone
FSV	Fort Saint Vrain
GA	General Atomics
GT-MHR	Gas Turbine Modular Helium Reactor
HTGR	High Temperature Gas-Cooled Reactor
HTTR	High Temperature Test Reactor
ILRT	Integrated Leak Rate Test
I-PyC	Inner Pyrolytic Carbon Coating
LLRT	Local Leak Rate Test
LWR	Light Water Reactor
NRC	U S Nuclear Regulatory Commission
O-PyC	Outer Pyrolytic Carbon Coating
PBMR	Pebble Bed Modular Reactor
PCPB	Primary Coolant Pressure Boundary
PIRT	Phenomena Identification and Ranking Table
PRA	Probability Risk Assessment
R/B	Release/Birth Ratio

Acronym	Meaning
RCCS	Reactor Cavity Cooling System
SiC	Silicon Carbide
Sr	Strontium
SSC	Structures, Systems and Components
TCFP	TRISO Coated Fuel Particle
TEDE	Total Effective Dose Equivalent
TRISO	Tri-isotropic
UCO	Uranium Oxycarbide
UO_2	Uranium Dioxide
VHTR	Very High Temperature Gas-Cooled Reactor
VLPC	Vented Low Pressure Containment

B
WORKSHOP SUMMARY - KEY POINTS

The staff held public workshops on November 19, 2003, and January 14, 2004, to discuss and solicit comments on the staff's initial efforts to develop functional performance requirements and criteria for the containment design of new non-light water reactors (LWRs).

The scope of the workshops addressed options and issues for developing criteria to address the following potential containment functional areas:

1. Containing fission products
2. Preventing and mitigating severe core damage accidents
3. Removing heat during accidents
4. Protecting safety equipment from natural phenomena, and dynamic effects
5. Protecting onsite workers from radiation
6. Physically protecting vital equipment (security events)

Key points raised by stakeholders at the workshops are summarized below.

<u>Potential Containment Functional Performance Areas</u>

- Workshop participants stated that there needed to be a clarification of reactor safety functions, containment safety functions, and where the two overlap.
- NRC should look at how to apply these safety functions to radiation outside of the core.
- Specific aspects of the containment building can only be addressed in the consideration of a specific technology and design, and generic requirements are not very practical at a low level.
- NRC should make functional performance requirements technology-neutral (within the non-LWR arena), but functional performance criteria should be done on a design-specific basis, and for now, the NRC should focus on high-temperature, gas-cooled reactors (HTGRs) in developing functional performance criteria.
- Functional performance areas 1 and 2 should be combined and modified so to "manage the release of fission products during accidents."
- Many of the functional performance areas are not exclusive functions of the containment and can be accomplished by other systems.
- "Containment" should refer to a structure and "containment building" and "reactor building" can be used interchangeably; i.e., the use of the word "containment" does not necessarily imply a building with pressure-retaining capability.

Contain Fission Products

This functional area should be combined with the "Prevent and Mitigate Severe Core Damage Accidents" area, with some changes to the wording.

- NRC should not require a containment building to be pressure-retaining, or to have the capability to filter fission products. The regulations should focus on what dose acceptance criteria need to be met outside of the reactor/ containment building.
- NRC should give credit to design features that enhance operator recovery.
- Functional performance requirements need to account for the role of time in assessing functional performance requirements, and consider time available for taking mitigative actions.
- Stakeholders encouraged the NRC to understand the design philosophy of the new reactors and to take that (along with how to deal with very low core damage frequency) into consideration in determining the functional performance requirements.

Prevent and Mitigate Severe Core Damage Accidents

- NRC should allow the designers to have flexibility in dealing with uncertainties, and should not assume that the prevention of severe core damage (however it is defined) is a function of the containment building.
- If there is reasonable assurance of fuel quality, is a containment building a necessary element of defense-in-depth? Is a containment building with the capability to be pressure-retaining necessary just in case the fuel integrity is not as good as it was thought to be?
- There is no clear separation between prevention and mitigation. New reactor designs are not relying on just one thing to prevent accidents.
- NRC should focus on putting requirements on the conditions of an area immediately outside of the containment (or reactor) building, and not look at how those conditions are achieved. NRC should establish basic dose criteria that need to be met under accident conditions while allowing the designer flexibility in how to meet that criteria.
- In accounting for completeness uncertainties, the impacts on the integrated cost of the plant need to be considered.

Remove Heat During Accidents

- This function should not be assigned to the containment building as heat removal can be accomplished with other systems.
- Clarify whether this function is necessary for maintaining structural integrity of the containment building or whether it is important for the retention of fission products in the fuel.

- How much redundancy and diversity should be required for passive components?
- The reactor/containment building, no matter what it looks like, must accommodate and not interfere with heat removal and recovery actions for the purposes of maintaining fuel integrity, terminating fuel damage if such damage is underway, ensuring building and structural integrity, and facilitating recovery actions after an accident.

Protect Safety Equipment From Natural Phenomena and Dynamic Effects

- Stakeholders said they would prefer that this requirement not be assigned specifically to containment. The NRC should only require that provisions are provided within the systems structures, and components to protect against the adverse effects of natural phenomena.

Protect Onsite Workers from Radiation

- There need not be additional regulations for worker protection under accident conditions as the existing regulations are adequate. New plants should use something similar to the NEI Severe Accident Management Guidelines (SAMGs).
- This functional performance requirement should require the reactor (or containment) building to accommodate and not to interfere with recovery actions.

Physically Protect Vital Equipment (Security Events)

This topic was not discussed at the workshops.

The Electric Power Research Institute (EPRI)

The Electric Power Research Institute (EPRI), with major locations in Palo Alto, California, and Charlotte, North Carolina, was established in 1973 as an independent, nonprofit center for public interest energy and environmental research. EPRI brings together member organizations, the Institute's scientists and engineers, and other leading experts to work collaboratively on solutions to the challenges of electric power. These solutions span nearly every area of power generation, delivery, and use, including health, safety, and environment. EPRI's members represent over 90% of the electricity generated in the United States. International participation represents nearly 15% of EPRI's total R&D program.

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