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NGNP and Hydrogen Production Conceptual Design Study

Reactor Building Functional and Technical Requirements and Evaluation of Reactor Embedment

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BACKGROUND INTELLECTUAL PROPERTY

Section	Title	Description
A3	Evaluation of Reactor Building Pressure Response and Radiological Retention Capability Reference [14]	Radionuclide retention scoping calculations and source term assumptions
		Reactor Building pressure and temperature calculations
B1.2	DLOFC transient Analysis of RCCS Standpipes, PBMR Calculation TA00024/A issued 5 June 2004. Reference [27]	Scoping analysis of temperature transient without RCCS operating.
B1.2	Heat transfer calculation for Reactor Embedment Task Design Information Transmittal DIT001254 taking information from Calculation T001284 August 2008. Reference [28]	Scoping analysis of differences of heat transfer from reactor to exterior with soil and air.

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ACRONYMS

Abbreviation or Acronym	Definition
ACI	American Concrete Institute
ACRS	Advisory Committee on Reactor Safeguards
AEC	Atomic Energy Commission
ALARA	As Low As Reasonably Achievable
ALOHA	Areal Locations of Hazardous Atmospheres
ANPR	Advance Notice of Proposed Rulemaking
ANSI	American National Standards Institute
AOO	Anticipated Operational Occurrence
ASCE	American Society of Civil Engineers
ASME	American Society of Mechanical Engineers
ASTM	American Society of Testing and Materials
AVR	Arbeitsgemeinschaft Versuchsreaktor
BDBE	Beyond Design Basis Event
BLEVE	Boiling Liquid Expanding Vapor Explosion
BOD	Basis of Design
CBCS	Core Barrel Conditioning System
CCS	Core Conditioning System
CEPPO	Chemical Emergency Preparedness and Prevention Office
CFR	Code of Federal Regulations
CIP	Core Inlet Pipe
COL	Combined License
COP	Core Outlet Pipe
CP	Construction Permit
DBA	Design Basis Accident
DBE	Design Basis Event
DBT	Design Basis Threat
DC	Design Certification

Abbreviation or Acronym	Definition
DCD	Design Control Document
DEGADIS	Dense Gas Dispersion
DEGB	Double Ended Guillotine Break
DID	Defense-In-Depth
DLOFC	Depressurized Loss of Forced Cooling
DOE	Department of Energy
DPP	Demonstration Power Plant
DSA	Deterministic Safety Analysis
EAB	Exclusion Area Boundary
EPA	Environmental Protection Agency
EPRI	Electric Power Research Institute
EPZ	Emergency Planning Zone
FHSS	Fuel Handling and Storage System
GDC	General Design Criteria
HPB	Helium Pressure Boundary
HPF	Hydrogen Production Facility
HPS	Hydrogen Production System
HSS	Helium Service System
HTGR	High Temperature Gas Cooled Reactor
HVAC	Heating, Ventilation, and Air Conditioning
IBC	International Building Code
ICM	Interim Compensatory Measures
IDLH	Immediate Danger to Life and Health
IHX	Intermediate Heat Exchanger
INL	Idaho National Laboratories
LBE	Licensing Basis Event
LWR	Light Water Reactor
NGNP	Next Generation Nuclear Plant
NHSB	Nuclear Heat Supply Building
NHSS	Nuclear Heat Supply System

Abbreviation or Acronym	Definition
NIOSH	National Institute of Occupational Safety and Health
NOAA	National Oceanographic and Atmospheric Administration
NPH	Natural Phenomenon Hazard
NRC	Nuclear Regulatory Commission
NSR	Non Safety Related
OBE	Operating Basis Earthquake
PAG	Protective Action Guideline
PBMR	Pebble Bed Modular Reactor
PCDR	Pre-Conceptual Design Report
PCHX	Process Coupling Heat Exchanger
PCS	Power Conversion System
PHTS	Primary Heat Transport System
PIRT	Phenomenon Identification and Ranking Tables
PLOFC	Pressurized Loss of Forced Cooling
PRA	Probabilistic Risk Assessment
PRS	Pressure Relief System
PSP	Physical Security Plan
RB	Reactor Building
RBVV	Reactor Building Vent Volume
RCCS	Reactor Cavity Cooling System
RDC	Regulatory Design Criteria
RG	Regulatory Guide
RIM	Reliability and Integrity Management Program
RI-PB	Risk Informed – Performance Based
SC	Seismic Category
SC-H	Safety Class - High
SC-L	Safety Class - Low
SC-M	Safety Class - Medium
SG	Steam Generator
SGI	Safeguards Information

Abbreviation or Acronym	Definition
SHTS	Secondary Heat Transport System
SR	Safety Related
SRM	Staff Requirements Memorandum
SSC	Systems, Structures and Components
SSE	Safe Shutdown Earthquake
ST	Special Treatment
T&FR	Technical and Functional Requirements
TEDE	Total Effective Dose Equivalent
TLRC	Top Level Regulatory Criteria
TNT	Trinitrotoluene
URD	Utility Requirements Document
UVCE	Unconfined Vapor Cloud Explosion

SUMMARY AND CONCLUSIONS

This study develops technical and functional requirements (T&FRs) for the Next Generation Nuclear Plant (NGNP) and High Temperature Gas-cooled Reactor (HTGR) Reactor Building (RB), including consideration of reactor embedment. The following Summary and conclusions are organized into three parts. The first part (Part A) summarizes the overall technical and functional requirements for the reactor building and the evaluation of various design strategies for meeting these requirements. The second part (Part B) of the summary and conclusions identifies criteria and requirements relevant to determining the degree of reactor building embedment and evaluates and ranks embedment depth alternatives. The third part of the summary and conclusions focuses on the role that embedment can contribute to meeting T&FRs identified in the previous discussion.

Reactor Building Functions and Requirements (Part A)

A comprehensive set of technical and functional requirements for the NGNP Reactor Building was developed in this study for use in the Conceptual Design Phase to follow. These requirements were developed in a top down fashion starting with the NGNP Top Level Requirements identified in the Preconceptual Design Report (Reference [1]) and the following considerations:

- Role of the Reactor Building (RB) in the Pebble Bed Modular Reactor (PBMR) NGNP safety design approach
- Review of NRC requirements and guidance applicable to the Reactor Building including those for safety and physical security
- The Risk-Informed and Performance Based Licensing approach envisioned for the PBMR NGNP (References [3] through [6])
- Selection of Preliminary Licensing Basis Events (LBEs) for the NGNP

The reactor building is designed to house and protect the Systems, Structures, and Components (SSCs) within the Nuclear Heat Supply System (NHSS), which include, among other SSCs, the Reactor, Primary Heat Transport System (PHTS), portions of the Secondary Heat Transport System (SHTS), Helium Services System (HSS) and the Fuel Handling and Storage Systems (FHSS). In devising a comprehensive set of Reactor Building functions, the following categories of functions were considered:

- Safety Functions
    - Required Safety Functions: those functions that are necessary and sufficient to meet the dose limits for Design Basis Events (DBE) and deterministically selected Design Basis Accidents (DBA)

    - Supportive Safety Functions: all other functions that contribute to the prevention or mitigation of accidents and support the plant capabilities for defense-in-depth

- Physical Security Functions: functions to protect the reactor and vital equipment from design basis threats associated with acts of sabotage and terrorism
- Non-safety/security Functions: those functions necessary for plant construction, operation, maintenance, access, inspection, worker protection, and control of routine releases of radioactive material during normal operation

The RB shall perform the following required safety and security functions:

- House and provide civil structural support for the reactor and all the SSCs within the scope of the NHSS (building geometry/space)
- Resist all structural loads as required for required safety functions allocated to NHSS SSCs include those associated with:
    - Prevention and mitigation of releases of radio-nuclides
    - Control core heat generation
    - Control of core heat removal
    - Control of chemical attack by protecting the PHTS Helium Pressure Boundary (HPB) integrity
    - Maintenance of core and reactor vessel geometry
    - Maintenance of reactor building structural integrity
- Protect the SSCs within the NHSS that perform required safety functions from all internal and external hazards as identified in the LBEs
- Provide physical security of vital areas of the NHSS against acts of sabotage and terrorism

The RB shall perform supportive safety functions involving the retention of radio-nuclides released from the PHTS HPB and the limiting of air ingress following HPB leaks and breaks. These supportive safety functions are shared by the RB and the RB Pressure Relief System (PRS) and the Reactor Building HVAC System.

A preliminary set of Licensing Basis Events (LBEs) was selected to formulate the Reactor Building Technical and functional Requirements. A final set of LBEs shall be defined with input from a full scope PRA during conceptual and preliminary design stage. For this study LBEs are defined by applying engineering judgment based on reviews of the plant design per PCDR and experience with HTGR and specifically PBMR PRAs.

Categories of LBEs include:

- Internal Events
    - HPB leaks and breaks in PHTS and SHTS
    - Transients with and without reactivity addition
- Leaks and breaks in PHTS Heat Exchangers such as those in the Core Conditioning System (CCS) and Intermediate Heat Exchanger (IHX)
    - Internal Plant Hazard Events
    - Internal fires and floods
    - Rotating machinery missiles
- External Plant Hazard Events
    - Hydrogen Process events
    - Seismic Events
    - Aircraft crashes and transportation accidents
    - High winds and wind generated missiles

A preliminary set of LBEs involving leaks and breaks in the PHTS HPB was developed based on a review of similarities and differences of the NGNP design against other PBMR designs for which leak and break initiating event frequencies have been quantified. These LBEs were used to evaluate the Reactor Building capabilities for radiological retention and mitigation of temperature and pressure loads in the formulation of requirements.

Evaluation of Alternative Reactor Building Design Concepts (Part A)

A key result of this study was the evaluation of alternative Reactor Building concepts for the satisfaction of the building safety functions involving pressure relief for HPB leaks and breaks, retention of radioactive material that may be released from the fuel and the HPB, and the limiting of the potential for air ingress following large HPB breaks. A summary of the alternative concepts is provided in Table 1 and includes two alternatives for a vented and unfiltered building, three alternatives for a filtered and vented building, and two alternatives for a leak tight or pressure retaining reactor building.

In order to gain insight into alternative strategies for implementing the safety functions assigned to the NGNP reactor building, a set of alternative design concepts were developed and evaluated. The evaluations were supported by scoping calculations, including an assessment of the capabilities of each alternative to mitigate the releases from a broad spectrum of LBEs involving a Depressurized Loss of Forced Cooling (DLOFC) condition. These scoping calculations should not be confused with the comprehensive safety analyses that will be required to demonstrate that the actual building design is capable of meeting the functional requirements in the next phases of the NGNP design. Such comprehensive analyses require design details not currently available and are outside the scope of this study.

Table 1 Alternative Reactor Building Strategies for Performing Safety Functions

No.	Design Description	Vented Area Leak Rate Vol% /day	Pressure Relief Design Features	Post blow-down re-closure of PRS shaft?	Radionuclide Filtration	
					Blow-down phase	Delayed fuel release phase
1a.	Unfiltered and vented	50-100	Open vent	No	None	Passive
1b	Unfiltered and vented with rupture panels	50-100	Internal + External rupture panels	No	None	Passive
2	Partially filtered and vented with rupture panels	25-50	Internal + External rupture panels	Yes	None	Active HVAC
3a	Filtered and vented with rupture panels	25-50	Internal + External rupture panels	Yes	Passive	Active HVAC
3b	Filtered and vented with rupture panels and expansion volume	25-50	Internal + External rupture panels + expansion volume	Yes	Passive	Active HVAC
4a	Pressure retaining with internal rupture panels	0.1-1	Internal rupture panels	N/A	Passive	Passive
4b	Pressure retaining with internal rupture panels and expansion volume	0.1-1	Internal rupture panels + expansion volume	N/A	Passive	Passive

A comparison of a set of alternatives that were assumed to have the same volume of the Reactor Building vented area, approximately 10,000 m^3 with respect to their capability to retain radio-nuclides released from the PHTS pressure boundary for a range of HPB break sizes from 2 mm to 1,000 mm is shown in Figure 1. As seen in this figure the filtered vented reactor building Alternative 3a provides the most effective radionuclide retention capability among the options considered at this vented area volume including the pressure retaining Alternative 4a. When Alternatives with larger vented area volumes are included, Alternative 3b, which is also a vented filtered configuration, was found to provide superior retention capability among all the alternatives considered. Alternative 4a performs more poorly than 3a or 3b because that option does not remove the pressure driving force that exists following HPB depressurization. Alternatives 3a and 3b on the other hand, despite having much higher leak rate than Alternative 4a or 4b, manage the pressure relief function and eliminate the pressure driving force for the delayed fuel release after the time of depressurization, while using filtration to mitigate the blow-down release and the part of the delayed fuel release that occurs prior to depressurization.

As seen in Figure 1, all of the evaluated alternatives meet the EPA Protective Action Guideline limit and the dose limits for design basis accidents from 10 CFR Part 50.34[8] for design basis breaks up to 100 mm as well as the beyond design basis breaks from 100 mm to 1,000 mm equivalent break size by large margins based on these scoping calculations. The break size of 1,000 mm corresponds to the equivalent single ended break size for a double ended guillotine break of the largest pipe in the PHTS which has an internal diameter of 710 mm for the PBMR NGNP design.

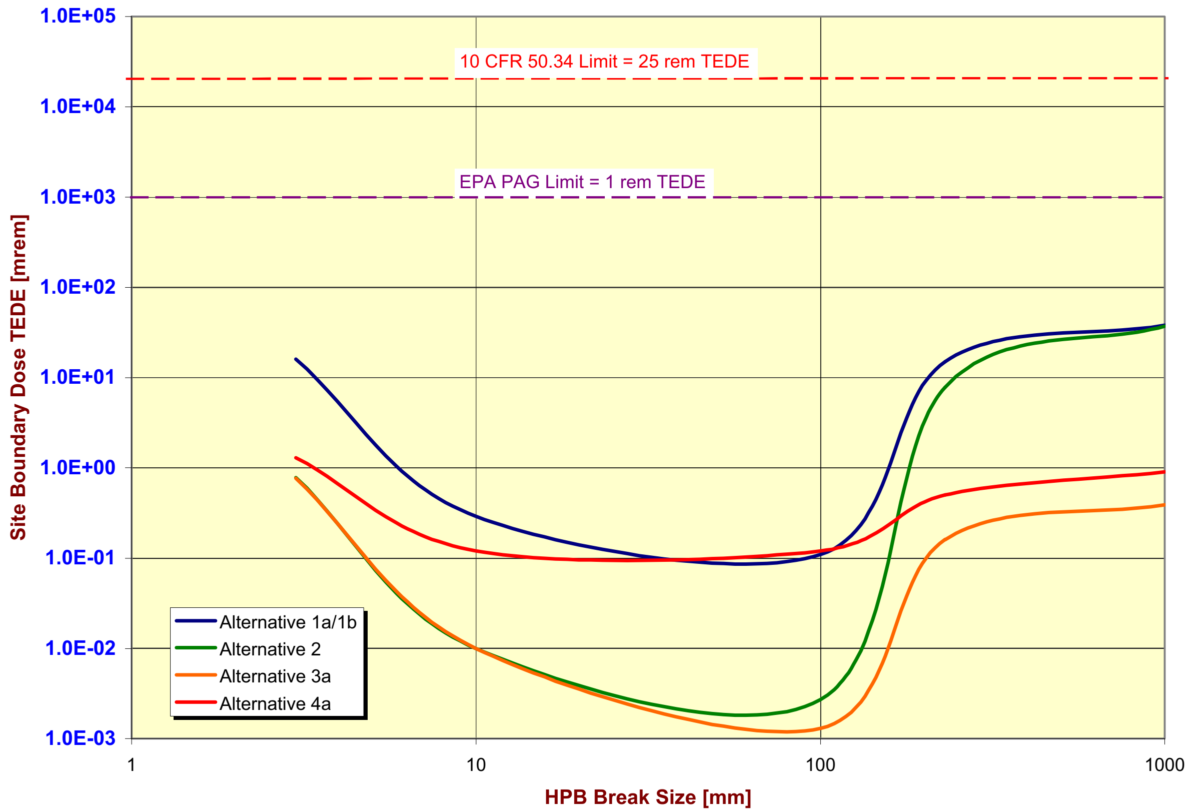

Figure 1 Site Boundary Dose (TEDE) vs. HPB Break Size for LBEs Involving DLOFC for Alternative Reactor Building Concepts with Comparable Volumes

Evaluation Summary of Alternative Reactor Building Concepts (Part A)

In general, the vented options that were considered (1a, 1b, 2, 3a, and 3b) were found to be superior to the pressure retaining options (4a and 4b) based on the following considerations:

- Greater compatibility with a non-condensable and inert primary coolant.
- Venting of the primary coolant inventory to atmosphere with or without filtration eliminates a driving force for subsequent fission product transport of the delayed fuel release source term.
- When used with filtration (2, 3a, and 3b) provides more effective retention of radionuclides for the design basis event spectrum up to 100 mm. Alternatives 3a and 3b provide superior retention for beyond design basis event break sizes up to 1,000 mm as well.
- Lower capital and operating costs.
- Easier and less costly to engineer interfaces with RCCS, SHTS, FHSS, HSS and other NHSS and auxiliary systems.

The highest rating of integrated evaluation for alternatives examined was Alt 2 (Partially filtered and vented with rupture panels) followed by Alt 3a (Fully filter and vented with rupture panels).

- Both alternatives (2 and 3a) provide superior radionuclide retention capability for design basis HPB breaks with DLOFC than the pressure retaining alternatives (4a and 4b); Alt 3b closely followed by 3a is superior to all evaluated alternatives across the entire HPB break spectrum including AOOs, DBEs, and BDBEs.
- Alts 2, 3a, and 3b are expected to have greater licensability than either of the open vented options (1a and 1b) due to their superior capability to mitigate releases and air ingress.
- Another alternative for future study is a vented building with a passive re-closable damper without a filter. This is expected to have delayed fuel release retention capabilities approaching that of Alt 2 due to the capability to achieve a lower leak rate.
- Results of the radionuclide retention study show that all the evaluated alternatives provide sufficient margins to offsite dose limits based on inherent and passive safety characteristics of the PBMR NGNP.
- Added engineered features such as filters and re-closable dampers add additional margins.
- This study confirms that radiological retention is not a required safety function but rather a supportive safety function for the NGNP reactor building according to how these terms are defined in the NGNP risk informed and performance based licensing approach. Having said that it is noted that the required safety functions of the reactor building that

involve the structural protection of the reactor and its inherent and passive safety characteristics also serve to maintain the fundamental safety function of controlling radionuclide releases. It supports this function primarily by keeping the radio-nuclides inside the coated particle fuel and PHTS HPB and secondarily by retaining radio-nuclides that may be released from the fuel and HPB.

- In order to support the PBMR NGNP capabilities for defense-in-depth, it is recommended that a design goal be set for a radiological retention capability of a factor of 10 reduction in releases from the RB relative to that released from the HPB for I-131 and Cs-137 for DBE and BDBE HPB breaks.

Conclusions regarding radiological retention capability of evaluated RB options are subject to limitations due to:

- Lack of design maturity and associated full scope PRA model
- Need to evaluate different HPB break locations and a fuller set of licensing basis events
- Need to consider the impact of natural convection on core temperatures during small leaks (2 to 10 mm)
- Lack of a fully integrated mechanistic source term model
- Need to consider the failure probabilities of various design features as well as RB structural capability to withstand loads from a full set of licensing basis events
- Lack of a full uncertainty analysis in the source term and consequence modeling

Evaluation of Reactor Embedment (Part B)

This portion considers many factors that influence the selection of embedment depth. There are trade offs to consider between features providing protection from hazards, and design and construction elements that contribute significantly to project cost. The detailed discussion of these factors is included in section B1.

The following is a list of factors that are considered in section B1 and basic finding of the study relative to influence on selection of embedment depth:

- Operational Needs including equipment layout

  The PBMR allows for flexibility in selecting embedment depth. Access for operation and maintenance activities can be accomplished at practically any level. There is no need for routine access to the reactor from above as there is with other HTGR designs. The pressure relief system and ventilation systems will discharge to atmosphere at the top of the building. There could be some advantage to having a tall building to elevate this release point. At this time it is not expected that an elevated release will be required to meet off site dose limits. A partially embedded building offers an advantage of reduced travel distances for operations and maintenance activities and associated reduction in operating Life Cycle cost. Therefore, it is concluded the partial embedment is favored slightly in meeting operational needs.

- Heat Dissipation to Environment

  Review of this factor indicates that heat dissipation to the ground or air will not be a design basis mechanism. It is observed, however, that there will be very little difference between heat transfer to soil or to air if this must be considered as last resort. Therefore, it is concluded that this factor has very little, if any, influence on embedment depth selection.

- Water Table Effects

  This factor clearly favors minimum embedment, as dewatering is a cost and potential environmental permitting concern. Hydrostatic pressure causes buoyancy concerns at lower embedment depths. This factor is less important when considering a rock site or soil site with a very deep water table.

- Geotechnical Constraints and Foundation Performance

  Partial embedment is favored to achieve improved resistance to overturning moments. Deeper embedment becomes less favorable due the complexity and cost of excavation and foundation design.

- Construction Considerations

  Construction is more complex and costly at deeper levels of embedment. Safety challenges will also be more rigorous, in a deeper excavation. This factor favors shallower embedment.

- Cost Benefit

  It is shown that costs for excavation, backfilling, dewatering, and external concrete wall construction increase exponentially with embedment depth. This takes into account that robust exterior walls required for hazard protection will be extensive for above ground alternatives, but that below ground level wall and excavation costs will increase dramatically with deep embedment. This concern is more dramatic with a rock site than for a soil site. Therefore this factor favors minimal embedment.

- Malevolent Hazards

  Protection from Design Basis Threats (DBT) and Beyond Design Basis Threats BDBT is most easily achieved by full embedment. This is a major concern. However it has been shown on many projects that protective features can be designed for the these hazards, along with natural hazards.

- Natural Phenomenon Hazards

  It has been shown on many projects that protective features can be designed for these hazards. This is a moderate concern that slightly favors full embedment.

- Natural Geological Hazards

  It is advantageous to be able to reduce seismic accelerations by partially embedding the structure. This reduction is allowed for up to 50 percent embedment. Below 50 percent embedment, further reductions are not allowed. Other geological hazards are discussed but do not influence embedment depth significantly. Therefore, this factor favors partial embedment

- Chemical Releases, Explosions, and Manmade Hazards

  The hazard imposed by toxic and flammable chemicals on or near the site is dependent on the amount stored and the distance from the reactor building and the control room. Future analysis will be required to ensure that distances and quantities are limited to prevent substantial damage to safety related SSCs. It is expected that the structure will be robust for other reasons. There is a concern that heavier than air gasses could enter a below ground structure. It is concluded that this factor is minor and that it favors a minimally partially embedded structure, only slightly.

Embedment Alternatives Considered for Rock Sites and Soil Sites (Part B)

The following alternatives are considered in this study for scoring and ranking purposes. It is expected that the optimal embedment depth will be developed in conceptual design and will be dependent on specific site conditions and ongoing analysis of operations needs.

- Minimal embedment of the building at approximately 7 to 10 meters
- Partial embedment 20 to 30 meters
- Full Embedment 60+ meters

Scoring Summary for Reactor Embedment (Part B)

The Alternatives for embedment depth as indicated above and discussed in section B2 for rock and soil sites were scored as shown in Table 2. Scoring is conducted for a rock site with deep water table similar the INL site. A soil site with high water table is also scored as it is expected that this will be a very common set of conditions for many attractive commercial sites. A soil site with a low water table is expected to exhibit similar characteristics to a rock site and therefore is not specifically evaluated.

Table 2 Evaluation of Embedment Alternatives for Rock and Soil Sites

Design Alternative	Total Weighted Score	Remarks
Rock Site Minimum Embedment	510	Maximum exposure to hazards
Rock Site Partial Embedment	610	Best Score for rock site with balance of protection from hazards and reduced cost
Rock Site Full Embedment	465	Water table is not a concern. Excavation and foundation complexity contributes to construction

Design Alternative	Total Weighted Score	Remarks
		cost
Soil Site Minimum Embedment	580	Maximum exposure to hazards but Benefit with reduced water table concern
Soil Site Partial Embedment	650	Best score for soil site
Soil Site Full Embedment	395	Water table and foundation complexity contribute to very high costs

Recommendations Regarding Embedment (Part B)

The partial embedment scheme scores best for both rock site and soil site, and will therefore be recommended. It offers some protection from hazards without driving costs up excessively as compared to deep embedment. The degree of partial embedment will be optimized for the site during the conceptual design phase.

The PBMR design is quite flexible to accommodate varying site geotechnical conditions. Access to the reactor building can be adjusted for a given site without a significant impact on the layout of major systems within the building.

Role of Reactor Embedment in Satisfying Reactor Building Functions and Requirements (Part B)

The Reactor Building Functions and Requirements portion of this study (Part A) defines reactor building functions and requirements that must be fulfilled independently of the extent of embedment. Part A of this study provides as a preliminary set of licensing basis events involving leaks and breaks in the PHTS and SHTS piping that the building must withstand. Part A also evaluates a number of alternative design strategies for mitigating pipe breaks and minimizing radiological releases from the building. These options can all be applied to any level of reactor embedment. Hence the functions and requirements section is not dependent on the outcome or conclusions of the reactor embedment section.

The reactor embedment section (Part B) is somewhat dependent on the requirements identified in Part A. Also, Part B describes external hazards and physical security requirements that the building design must be able to accommodate. The capabilities of the reactor to protect against these hazards can be significantly influenced by the level of embedment.

Basic functional needs to be met by the reactor building that are dependent on level of embedment are to:

- House and provide civil structural support for the reactor and all the SSCs within the scope of the NHSS (building geometry/space)
- Resist all structural loads as required to support both required and supportive safety functions allocated to the NHSB
- Protect the SSCs within the NHSS that perform safety functions from all internal and external hazards as identified in the LBEs as necessary to meet the TLRC.
- Provide physical security of vital areas within the NHSB against acts of sabotage and terrorism
- Provide radionuclide retention during blow-down and delayed fuel release phases of depressurization events

A complete correlation between identified T&FRs and the consideration of reactor embedment is included in Table 31 in section B2.6. This table shows the full list of T&FRs; the role, if any, that embedment serves to accommodate these requirements; and the rational for selecting a particular embedment alternative. T&FRs are also shown in Table 6 in the functions and requirements section.

Open Issues and Additional R&D and Engineering Studies

Lists of open items to be considered in conceptual design and beyond are included in Sections A5 and B3. There are specific R&D items identified. Pending a more thorough design iteration, DDNs will be formulated leading to potential technology development, primarily in the fuel, reactor, and HPB.

INTRODUCTION

This study supports the development of the technical and functional requirements (T&FRs) for the NGNP reactor building, including consideration of requirements for embedment. The approach to the definition of these requirements includes consideration of:

- NGNP top level requirements
- Role of Reactor Building in the PBMR NGNP safety design approach
- NRC regulations and guidance relevant to the Reactor Building
- NRC regulations on design basis threats and hazards
- Definition of Reactor Building safety functions including those that are required to meet requirements for mitigating design basis accidents as well as those that support the prevention and mitigation of a wide spectrum of licensing basis events

This study shall be based on the current understanding of expected source terms for the NGNP licensing basis events, including a comprehensive set of fission product transport phenomena from the fuel into the PHTS circuit, from the PHTS circuit into the Reactor Building, and from the Reactor Building to the environment. As such, it is understood that this will be a scoping study helping to frame the issues associated with development of the T&FRs for the Reactor Building. Accordingly, an objective of this study is to identify the issues and further R&D and engineering studies that are required to resolve these issues.

This study also develops the requirements and criteria for determining the degree of embedment of the reactor building. This includes embedment studies for the NGNP reactor building concepts, considering the interaction among factors that influence the depth of the embedment. These factors include cost, design basis threats, seismic effects, hazards resistance, etc. The results of this study will be used to characterize the interactions of these factors on embedment depths for commercial application of this technology.

OBJECTIVE AND SCOPE

The objective of this study is to develop the technical and functional requirements (T&FRs) for the NGNP and HTGR reactor building, including consideration of reactor embedment. The scope of the study includes Part A, in which the overall technical and functional requirements for the reactor building are defined and various design strategies to meeting these requirements are evaluated. In Part B an evaluation is made of strategies involving reactor embedment.

This study is organized into two parts based on the assumption that the type of reactor building can be separated to a certain extent from the degree of reactor embedment. However, these parts are organized into a single report in view of the expected synergy between the tasks of identifying a comprehensive set of reactor building requirements and the evaluation of reactor building design options, and the reactor embedment options against these requirements.

The development of T&FRs for the reactor building will be based on a review of:

- NGNP user requirements associated with safety, availability, investment protection, plant costs, and the commercialization business case.
- NRC regulations and HTGR objectives regarding the frequency and radiological consequences of licensing basis events, SSC safety classification, and defense-in-depth.
- NRC policy issues (SECYs) and staff requirements memoranda (SRM) on the expected safety characteristics of advanced reactors and the development of containment performance criteria for HTGRs.
- NRC regulations on design basis and beyond design basis threats and hazards to be addressed for existing and advanced reactors.
- Requirements for retention of radio-nuclides within each barrier including the fuel, primary helium pressure boundary, and the reactor building, and, if needed, definition of reactor building release rate and filtration requirements.

This study is based on the current understanding of expected categories of Licensing Basis Events (LBEs) as well as an expected range of frequencies and mechanistic source terms for these LBEs that are appropriate for the PBMR. As such, it is understood that this is a scoping study helping to frame the issues associated with development of the T&FRs for the reactor building. Accordingly, an objective of this study is to identify the issues and further R&D and engineering studies that are required to resolve these issues. This study investigates alternative design strategies to meet the reactor building technical and functional requirements including the use of various design approaches to address the required and supportive safety functions of the reactor building. The required safety functions are focused on the preservation of the reactor core geometry and structural protection of the reactor and safety classified SSCs from a range of internal and external hazards and challenges represented in the LBEs. Retention of radio-nuclides that are released from the fuel and the PHTS HPB is a supportive safety function for the reactor building because the primary safety function of retaining radio-nuclides is assigned to the fuel in the NGNP safety design approach

This study includes embedment studies for the PBMR RB concepts, considering the interaction among factors that influence the depth of the embedment. These factors include cost, design basis threats, seismic effects, hazards resistance, etc. The results of this study are used to characterize the interactions of these factors on embedment depths for commercial application of this technology. The recommendations from relevant sections of the Electric Power Research Institute (EPRI)'s *Advanced Light Water Reactor Utility Requirements Document* [24] are evaluated for applicability in this study. This phase of the study also includes a review of prior NRC reviews of modular HTGR designs and the conclusions from those reviews concerning the embedment feature of the technology on licensing considerations.

ORGANIZATION OF REPORT

This study is organized into two parts based on the assumption that the type of reactor building can be separated to a certain extent from the degree of reactor embedment. However these parts are organized into a single report in view of the synergy between the tasks of identifying a comprehensive set of reactor building requirements and the evaluation of reactor embedment options against these requirements.

Part A of this report covers the development of T&FRs for the reactor building and the identification and evaluation of various design strategies for meeting these requirements. These strategies include those to perform required safety functions such as pressure relief from HPB depressurization events as well as those to perform the supportive safety function of retention of radioactive material releases from the HPB and limiting of air ingress. All the strategies evaluated in Part A assume a nominal extent of reactor embedment consistent with the design of the PBMR Demonstration Power Plant. Part B addresses that part of the scope of work dealing with reactor embedment, whose range of options vary with the site, from fully above to fully below grade.

PART A: REACTOR BUILDING TECHNICAL AND FUNCTIONAL REQUIREMENTS AND EVALUATION OF ALTERNATIVE CONFINEMENT STRATEGIES

The purpose of Part A is to define the technical and functional requirements for the reactor building and to evaluate different design concepts toward fulfilling these requirements independent of embedment, which is evaluated in Part B. In Section A1 the derivation of the technical and functional requirements for the Reactor Building is documented. An evaluation of alternative design concepts for implementing these requirements is discussed in Section A2. In Section A3 the results, conclusions, and recommendations are summarized.

A1 DEFINITION OF TECHNICAL AND FUNCTIONAL REQUIREMENTS

A1.1 TECHNICAL APPROACH TO DEFINITION OF REQUIREMENTS

The technical and functional requirements (T&FR) for the NGNP Reactor Building are comprised of a list of reactor building functions and a specification of the technical requirements for satisfying these functions. The technical approach to defining these functions is outlined in Figure 2 and consists of the following steps implemented in a top-down fashion:

- Review the NGNP Top Level Requirements in the PCDR (Reference [1])
- Review NRC requirements applicable to the Reactor Building
- Identify the role of the Reactor Building in the PBMR NGNP safety design approach (References [2] through [6])
- Consideration of the approach to defining safety functions in the risk-informed and performance based approach envisioned for the PBMR NGNP
- Development of a preliminary list of licensing basis events for the Reactor Building
- Definition of required and supportive safety functions for the Reactor Building and associated technical requirements
- Definition of physical security requirements for malevolent human actions
- Definition of non-safety related requirements for the Reactor Building
- Compilation of a complete list of Reactor Building technical and functional requirements

The results of each step in this process are presented in the sections below.

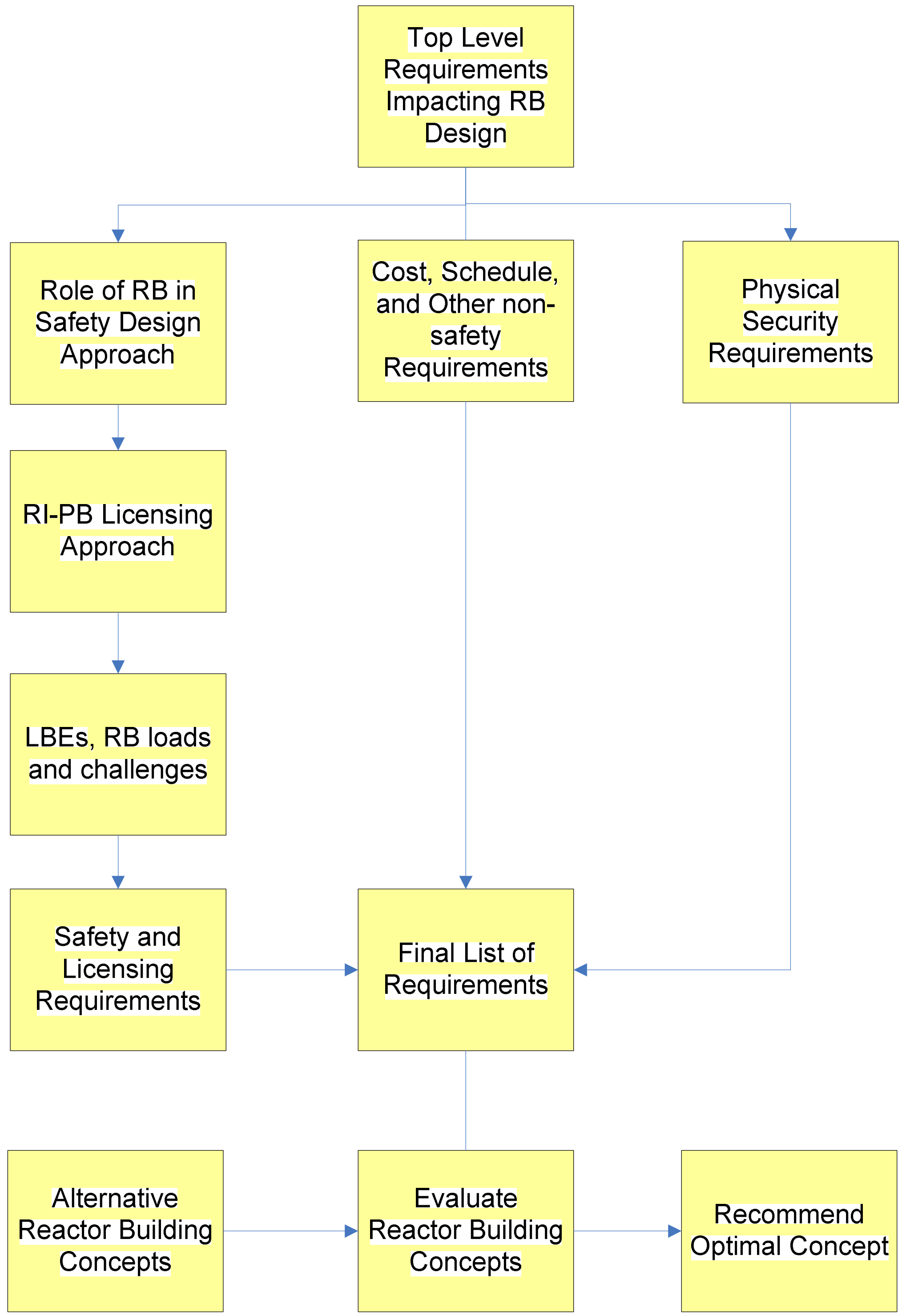

Figure 2 Flow Chart of Part A: Development of Reactor Building Requirements and Evaluation of Alternative Concepts

A1.2 NGNP TOP LEVEL REQUIREMENTS

The NGNP top level requirements for the reactor building and the rest of the plant flow from the Top Level Requirements set forth in of the NGNP *Preconceptual Design Report* (PCDR), Section 2 [1]. These Top Level Requirements flow down to requirements impacting the reactor building and include:

- Plant performance requirements
- Plant availability and investment protection requirements
- Plant cost targets that are in alignment with the NGNP commercialization strategy
- Plant safety and associated licensing requirements

The NGNP safety and associated licensing requirements include the overall NGNP requirements as well as specific requirements for the Nuclear Heat Supply System (NHSS) and other NGNP facilities. Five safety and licensing design basis requirements are listed for the NHSS:

- The NHSS shall be designed in accordance with the generic top-level, nuclear regulatory criteria that are direct and quantitative measures of nuclear-related risks and consequences, plus all applicable governmental codes and regulations.
- The NHSS shall meet the top-level regulatory criteria without credit for sheltering or evacuating the public beyond the plant's exclusion area boundary, with the intended result that the Emergency Planning Zone (EPZ) is limited to the Exclusion Area Boundary (EAB).
- The NHSS shall meet the top level regulatory criteria without reliance on prompt operator action, the control room or its contents, including the automated plant process control system, or on auxiliary power supplies (other than batteries).
- Design requirements supporting safety and licensing objectives, that are incremental to those employed in conventional power plant design, construction, operation and maintenance and quality assurance practices, shall be developed using the principles of risk-informed, performance-based regulation.
- The NHSS shall be equipped with provisions to safely shutdown the reactor in the unlikely event the central control room becomes uninhabitable.

Three additional safety and licensing requirements are listed for the Hydrogen Production Facility (HPF):

- The hydrogen production facilities, including the conversion, storage, and distribution systems, shall comply with the requirements of 29 CFR 1910.103, Occupational Safety and Health Standards, Subpart H - Hazardous Materials, Hydrogen [51].

- In the event that the HPU facility also produces and stores significant quantities of oxygen, the requirements of 29 CFR 1910.104, Oxygen [52], shall be applied.
- The design, operation and maintenance of the HPU shall comply with 29CFR 1910.119, "Process Safety Management of Highly Hazardous Chemicals" [53].

The above listed requirements shall be used as a starting point for formulating reactor building-specific technical and functional requirements.

A1.3 REVIEW OF APPLICABLE NRC REQUIREMENTS

Existing regulatory requirements for reactor buildings and containment of radioactive material are specific to LWRs and reflect the safety design approach of LWRs. That approach involves the use of active engineered safety systems to prevent core damage over a spectrum of design basis events with a focus on loss of coolant accidents and a pressure retaining containment structure to mitigate the consequences of design basis and to contain radioactive material that could be released in beyond design basis core damage accidents. The PBMR NGNP is based on a fundamentally different safety design approach that identifies a different set of safety requirements for the barriers to radionuclide release, including the reactor building, as discussed more fully in the next section. As a result of these differences, the existing requirements for reactor buildings and containment of radioactive material need to be interpreted with great care. The regulatory requirements for the NGNP Reactor Building will need to be resolved in the course of implementing the RI-PB approach.

Table 3 Elements of NGNP Risk-Informed Performance-Based Licensing Approach

Elements of Approach	Purpose
• Top Level Regulatory Criteria (TLRC)	• Establish what level of safety must be achieved in terms of the frequencies and radiological doses of event sequences
• Licensing Basis Events (LBEs)	• Define when and under what conditions the TLRC must be met based on event sequences selected from the PRA
• Required Reactor Specific Safety Functions • Regulatory Design Criteria (RDC) • Safety Classification of SSCs • Plant Capability Defense-in-Depth	• Establish how it will be assured that the TLRC are met

Elements of Approach	Purpose
• Design Basis Accident (DBA) Conditions • Special Treatment Requirements • Programmatic Defense-in-Depth	• Provide assurance as to how well the TLRC are met by satisfying deterministic requirements to reduce uncertainties in the probabilistic results

The NGNP PCDR specifies a risk informed and performance based (RI-PB) licensing approach to derive regulatory requirements for the NGNP. The key elements of the RI-PB licensing framework are summarized in Table 3.

NRC Research has developed a similar RI-PB licensing approach for advanced non-LWRs referred to as the Technology Neutral Framework as described in NUREG-1860 [7]. A review of this document provides an indication of the NRC research staff current thinking on how to develop a set of regulatory requirements for an advanced non-LWR. The NUREG-1860 [7] method is often referred to as the NRC Technology Neutral Framework (TNF). A comparison of the NGNP RI-PB approach as described in Section 14 of the PCDR and the TNF indicates a number of comparable features including the following.

Both approaches:

- Use frequency vs. dose criteria derived from regulations to judge the acceptability of licensing basis events (LBEs)
- Derive LBEs from a full scope all modes PRA
- Assign LBEs to three categories based on frequency of the underlying event sequences
- Include conservative deterministic analysis to augment the PRA
- Derive special treatment requirements for SSCs based on the LBE results and frequency-dose criteria
- Apply defense-in-depth criteria to address uncertainties
- Include a need to screen the existing requirements for applicability and to develop new technology specific requirements

Some of the key differences between the respective approaches are listed in Table 4.

Appendix H of NUREG-1860 [7] provides NRC staff screening of regulations for applicability to new non-LWRs. The process that was followed and the results of the screening appear to be similar to screening of regulations performed by Exelon during earlier pre-licensing interactions associated with the PBMR. The scope of this screening includes all of 10 CFR and each of the GDCs in Appendix A to 10 CFR Part 50 [54]. Most GDCs are either retained as currently written or revised to make the criteria reactor technology neutral (54 GDCs fall into this category). Some GDCs were assessed by the staff to be specific to LWRs and not applicable to non-LWRs (GDC 33, 35-40, 44-46). It is noteworthy that GDC-3 on fire protection is revised in this research report revised to include graphite fires.

The following conclusions and insights for the NGNP Reactor Building were developed as a result of this review of NRC regulatory requirements:

- The current NRC requirements for reactor buildings and containment of radioactive material are specific to LWRs and need to be interpreted for applicability to the PBMR NGNP.
- Formulation of NGNP Reactor Building requirements requires some assumptions about the outcomes of applying and NRC acceptance of the NGNP RI-PB approach
- Similarities with NUREG-1860 [7] approach encouraging as to the acceptability of the NGNP RI-PB approach
- NRC staff appears to expect a radiological confinement function for the NHSB and has concerns about graphite fires which will need to be addressed.
- In principle, it appears the staff is prepared to accept a non-pressure retaining vented confinement with some controls on leak rate and significant radiological retention capability within the NHSB and associated systems.
- A key technical issue to be resolve is getting agreement with the NRC on the selection of licensing basis events for evaluating the capability of the Reactor Building to satisfy its safety functions and associated licensing requirements.
- Key issues will be acceptance of NGNP PRA, mechanistic source terms for LBEs, and reasonable resolution of the so-called “deterministic” design basis event for evaluation of the third barrier and siting.

Table 4 Key Differences between NGNP and NUREG-1860 RI-PB Approaches

Characteristic	NGNP RI-PB Approach	NUREG-1860 RI-PB Approach
Frequency metric	Events per plant-year	Events per reactor-year
LBE Categories	AOO; = 10^{-2}/plant-year	Frequent: 10^{-2}/Rx-year
	DBE; 10^{-4} to 10^{-2}/plant-year	Infrequent: 10^{-5} to 10^{-2}/Rx-year
	BDBE: $5x10^{-7}$ to 10^{-4}/plant-year	Rare: 10^{-7} to 10^{-5}/Rx-year
SSC Safety Categories	Safety related (SR)	Safety significant (has special treatment)
	Non-safety related with special Treatment (NSRST)	Non-Safety Significant (no special treatment
	Non-safety related without special treatment	

Characteristic	NGNP RI-PB Approach	NUREG-1860 RI-PB Approach
Deterministic Safety Analysis	Deterministic DBAs must meet design basis dose limits with mitigation credit for only SR SSCs and conservative assumptions	Frequency vs. dose criteria must be met with mitigation credit for only safety significant SSCs Specific additional deterministic requirements for frequent and infrequent events; Additional deterministically selected LBE

A1.4 DEFINITION OF SAFETY AND LICENSING REQUIREMENTS

A1.4.1 Role of Reactor Building in NGNP Safety Design Approach

In order to provide a context for defining the safety and licensing issues that influence the reactor building design it is necessary to review the NGNP safety design approach and the role of the reactor building in prevention and mitigation of licensing basis events.

The risk-informed and performance-based safety design approach is derived from that developed by PBMR (Pty) Ltd. in support of the design certification for future U.S.-sited plants. This PBMR approach builds upon the successful application of risk-informed methods undertaken as backfits for the current fleet of licensed, operating reactors and is an extension to the methods used in other design certification applications. (References [3], [4], [5] and [6])

The safety design philosophy is to apply the principles of defense-in-depth at a fundamental level in which a diverse combination of inherent reactor characteristics, passive design features and Structures, Systems and Components (SSCs), active engineered systems, and operator actions are deployed to maintain the integrity of robust passive barriers to radionuclide release. These barriers include the fuel barrier, the primary helium pressure boundary, and the reactor building. The plant capabilities that support defense-in-depth for any reactor technology are illustrated conceptually in Figure 3. For any reactor, these capabilities are based on a foundation of inherent reactor characteristics, a set of barriers to the release of the reactor's inventory radioactive material, and a set of passive and active SSCs that perform safety functions that can be tied to the preservation of the integrity of one or more barriers. For different reactor concepts, the inherent characteristics are different, and reactor safety design approach for use of combinations of passive and active design features is also necessarily different.

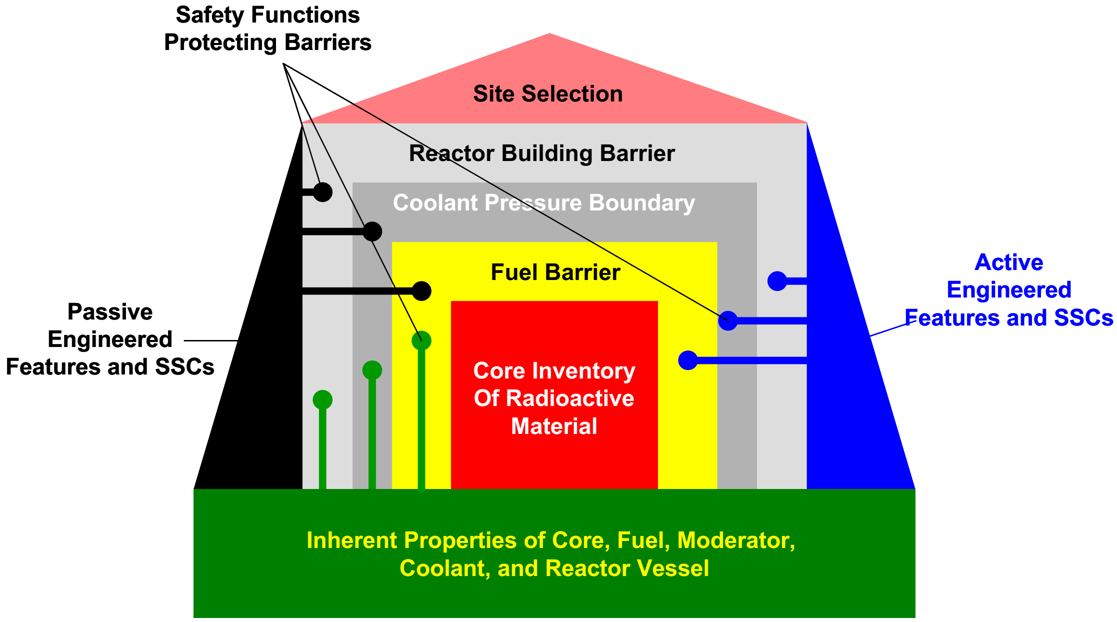

Figure 3 Key Elements of Plant Capabilities Supporting Defense-in-Depth

All reactors must fulfill a set of fundamental safety functions that are derived from a need to prevent the release of radioactive material and to mitigate the extent of releases during accidents. The specific set of safety functions and the allocation of passive and active SSCs to perform these safety functions are necessarily specific to each reactor and dependent on its safety design philosophy. The safety design approach allocates responsibilities to each barrier for the prevention and mitigation of accidents which are defined in terms of the events and event sequences that challenge the safety case.

The NGNP-specific key safety functions are derived in a top-down manner with the objective of protecting the integrity of the multiple barriers to radionuclide release. A fundamental aspect of the safety design philosophy is to provide the capability to perform safety functions first through the selection of inherent reactor characteristics and engineered systems that operate on passive design principles and then to support these safety functions with combinations of diverse active engineered systems and operator actions.

PBMR safety philosophy starts with the application of defense-in-depth principles in making fundamental design selections:

- Inherent reactor characteristics
- Multiple, robust, and concentric barriers to radionuclide release
- Conservative design approaches to support stable plant operation with large margins to safety limits

- In the event safety functions are challenged, the safety philosophy is to utilize the inherent reactor characteristics and passive design features to fulfill the required safety functions
- Additional active engineered systems and operator actions are provided to reduce the challenge to plant safety and provide defense-in-depth in preventing and mitigating accidents
- Avoid need for early operator intervention, or early functioning of any active systems to maintain safe stable state

The NGNP safety design approach is framed in terms of reactor-specific safety functions that were developed from the top goal of retaining the inventory of radio-nuclides primarily within the fuel and then considering the specific functions that, when satisfied, would protect the integrity of the fuel and other radionuclide transport barriers. These safety functions, which are illustrated in Figure 4, are comprised of two general categories referred to as required and supportive safety functions. These categories are defined as follows:

- **Required Safety Functions**—Those functions that are necessary and sufficient to meet the dose limits for Design Basis Events and deterministically selected Design Basis Accidents.
- **Supportive Safety Functions—**All other functions that contribute to the prevention or mitigation of accidents and support the plant capabilities for defense-in-depth.

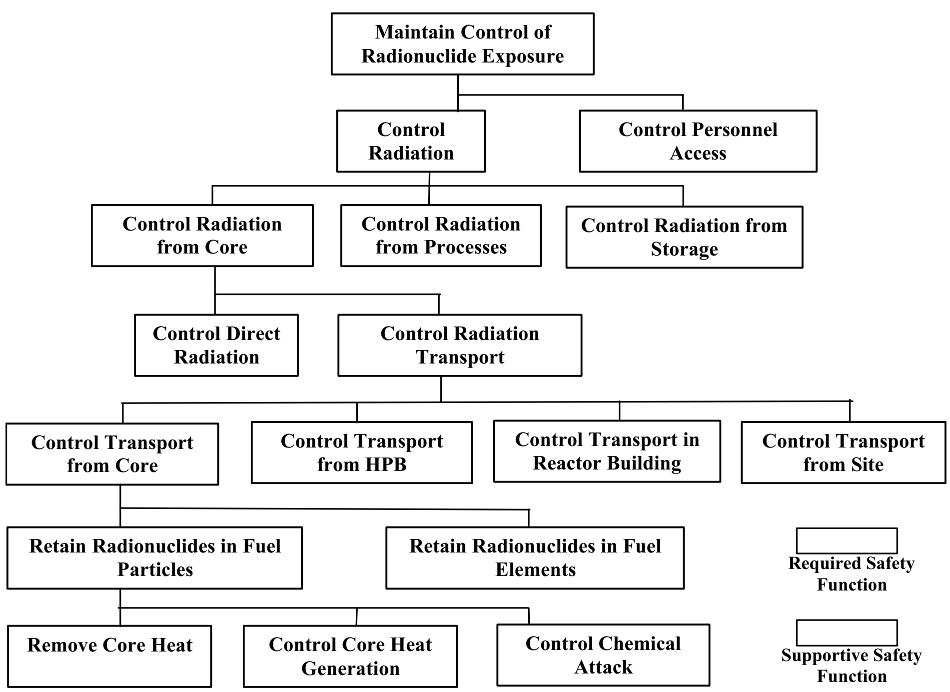

Figure 4 Top Down Development of NGNP Safety Functions

In the future NGNP licensing submittal, analyses will be included to demonstrate which safety functions are classified as required and which as supportive according to the NGNP RI-PB as discussed more fully in Section 14 of the PCDR (Reference [2]). The required safety functions for the NGNP, all of which contribute to the fundamental safety function of preventing and mitigating the release of radioactive material, include those functions to:

- Control heat generation
- Control heat removal
- Control chemical attack
- Maintain core and reactor vessel geometry
- Maintain reactor building structural integrity

In formulating requirements for the Reactor Building, the following types of functions need to be considered:

- **Safety Functions**
    - Required Safety Functions assigned to the reactor building
    - Supportive Safety Functions assigned to the reactor building
- **Physical Security Functions**—Those functions assigned to the building protect the reactor and vital equipment from design basis threats associated with acts of sabotage and terrorism
- **Other Functions**—Functions necessary for plant construction, operation, maintenance, access, inspection, worker protection, and control of routine releases of radioactive material during normal operation

The scope of requirements and technical functions for the reactor building are allocated in the PCDR to the Nuclear Heat Supply Building (NHSB), NHSS HVAC System, and NHSB Pressure Relief System (PRS). Based on the NGNP safety design philosophy and the NGNP RI-PB licensing approach, the following required and supportive safety functions can be defined:

The NHSB shall perform the following required functions:

- House and provide civil structural support for the reactor and all the SSCs within the scope of the NHSS
- Resist all structural loads as defined for NGNP required safety functions allocated to the NHSS
- Protect the SSCs within the NHSS that perform required safety functions from all internal and external hazards as identified in the LBEs

- Provide physical security of vital areas of the NHSS against acts of sabotage and terrorism

The RB shall perform the following supportive safety functions:

- Provide retention of radio-nuclides released from the PHTS HPB
- Limiting the potential for air ingress into the PHTS following HPB leaks and breaks as defined in the licensing basis events

Note that the final list of SSCs that are classified as safety related and therefore must be protected by the building from DBE loads is to be determined in future stages of design and is outside the scope of this study. In reviewing the above allocation of safety functions to required and supportive categories, it must be kept in mind that all the NGNP safety functions support the fundamental safety function of preventing and mitigating the release of radioactive material. According to the NGNP safety design approach, the allocation of safety functions to the reactor building is of a structural nature. If the NGNP Reactor Building fulfills its required safety function of structurally protecting the core geometry and the structural integrity of the PHTS pressure boundary, it will also be supporting the required safety functions of controlling core heat removal, preventing chemical attack and controlling heat generation. These in turn serve to keep the radionuclide inventory inside the coated fuel particles and when radioactive material is released from the fuel particles during normal operation or during an accident, helps to keep the releases from the PHTS HPB and Reactor Building within their required limits. Having met those required safety functions, the functions of retaining radionuclide that may be released from the PHTS and those of limiting the potential for air ingress are classified as supportive because they are not required or credited in the performance of the deterministic design basis accidents. SSCs that perform required as well as supportive requirements are subjected to special treatment requirements to the extent necessary to keep the frequencies and consequences of the licensing basis events within the TLRC, according to the NGNP RI-PB licensing approach as discussed more fully in Section 14 of the PCDR (Reference [2]).

It is noted that the Reactor Cavity Cooling System (RCCS) also provides important functions that contribute to the structural integrity to the reactor vessel and the maintenance of core geometry. Those functions are primarily supported during normal plant operation prior to the initiating event by keeping the vessel support temperatures and reactor cavity concrete temperatures within code limits. In the event that continued operation of the RCCS following any design basis event is deemed to be required to maintain structural integrity, a required safety function of the reactor building will be to protect the RCCS during all the design basis events. However, further study will be needed to establish whether there is a need for such a requirement.

A1.4.2 Implications of RI-PB Licensing Approach

In order to obtain a license to operate the NGNP, the reactor building design and the remaining safety features of the plant will ultimately be subjected to a rigorous deterministic and probabilistic safety analysis guided by a risk-informed and performance-based licensing approach that is described in References [3] through [6]. The safety evaluation for the NGNP will help to define and finalize reactor building requirements and to demonstrate the building design is capable of meeting these requirements. The key elements of this RI-PB approach include:

1. The use of accident frequency vs. radiological dose criteria that are derived from current U.S. licensing requirements, referred to as Top Level Regulatory Criteria (TLRC). These regulatory criteria are expressed in terms of acceptable combinations of event sequence frequencies and radiological doses to the public at or beyond the site boundary as well as the aggregate public health and safety risk due to accidents for demonstrating conformance to the NRC Quantitative Safety Objectives and associated safety goals.
2. Use of a full-scope Probabilistic Risk Assessment (PRA) to select the Licensing Basis Events (LBEs). Full scope means that the PRA will cover all operating and shutdown modes, all internal and external plant hazards, mechanistic source terms and offsite radiological consequences. These LBEs include: Anticipated Operational Occurrences (AOOs) whose frequencies are above 10^{-2} per plant year and are evaluated using realistic assumptions; Design Basis Events (DBEs); whose frequencies are between 10^{-4} and 10^{-2}, per plant year and are evaluated using realistic and conservative assumptions, the probabilistic and deterministic safety analysis, respectively, and Beyond Design Basis Events (BDBEs) with frequencies less than 10^{-4} and whose consequences are evaluated using realistic assumptions. These rules for conservative and realistic assumptions will impact the definition of reactor building performance requirements for different LBEs.
3. Development of reactor-specific functions, selection of the corresponding safety-related SSCs, and their regulatory design criteria. The SSC of interest in this evaluation is the reactor building.
4. Deterministic design conditions and special treatment requirements for the safety-related SSCs including certain aspects of the reactor building design.
5. A risk-informed evaluation of defense-in-depth.

In this evaluation, a qualitative review of this RI-PB approach was be used to help formulate the preliminary set of reactor building requirements. Insights from existing and previous PBMR and HTGR PRAs and a review of the NGNP design features are used in the next section to identify a set of LBEs for use in this evaluation, in lieu of actually performing a PRA. However, the performance of a full scope PRA and the execution of each of the above steps in the safety evaluation approach will need to be performed to finalize the Reactor Building requirements and to demonstrate the adequacy of the safety case.

A1.4.3 Preliminary Licensing Basis Events

The Licensing Basis Events (LBEs) for the NGNP will be defined via full scope PRA during conceptual and preliminary design. For the purposes of this study, LBEs are defined by engineering judgment based on reviews of the plant design as describe in the PCDR and experience by the authors in the performance of HTGR and specifically PBMR PRAs.

Categories of LBEs include:

- Internal Events
    - HPB leaks and breaks in PHTS and SHTS HPB
    - PHTS HPB Heat exchanger failures (e.g. CCS Hx, IHX)
    - Transients with and without reactivity addition
- Internal Plant Hazard Events
    - Internal fires and floods
    - Rotating machinery missiles
    - High energy line and tank breaks
- External Plant Hazard Events
    - Hydrogen Process events
    - Seismic Events
    - Aircraft crashes and transportation accidents
    - High winds and wind generated missiles

For each of the above categories the LBEs may include events in either of the three LBE categories (i.e., AOOs, DBEs, or BDBEs). In the formulation of Reactor Building requirements, a key consideration is the definition of LBEs associated with leaks and breaks on the PHTS and SHTS helium pressure boundary (HPB). This is key consideration because leaks and breaks result in depressurization of large inventories of pressurized Helium and relatively high temperature and resulting pressure and temperature loads on the building and the NHSS SSCs contained within it. Importantly, leaks and breaks in the PHTS HPB also involve a breach in one of the three radionuclide transport barriers, which means that radioactive material released from the fuel, either during normal operation or during an accident, has the potential for release into the building – and if not confined within the building, to the environment. Based on a qualitative review of the NGNP design and a comparison with other designs for which HPB leak and break initiating event frequencies have been quantified, the following categories of HPB leak and Break events were defined for the reactor building. The leaks and breaks in the PHTS are also summarized in Table 5. Aircraft crashes may include those within and outside the design basis depending on the frequency of occurrence.

Table 5 Preliminary LBEs Involving HPB Leaks and Breaks

<table>
<tr><th>LBE Type</th><th>Frequency Range (per plant year)</th><th>Break Category</th><th>Equivalent Break Size (mm)</th></tr>
<tr><td>Anticipated Operational Occurrences</td><td>= 10^{-2}</td><td>Small PHTS HPB break with forced cooling</td><td rowspan="2">1 to 10</td></tr>
<tr><td rowspan="3">Design Basis Events</td><td rowspan="3">10^{-4} to 10^{-2}</td><td>Small PHTS HPB break with loss of forced cooling</td></tr>
<tr><td>Medium PHTS HPB break with loss of forced cooling</td><td>> 10 to 100</td></tr>
<tr><td>Large SHTS break with intact PHTS HPB</td><td rowspan="2">> 100 to 1,000*</td></tr>
<tr><td>Beyond Design Basis Events</td><td>$<10^{-4}$</td><td>Large PHTS HPB break with loss of forced cooling</td></tr>
</table>

* 1,000 mm is the equivalent single ended break size for a double ended guillotine break of the largest pipe on the PHTS pressure boundary with inside diameter of 710 mm.

- Anticipated Operational Occurrences (AOOs) assumed to include leaks up to 10 mm in break size anywhere along the PHTS and SHTS HPB inside the RB–PRS and HVAC designed to keep HVAC running during such leaks
- Design Basis Events (DBEs) are expected include:
  - Up to 50 mm breaks in the FHSS pipes above or below the RPV
  - Up to 100 mm breaks in the CIP, COP or other PHTS piping above or below the core including pipe to vessel nozzle welds
  - Breaks in HSS piping bounded by 100 mm break size
  - Up to 1 meter (SEGB) breaks in the SHTS piping inside the NHSB
  - Requirement for DBEs is to maintain structural integrity and avoid damage to any safety related SSC inside the NHSB
- Beyond Design Basis Events (BDBEs) are expected to include
  - Up to 1 meter (SEGB) breaks in the PHTS
  - Requirement is to meet TLRC using realistic assumptions and to show that structural integrity of the RPV and major PHTS SSCs is maintained

In the above for BDBEs the term realistic is used to describe best estimate evaluations that are technically justified and supported by an evaluation of the underlying uncertainties.

A1.5 DEFINITION OF PHYSICAL SECURITY REQUIREMENTS

The design of the reactor building should include consideration of requirements imposed by the need for physical security protection of the plant from acts of terrorism and sabotage. The basis for these requirements and for satisfying them comes from Rules and guidance of the NRC and industry standard responses to these Rules and guidance. The overall goal of physical security is to protect against radiological sabotage. This goal has been further defined as having a physical design and trained response force available to deny access of the "Design Basis Threat (DBT)" from target sets of "Vital Equipment."

The Design Basis Threat (DBT) is imposed by Title 10 of the Code of Federal Regulations. It is defined by NRC and classified as Safeguards Information (SGI). The DBT defines sets of attackers with definitions of their capabilities, skills and equipment. It defines the number and type of personnel within the DBT, both from inside and outside the facility. It defines the transportation equipment, weapons, explosives and other equipment that are part of the DBT. It defines allowable deployment of the DBT for evaluation.

Security for the reactor building is governed by Parts 50, 52 and 73 of Title 10 of the Code of Federal Regulations [54],[55] and [56]. Security, as defined by 10CFR73 [56] is the protection from radiological sabotage or protection from theft and diversion of special nuclear materials. It must be provided throughout the life cycle of nuclear power plant fuel. Additional, more detailed requirements are in 10CFR73 [56]:

- *Vital area* means any area which contains vital equipment.
- *Vital equipment* means any equipment, system, device, or material, the failure, destruction, or release of which could directly or indirectly endanger the public health or safety by exposure to radiation. Equipment or systems which would be required to function to protect public health and safety following such failure, destruction or release are also considered vital.
- [V]ital equipment [shall be located] only within a vital area, which in turn, shall be located within a protected area such that access to vital equipment requires passage through at least two physical barriers of sufficient strength to meet the performance requirements of . . . this section. More than one vital area may be located within a single protected area.
- The physical barriers at the perimeter of the protected area shall be separated from any other barrier designated as a physical barrier for a vital area or material access area within the protected area.
- Unescorted access to vital areas . . . shall be limited to individuals who are authorized access to the material and equipment in such areas and who require such access to perform their duties

The designer must define a set of Vital Equipment. Vital Equipment and other spaces (the Control Room, Central Alarm Station and Secondary Alarm Station) must be included in Vital Areas. There must be two barriers between the general public and entry into a Vital Area. Usually one of these barriers is the site Protected Area Boundary, which includes a Vehicle Barrier System. For the Reactor Building, this means that it must support controlled access into Vital Areas and DBT denial from Vital Equipment.

The final design deliverable for physical design basis security is a Physical Security Plan (PSP) that is integrated with the physical design of the facility. The PSP is submitted by the plant owner/operator/licensee to the NRC. It is SGI and is a completion of the Security Plan template developed in cooperation between the nuclear industry (NEI) and the NRC. This template is NEI 03-12 [25] and is the basis for all Physical Security Plans used in operating plants and those currently obtaining licenses. It is assumed that the reactor building, its fuel handing facility and the hydrogen generation facility are collocated, but are such that they can have unique access controls.

The current working definition of “directly or indirectly endanger the public health or safety by exposure to radiation” includes releases from the site in excess of 10CFR100 [57] allowables, radiological sabotage and theft or diversion of formula quantities of special nuclear materials. Documented evidence of adequate protection is required during the Combined License application process. It must include description of design features to deny or delay unauthorized access of the Design Basis Threat (DBT) and the resistance to and mitigation of damage caused by beyond DBT. The configuration of perimeter fencing and intrusion detection will be developed by the combined License applicant. Denial and delay features of the reactor building must be included at the Design Certification phase of licensing. Release or theft of fuel spheres, especially irradiated spheres, could directly or indirectly endanger the public health or safety by exposure to radiation. This release of spheres could also lead to radiological sabotage.

The plant design and defensive strategy will be reviewed against the DBT. This review will include threat definitions, attack scenarios, target set definitions, defensive positions, delay features and access paths. Target sets are those sets of systems, structures or equipment that if compromised or used improperly will lead to radiological sabotage or theft or diversion of special nuclear materials. This means that target sets will be identified within the reactor building. Fault tree analysis and probabilistic review of potential target sets is conducted to determine if they need to be considered. The acceptance criterion is that no member of the DBT disables any target set to the point of release or sabotage.

The review is conducted by security professionals with a table top exercise. This effort is a virtual exercise in which the DBT is deployed against the plant security force and design features. A variety of assaults are conceived to ensure that there are no weak spots in the defensive strategy. This security assessment will use reasonable response parameters to determine the success of the design’s defensive strategy, including the physical design’s delay, denial and mitigation features. From these table-top exercises on other new reactor designs, as well as live force-on-force exercises at operating plants, many general guidelines have been developed. Some are delineated below.

General practice has a number of barriers between the general public and vital equipment. From the outside in, they are: the Owner Controlled Area boundary, the Protected Area (PA) boundary, the delay fence, and the Vital Area (VA) boundary. Each has a level of increasing access control requirements. As each barrier is crossed, different access permission is required and personnel searching is performed. Each barrier will have defined access ports for both pedestrian and vehicular traffic.

Of these barriers, the reactor building will be a part of only the VA boundary. The VA boundary is a limited access, hardened boundary that forms the final barrier to vital equipment or areas. The exact portions of the reactor building that forms part of the VA boundary will be determined after preliminary determination of target sets. Access control will include automatic and security officer features. The reactor building portions of the VA boundary must include security officer protection and response features. The VA boundary must be resistant to DBT blast definitions.

Rules of thumb and guidelines that, if followed, will make the nuclear plant more secure are:

1. Define all Vital Areas
2. Limit the number of access paths into the facility and the VA, consistent with life safety requirements
3. Harden the VA boundary consistent with the DBT and table top results
4. Provide for protection of response personnel
5. Provide for natural “funneling” of approaching attackers
6. Provide for access controls specific to the defense level of the area
7. Make the facility resistant to a large fire or explosion
8. Have clear fields of fire from areas inside the VA to areas outside the VA
9. Ensure that adequate spent fuel cooling is available following a beyond design basis event
10. Establish a preliminary guard force that meets requirements of alarm station monitoring and threat response

A1.6 REACTOR BUILDING REQUIREMENTS SUMMARY

This study uses the NGNP Pre-Conceptual-Design Report (PCDR) reactor building design as a starting point [1]. It is a rectilinear building, configured to house a 500 MWth PBMR which is coupled to an intermediate cooling loop via two in-series intermediate heat exchangers

(IHXs). The primary loop flow is provided by an electric-drive helium circulator. Thermal energy is transferred in the IHXs to a secondary helium loop, and then on to the power conversion and hydrogen production processes. Figure 5 shows a cut-away view of the PCDR reactor building design and its separation with adjacent buildings.

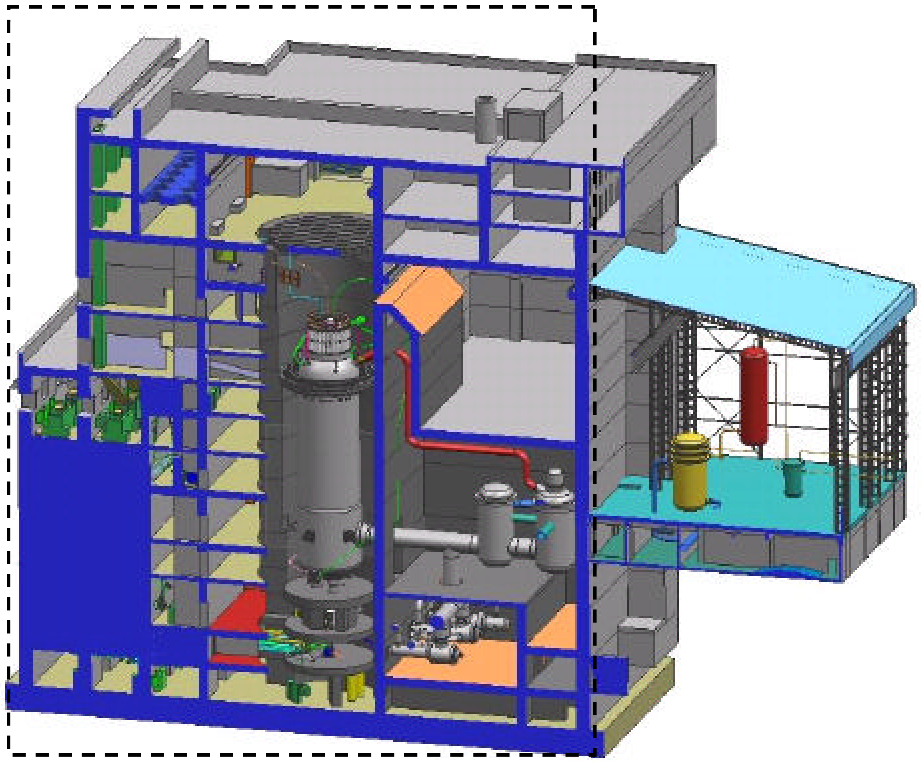

Figure 5 NGNP Reactor Building in Preconceptual Design Report

The major components and systems located within the PDCR reactor building include the following:

- Reactor Unit
- Core Conditioning System
- Reactor Cavity Cooling System
- Fuel Handling and Storage System
- Helium Services System
- Heating, Ventilating, and Air Conditioning Systems
- Pressure Relief System
- Other NHSS Support Systems
    - Portions of the Heat Transport System

- Primary Heat Transport System
- IHX vessels, circulators
- Parts of the Secondary Heat Transport System.

The top-level goals for the NGNP Project include the design, construction, and demonstration of a high-temperature gas-cooled nuclear reactor that will exhibit the capability of both electrical and hydrogen production. In addition, the project will demonstrate features and attributes that are unique to HTGRs, including passive safety and defense-in-depth that relies primarily on inherent design capability and minimal operator action.

This study is based on the use of a pebble-bed modular reactor (PBMR) derived from development work on-going in South Africa, and is focused on three key aspects of the NGNP reactor building:

- Protection of the Reactor Building structure and safety related SSCs contained therein from the effects of pressure and temperature loads from events involving Heat Transport System pipe breaks and depressurization
- Mitigation of radiological consequences following a postulated accident event
- The extent to which the NGNP reactor building should be embedded

The need for retention of radio-nuclides by the reactor building will dictate building features such as the boundary leak rate, the need for barriers to internal flow communication, and the need for engineered features like filters at the building exhaust point. These features are investigated in Section A2. The extent to which the building is set below grade will impact, and be impacted by, the need for radionuclide retention, so it is appropriate that these aspects be studied together.

The design of the reactor building is developed in response to a wide range of requirements, and retention of radioactivity released during an accident is only one dimension. The process to establish building design requirements in general is called Requirements and Functional Analysis. Typically, goals expressed at the highest level must be satisfied by design selections made at that level. These top level design selections then become input requirements at the next level, and must be analyzed and satisfied by design selection. The process continues downward until all details in the design are specified. Periodically, during the design process, the results must be reviewed from top to bottom and from bottom to top, to ensure that selections made during the process are optimal and consistent with top level requirements.

Through this process, a set of NGNP building functional requirements has been derived. Key drivers for the building design can be grouped in several ways. Spaces within the building are driven primarily by the geometry of the equipment and systems that will be housed within the building. The strength of the building members will be driven primarily by the loads imposed by the equipment and systems, or by other building members. Their thickness is driven primarily by strength or radiation shielding requirements. However, the building is also allocated functional requirements to protect the equipment and systems within it for both investment protection and to satisfy safety requirements for nuclear plants. The building is allocated the requirement to protect nuclear systems during internal and external hazard events (seismic, tornado missile, aircraft impact, malevolent attack, and others). This study examines the need to allocate a functional requirement to retain radioactive material, and the extent to which it should be embedded to meet this and other requirements.

Based on the information presented in this section, a set of technical and functional requirements for the Reactor Building was developed that incorporates the required and supportive safety functions of the reactor building; the physical security requirements; and other requirements to support non-safety function such as plant operation, maintenance, and access for testing and inspections. These requirements are listed in Table 6.

Table 6 Preliminary Technical and Functional Requirements for NGNP Reactor Building

T&FR Number	Requirement
NHSB-1.0	The scope of functions and requirements for the reactor building shall be allocated to the Nuclear Heat Supply Building (NHSB), NHSS HVAC System, and NHSB Pressure Relief System (PRS) as described in the PCDR.
NHSB-2.0	The Nuclear Heat Supply Building (NHSB) shall perform the following functions:
NHSB-2.1	House and provide civil structural support for the reactor and all the SSCs within the scope of the NHSS (building geometry/space)
NHSB-2.1.1	Provide personnel and equipment access for plant construction, maintenance, operation, and inspection of SSCs in the NHSS and for routine and emergency ingress and egress of plant workers
NHSB-2.1.2	Provide radiation shielding for plant workers and the public during normal operation to keep radiation exposures ALARA
NHSB-2.1.3	Limit air flow in neutron fields to keep routine releases of activation products and other releases during normal plant operation from all sources of radioactivity inside the NHSB ALARA
NHSB-2.1.4	Maintain internal environmental conditions (temperature, humidity, air refresh) for NHSS SSCs and operators..
NHSB-2.1.5	Support zoning requirements for HVAC, radiation, fire protection, flood protection, and physical security protection

T&FR Number	Requirement
NHSB-2.1.6	The NGNP shall be designed to physically protect the safety related SSCs in the NHSS from hazards associated with the Hydrogen Production System, Power Conversion System and BOP facilities.
NHSB-2.2	Resist all structural loads as required to support safety functions allocated to the NHSB
NHSB-2.2.1	Provide structural support for the reactor pressure vessel and its internals
NHSB-2.2.2	Provide structural support for the PHTS Helium Pressure Boundary (HPB) and the part of the SHTS HPB inside the NHSB
NHSB-2.2.3	Provide structural support and protection for the NSSS SSCs that contain sources of radioactive material outside the reactor vessel including the FHSS, HSS, and systems containing radioactive waste
NHSB-2.2.4	Protect the NHSB, PHTS HPB and other NHSS SSCs against loads imposed by faults including pipe breaks, chemical releases, fires, seismic failures, and explosions located in the HPS Acid Decomposer Building, SG Building, and other adjacent buildings associated with the HPS, PCS, and balance of plant. Protect the PHTS HPB from loads imposed by SHTS piping resulting from faults including structural failures in the HPS Acid Decomposer building and SG Building
NHSB-2.2.5	Provide structural support for all NHSS SSCs that provide a required safety function for all sources of radioactive material within the NHSB
NHSB-2.2.5.1	Provide structural support for SSCs required for confinement of radioactive material
NHSB-2.2.5.2	Provide structural support for SSCs required to maintain core and reactor pressure vessel geometry
NHSB-2.2.5.3	Provide structural support for SSCs required to control core heat removal including the core, reactor vessel, reactor cavity and RCCS
NHSB-2.2.5.4	Provide structural support for SSCs required for control of heat generation including the reactivity control rods
NHSB-2.2.5.5	Provide structural support for SSCs required for control of chemical attack
NHSB-2.2.6	Provide structural support for all NHSS SSCs that provide a supportive safety function for all identified LBEs as necessary to meet the TLRC (supportive safety function)

T&FR Number	Requirement
NHSB-2.2.7	Provide confinement of radioactive material during normal operation and during LBEs (AOOs, DBEs, and BDBEs) as necessary to meet the TLRC. This function is also supported by the NHSB PRS and NHSS HVAC (supportive safety function) (A design goal of a source term reduction factor of 10 is set in Section A3 for I-131 and Cs-137 for releases from the PHTS during design basis events involving DLOFC)
NHSB-2.2.8	Control building leakage and limit air ingress to the core following a large breach or breaches in the PHTS HPB, FHSS, and HSS for all identified LBEs as necessary to meet the TLRC (supportive safety function) (quantification of this requirement to be determined)
NHSB-2.3	Protect the SSCs within the NHSS that perform safety functions from all internal and external hazards as identified in the LBEs as necessary to meet the TLRC
NHSB-2.3.1	Protect the SSCs within the NHSS that perform required safety functions from LBEs involving PHTS and SHTS HPB leaks and breaks. This function is also supported by the NHSS Pressure Relief System.
NHSB-2.3.2	Protect the SSCs within the NHSS that perform required safety functions from LBEs involving missiles from rotating machinery and other internal sources
NHSB-2.3.3.	Protect the SSCs within the NHSS that perform required safety functions from LBEs involving internal fires
NHSB-2.3.4	Protect the SSCs within the NHSS that perform required safety functions from LBEs involving internal floods
NHSB-2.3.5	Protect the SSCs within the NHSS that perform required safety functions from LBEs involving hydrogen process hazards
NHSB-2.3.6.	Protect the SSCs within the NHSS that perform required safety functions from LBEs involving seismic events
NHSB-2.3.7	Protect the SSCs within the NHSS that perform required safety functions from LBEs involving accident aircraft crashes and transportation accidents
NHSB-2.3.8	Protect the SSCs within the NHSS that perform required safety functions from LBEs involving high winds and wind generated missiles
NHSB-2.3.9	Protect the SSCs within the NHSS that perform required safety functions from LBEs involving other internal and external hazards (requirements to be determined)

T&FR Number	Requirement
NHSB-2.3.10	Protect the SSCs within the NHSS that perform supportive safety functions from LBEs as necessary to meet the top level regulatory criteria as necessary to meet the TLRC (requirements to be determined)
NHSB-2.4	Provide physical security of vital areas within the NHSB against acts of sabotage and terrorism
NHSB-3.0	The NHSB Pressure Relief System shall perform the following functions:
NHSB-3.1	The PRS shall be compatible with the NHSB boundary functional requirement.
NHSB-3.2	The PRS shall be designed, and the NHSB compartments shall be sized so that leaks and breaks up to 10 mm equivalent break size on the PHTS HPB or that part of the SHTS HPB inside the NHSB do not open the PRS so that HVAC filtration capability shall be continuously maintained
NHSB-3.3	PRS shall open to prevent overpressure and thermal damage to any safety related SSC, remove the pressure driving force for radionuclide releases from the reactor building, and depending on the actual design that is chosen, may also be required to reclose to enable post-blow-down filtration for the following design basis event conditions.
NHSB-3.3.1	Breaks in the PHTS HPB in each NHSB compartment containing NHSS piping or vessels up to 100 mm equivalent single ended break size
NHSB-3.3.2	Breaks in the SHTS HPB in each NHSB compartment containing NHSS piping or vessels up to 1 meter equivalent single ended break size
NHSB-3.3.3	Breaks in the HSS piping up to 100 mm equivalent single ended break size
NHSB-3.4	The PRS shall open and prevent excessive damage to the NHSB in response to a Beyond Design Basis 1 meter break in the PHTS piping; excessive damage is defined by exceeding the TLRC using realistic assumptions
NHSB-4.0	The NHSS HVAC System shall perform the following functions:
NHSB-4.1	a) Maintain internal environmental conditions for all NHSS SSCs during normal operation, b) survive conditions for all AOO and DBE LBEs, and c) provide a post-event cleanup function. (supportive safety function)

T&FR Number	Requirement
NHSB-4.2	Maintain environmental conditions for SSCs that perform supportive safety functions during LBEs (supportive safety function)
NHSB-4.3	Perform radionuclide filtration functions as required to meet the TLRC for all LBEs (supportive safety function)
NHSB-4.4	Perform protective functions to isolate the HVAC during DBE and BDBE HPB breaks to prevent damage from high temperatures and pressures in building compartments during depressurization to enable post blow-down filtration (supportive safety function)

A2 DEFINITION OF ALTERNATIVE REACTOR BUILDING DESIGN CONCEPTS

In order to gain insight into alternative strategies for implementing the safety functions assigned to the NGNP reactor building, a set of alternative design concepts were developed and evaluated. The approach to identifying and evaluating these alternative strategies is comprised of the following steps:

1. Identify both required safety functions and supportive safety functions, as discussed in Section A1
2. Identify alternative design strategies to meet each safety function
3. Define a discrete set of alternative reactor building design configurations with appropriate combinations of features
4. Evaluate each alternative configuration against technical and functional performance criteria, including the following
    a. Normal Operating Functional Requirements
    b. Investment Protection Functional Requirements
    c. Safety Functional Requirements
    d. Physical Security and Aircraft Crash Functional Requirements
    e. Capital and O&M Cost Impact
    f. Licensability
5. Identify a basis for recommending one or more alternative configurations for further development in the Conceptual Design Phase.

Within the scope of this study, most of these dimensions are evaluated on the basis of the PDCR development and engineering judgment. For example, the alternative configurations of the reactor building are not expected to have any influence on its ability to provide space or physically support the reactor system equipment. Some features that are included to assess radionuclide retention, such as enhanced building leak-tightness, may impact the ability of operators to gain access to spaces inside the building. This type of impact is assessed by engineering judgment supported by radiological release calculations that are presented in the next section.

The PBMR primary coolant is high-temperature, high-pressure helium. In the event of a leak or pipe break, reactor coolant will fill the reactor building spaces which are normally filled with ambient pressure air. Expansion of the helium and heating of the air will both contribute to pressurization of the internal building spaces. There are two key building design strategies available to address the possibility of internal pressurization: to make the building strong enough to resist the pressure buildup, or to vent the building to limit the internal pressure. In either case, the over-arching requirement is that the building should maintain the geometry and functionality of the reactor shutdown and decay heat rejection systems when subjected to pressure transient loads (in combination with other loads as required by design codes).

If a primary design selection is to vent the building, there are several ways to achieve this goal. These strategies include the creation and maintenance of physical openings in the building boundary and between internal compartments (open vented building). There are several variants to the building vent concept, including closed vents that open when pressure is applied (rupture panels), and openings that re-close after pressure equalizes (re-closable dampers). Vent openings can be equipped with filters to reduce the release of entrained solids.

It follows that the reactor building members must be made strong enough to resist the load applied by the internal pressure. In a vented building design, that pressure may be low for the outermost building boundary, but may still be substantial for some internal structural members during the blow-down. In an un-vented building, the sub-compartment pressurization will still exist, but the outermost boundary of the building will also be subjected to a pressure load. This may require that the building members be made much stronger than would be required without the pressure load. In order to achieve a reasonable design, it may be necessary to make the outer boundary of the building from curved walls and slabs, so that they are not required to resist bending moments.

The reactor building must also protect SSCs from hazards that originate from internal sources. These might include missiles from rotating machinery, ejected parts, and fluid jet impingement forces, as well as fire and flooding events. NGNP will have available all the strategies usually applied to these hazards, including internal walls and slabs to create zoning separation, train separation, flood source isolation, and fire load management. Features in this class are not likely to be influential with respect to radionuclide retention, but they could have some influence on air ingress mitigation.

The reactor building must also protect SSCs from external hazards. Hazards in this class include seismic events, extreme weather and weather-generated missiles, aircraft crash, and resistance to malevolent attack. The design features used to make the reactor building resistant to earthquake, tornado missiles, and aircraft crash are usually sufficient to assure that it is a robust structure. It may turn out that the design needed for protection from external hazards is sufficient to resist internal pressurization without significant additional features. Additional strength in the reactor building will usually make it more resistant to internal, external, and pressurization loads. However, adding strength also increases the member size. In order to maintain adequate space for normal functions, the overall structure may increase in size and cost.

Unplanned discharge of reactor coolant into the building will result in the release any radio-nuclides present as circulating activity. Depending on the size of the HPB break, a percentage of the radioactive inventory characterized as dust and plateout activity will also be released. Design strategies available to enhance retention of these radio-nuclides by the reactor building include both active and passive filtration using HVAC systems designed for normal operation or specifically for post-event cleanup. This strategy can be affected by the building design for zoning and contamination control. It is also possible to enhance the deposition and plate-out that will occur on building surfaces by making the vent pathway tortuous and providing rough surfaces. While this strategy would increase the deposition in the building, it would also increase the sub-compartment peak pressure during the release. Some flow will also occur across the building boundary, in parallel with flow out any vent pathway. Making the building boundary more leak-tight will limit the potential for radionuclide release across the building boundary.

The reactor building may also play a role in limiting the possibility of air ingress after a break in the HPB. Air ingress could only occur after complete depressurization of the primary loop, and would be inherently limited if the HPB were broken only in one place or at a low point on the system. The extent to which air ingress is a concern is addressed in subsequent sections. The two distinct strategies available to address and limit ingress of air into the core following a pressure boundary failure are introduction of inert gas into the HPB, or introduction of inert gas into the reactor building. If it is deemed necessary to inject inert gas into the HPB, the logical strategy would be to introduce nitrogen using the connections that already exist at the fuel discharge pipes. These connections are normally used to provide cool helium to discharged fuel, but could readily be designed to be connected to a nitrogen source. If it is deemed necessary to inert the reactor building, a system could be provided to release nitrogen or carbon dioxide into the building. The sizing of a building inerting system would be affected by the use of re-closable vent path dampers and by the leak-tightness of the reactor building boundary. Inerting of the building would introduce some safety issues for plant workers.

To assess the need for radionuclide retention by the reactor building, a set of alternative designs have been developed as summarized in Table 7. The strategies to be examined include the following:

- Open vent path, unfiltered
- Filtered, with a controlled leak path
    - Designed to filter only the delayed source term
    - Designed to filter both the prompt and delayed source term
    - Participation in the blow-down over a range of sub-compartment volumes
    - Variation in building envelope leak tightness
    - Addition of expansion volume
- Pressure retaining building with low leak rate
    - Addition of expansion volume

These strategic design alternatives are driven by specific reactor building functional requirements. The requirement to maintain reactor building geometry leads to a requirement to resist loads imposed during pressure transients. The strength required for the reactor building, and therefore its cost, will be impacted by the vent path and its configuration, or the selection of a pressure retaining building. Rupture panels, re-closable dampers, and filter systems will also play a part in determining the design basis pressure loading. For any given configuration, additional expansion volume will reduce the pressure transient, but may significantly affect the cost.

The strength required to resist pressure transients may not change the design of the building, because other loads, such as seismic, may be controlling. Embedment of the reactor building could reduce the seismic loading, and any building spaces below grade are expected to have an external soil and groundwater pressure that would oppose pressure transient loadings. Seismic and pressure transient loads are not usually combined, because the events are not postulated to occur simultaneously.

Table 7 Alternative Reactor Building Strategies for Performing Safety Functions

No.	Design Description	Vented Area Leak Rate Vol % /day	Pressure Relief Design Features	Post blow-down re-closure of PRS shaft?	Radionuclide Filtration	
					Blow-down release	Delayed fuel release
1a.	Unfiltered and vented	50-100	Open vent	No	None	Passive
1b	Unfiltered and vented with rupture panels	50-100	Internal + External rupture panels	No	None	Passive
2	Partially filtered and vented with rupture panels	25-50	Internal + External rupture panels	Yes	None	Active HVAC
3a	Filtered and vented with rupture panels	25-50	Internal + External rupture panels	Yes	Passive	Active HVAC
3b	Filtered and vented with rupture panels and expansion volume	25-50	Internal + External rupture panels + expansion volume	Yes	Passive	Active HVAC

No.	Design Description	Vented Area Leak Rate Vol % /day	Pressure Relief Design Features	Post blow-down re-closure of PRS shaft?	Radionuclide Filtration	
					Blow-down release	Delayed fuel release
4a	Pressure retaining with internal rupture panels	0.1-1	Internal rupture panels	N/A	Passive	Passive
4b	Pressure retaining with internal rupture panels and expansion volume	0.1-1	Internal rupture panels + expansion volume	N/A	Passive	Passive

The reactor building is also required to protect other safety related components located within the building from events that include pressure transients, internal hazards like fire, and external hazards like seismic events and aircraft crashes. These requirements will generally cause the building to be very robust, and to be divided into rooms and levels by thick walls and slabs. Because the building must be designed to respond to a wide range of required functions, it also has attributes that frequently contribute to its safety performance in ways that were not specifically required. For example, selecting a tortuous vent path to limit the peak pressure loading also provides flow path turns, surfaces, and quiescent regions that will enhance radionuclide retention, even without any requirement. These functions are designated as supportive functional requirements.

There are also a number of design strategies available to enhance or capitalize on supportive safety features. Normal HVAC systems, not required to perform any safety function, can contribute to dose reduction for events that are analyzed realistically, if they are appropriately designed and are protected from event-related damage. The strategies available to enhance HVAC systems include the installation of blast dampers to prevent damage from rapid pressurization, isolation dampers, and physical separation of HVAC trains.

Other features that provide support to achieving safety-related goals, but are not necessarily safety related include systems and design features that can mitigate against radionuclide transport after an accident pressure transport. During normal operation, building features that enhance radiation and contamination control include the design of the HVAC systems, separation of components, and configuration of radiation and contamination control zones. During a blow-down event, passive filtration through engineered filter system or through the building boundary will reduce radionuclide transport. The leak tightness of the building boundary is important to the amount of retention that can be achieved. Following a pressure release, when air and coolant are moving very slowly through the building and across the boundary, the building will also act to impede radionuclide release to the environment. The primary mechanisms for radionuclide reduction in a post-blow-down phase are passive filtration and deposition on building surfaces, and active filtration by HVAC systems. The design of the building vent pathway and the building leak rate are parameters that can have significant influence on the radiological performance of the building during this phase.

The building also offers some important passive mitigation to air ingress following a HPB break event. The building limits the amount of air available for ingress. It also provides a space that would permit the establishment of an inert gas envelope surrounding the HPB. The building leak rate could be an important parameter affecting this capability. If air ingress is established as a requirement, it will also be possible to introduce inert gas to the HPB via the fuel discharge connections.

The functional requirements for the reactor building apply to all of the alternatives develop for this study. The alternatives have many features in common, including the following:

- Robust structural design to protect against loads from internal and external hazards including seismic events, hydrogen process hazards and physical security against malevolent acts as described in selected licensing basis events.
- HVAC filtration system provided to maintain environmental conditions and provide radionuclide retention ALARA during normal operation and for anticipated operational occurrences (AOOs).
- Blast panels or isolation devices provided to isolate/ protect normal operation HVAC from HPB breaks.
- Physical isolation/separation provided for fuel storage and safety related SSCs from blow-down vent path following HPB breaks.
- Each alternative configuration considered can be combined with all evaluated embedment alternatives.
- Design of the reactor and HPB pressure boundary and passive safety characteristics will limit the potential for air ingress; capability to support ad hoc emergency actions to inert the reactor core cavity via injection of helium or nitrogen is inherent in the PBMR fuel handling system design.
- All designs considered limit the supply of outside air by controlling building leak rate but leak rate varies among the considered alternatives.
- All alternatives have the capability to inject Helium or Nitrogen gas into the core via pipes connected to the Core Unloading Device in the event that this action is deemed necessary in the next phase of the NGNP design. The need for such action has not yet been identified based on current understanding of air ingress phenomena and is not considered further in this study.

A2.1 ALTERNATIVE 1a UNFILTERED AND VENTED

The reactor unit is arranged within the NGNP reactor building such that major components and component groups are placed within interior vaults or sub-compartments and interconnected with high-integrity reactor piping (Refer to Figure 5). Internal vent pathways have been established such that a break in the HPB which occurs in the reactor cavity will expand through a designed opening to the IHX cavity. If a building vent is provided, gas will then flow from the IHX cavity to a building vent vault (shown as the pressure relief pathway), and to the vent itself which connects to the environment. If a break occurs in the core conditioning system (CCS) vault (not shown in Figure 1), it will flow first to the reactor cavity, and then follow the same path as a break in the reactor cavity, to the IHX cavity, building relief pathway, and environment. If a break occurs in the spaces occupied by the fuel handling support systems (FHSS) or helium support systems (HSS) it will be able to flow directly to the building relief pathway. Any primary or secondary helium release in the IHX vault will be able to expand in the IHX cavity and then to the pressure relief pathway and to the environment. Other rooms and spaces within the reactor building will also have some participation in the blow-down pressure transient, because gasses will expand into the rest of the building through penetrations, doors, and other imperfect seals.

Alternate 1a is shown in schematic form in Figure 6. This design alternate assumes that open vent paths interconnect the reactor cavity to the IHX cavity. Open vent paths also connect the IHX cavity and the other rooms containing HPB components to the pressure relief pathway. The spaces that contain the HPB equipment (outlined in black in Figure 1) are designated as the pressurized zone, and designed to have a leak rate of 50-100 vol%/d. This means that the pressurized zone would leak at a rate such that 50-100% of its volume would leak out in one day if it was maintained at its design pressure. Its design pressure will be estimated by performing pressure transient calculations. A carefully designed and constructed, industrial-grade building, with reasonable care taken in the design of doors, windows, and penetrations, should be able to achieve such a leak rate. Gasses may leak from the pressurized zone into the other spaces in the reactor building, which are surrounded by the reactor building envelope (shown in magenta in Figure 6). The reactor building envelope is expected to also have a leak rate on the order of 100 vol%/d. The pressure relief pathway is assumed to be isolated from normal building HVAC systems so that normally required zoning and contamination control can be managed. This alternative is developed for purposes of this study only. It would be difficult to create a design with both open vent paths and isolated HVAC that would function properly. In alternate 1a, it is assumed that reasonable care is taken to create smooth transitions between volumes, avoid high flow loss entrance and exit effects, and minimize the pressure drop along the vent path. In Alternate 1a, there is no feature provided to close the vent path after the blow-down. No filters are provided in this alternative. Radioactivity reduction mechanisms include plate-out, condensation, settling, and decay of radio-nuclides. Following a release event, normal HVAC would be used to provide clean-up, if it is available, but HVAC has no safety function.

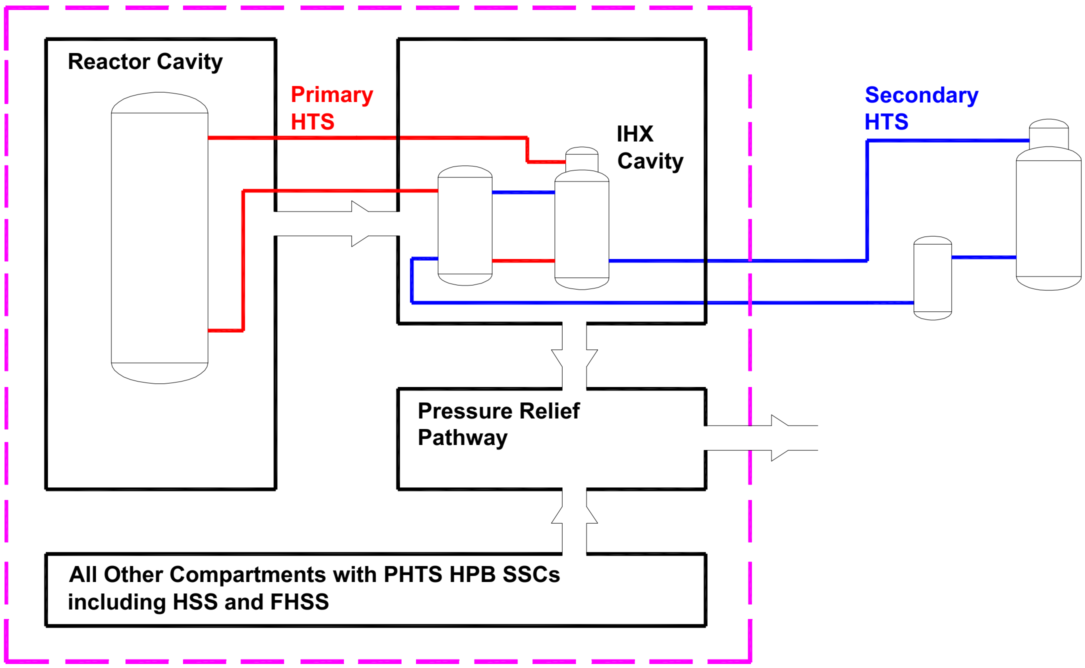

Figure 6 Reactor Building Alternative 1a – Unfiltered and Vented

The advantages for alternative 1a include the following:

- Minimizes pressure transient
- Inherently reliable as it is completely passive

The disadvantages for alternative 1a include the following:

- More difficult to control release of air activation products during normal operation
- No engineered features to retain accident related fission products or limit air ingress to building following blow-down
- May require very large HVAC equipment to maintain zoning and zone pressure gradients

A2.2 ALTERNATIVE 1b VENTED AND UNFILTERED WITH RUPTURE PANELS

Alternative 1b is shown in Figure 7 below.

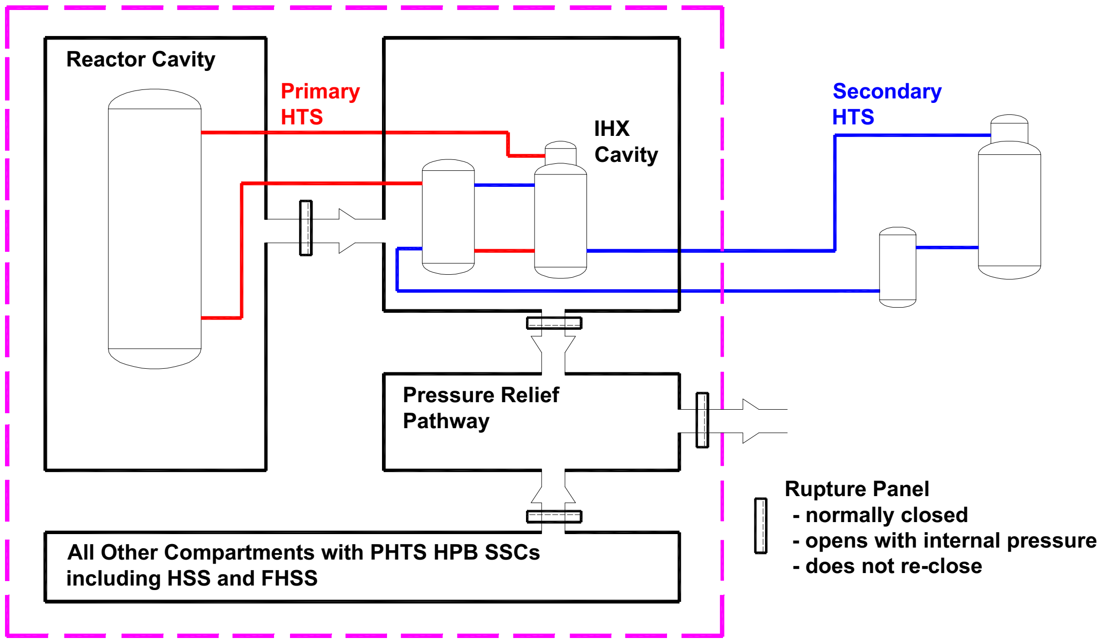

Figure 7 Reactor Building Alternative 1b – Unfiltered and Vented with Rupture Panels

Alternative 1b is similar to 1a, except that rupture panels have been added to separate internal spaces from each other, and to separate the pressure relief pathway from the environment. The various leak rates are still assumed to be the same as Alternative 1a. These panels are expected to consist of a thin metal or plastic membrane supported on one side. If pressurized from the supported side, the panel is designed to deform and rupture at a low, predictable pressure. There may be devices to aid the rupture process (knife points) that the membrane strikes when it distorts. If pressurized in the opposite direction, against the supports, the membrane will fail at a higher pressure. The introduction of rupture panels will increase the peak pressure transient for all events.

The advantages of Alternative 1b over alternative 1a include the following:

- Easier to control release of air activation products during normal operation
- Easier for HVAC to control pressure zones

The disadvantages of Alternative 1b include the following:

- Increases pressure transient and design pressure
- Rupture panel design will have lower reliability than open vent
- No engineered features to retain accident related fission products

A2.3 ALTERNATIVE 2 PARTIALLY FILTERED AND VENTED WITH RUPTURE PANELS

Alternative 2 is shown in Figure 8, below.

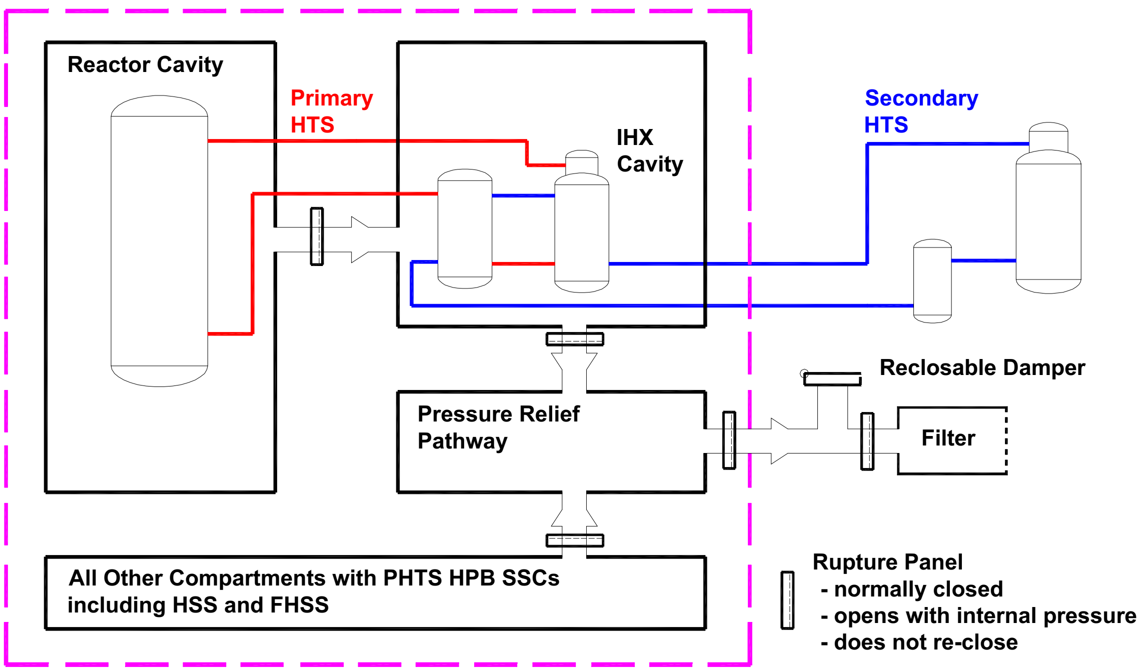

Figure 8 Reactor Building Alternative 2 – Partially Filtered And Vented With Rupture Panels

Alternative 2 is based on Alternative 1b, but is provided with two additional features. A re-closable damper and a filter are added in parallel at the point where the pressure relief pathway connects with the environment. In addition, the leak rate for the pressurized part of the building (outlined in black) is reduced to 25-50 vol%/d at 1.1Bar.

The re-closable damper is a device which will open with a small internal pressure, and when the inside and outside pressure have equalized, will re-close. It may be pressure actuated to open and gravity actuated to close, or other mechanism may be used. The filter, preceded by a rupture panel, is provided to reduce the concentration of particulate fission products in any gas stream that flows out of the building after pressure has equalized. The rupture panel preceding the filter is designed to fail open during the initial blow-down, which also opens the damper. The vent path becomes a feature that discriminates between the initial blow-down, which might include prompt source terms, and subsequent releases, which might include delayed source terms.

To achieve a lower leak rate for the pressurized spaces, it will be necessary to upgrade a number of components. Doors and penetrations will probably need to be nuclear grade and to use soft and inflatable seals. The building boundary will need to be sealed at element joints by gaskets, caulking, or welding. These features increase in cost and complexity with increasing design pressure.

The advantages of Alternative 2 include the following:

- Rupture panel on vent paths make it easier to control release of air activation products during normal operation, and for HVAC to control pressure zones
- Re-closable damper minimizes pressure transient by opening vent during large HPB break and increases effectiveness of filter
- Filter on vent path mitigates the delayed fuel release source term

The disadvantages of Alternative 2 include the following:

- Rupture panels increase the peak pressure transient
- Damper and filter add to capital cost

A2.4 ALTERNATIVE 3a FILTERED AND VENTED WITH RUPTURE PANELS

Alternative 3a is shown in Figure 9, below:

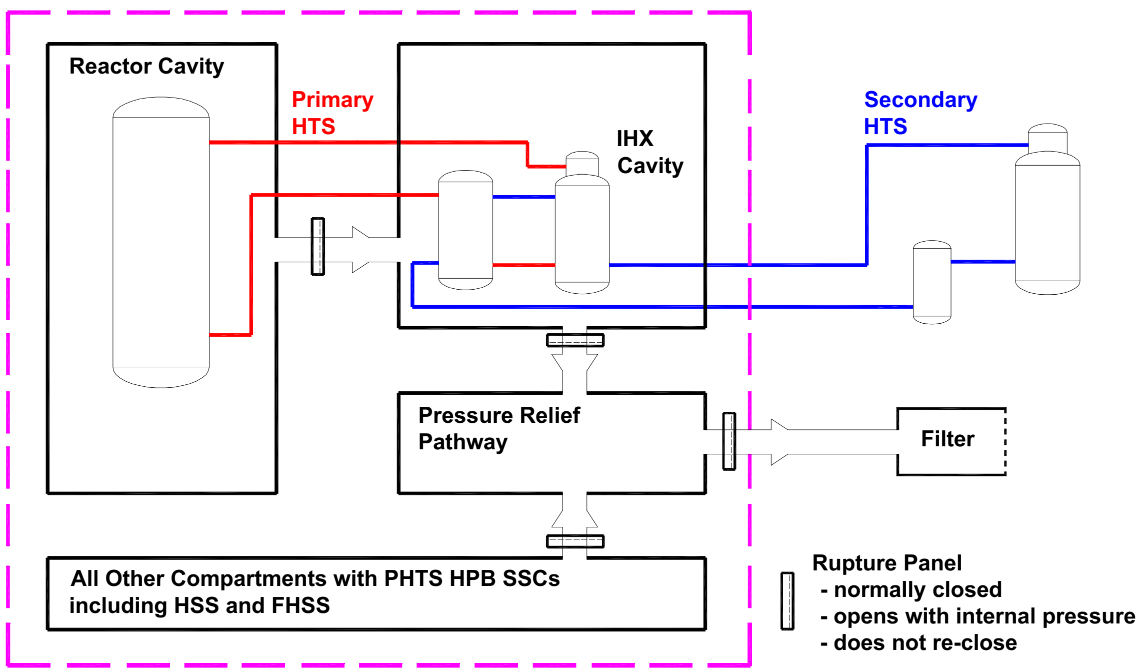

Figure 9 Reactor Building Alternative 3a - Filtered and Vented With Rupture Panels

Alternative 3a is similar to Alternative 2, except that it does not have a re-closable damper on the exit. In this alternative, any reactor building pressurization event must be relieved through the filter or by leaking through the building boundary. This design approach imposes potentially severe design constraints on the filter, and will increase the design pressure. It may be necessary to design the filter for high temperature as well as high flow. Increased design pressure will make the achievement of the leak rate more costly.

The advantages of Alternative 3a include the following:

- Internal rupture panels make it easier to control release of air activation products during normal operation, and control HVAC zones
- Filter reduces both the prompt release source term and the delayed fuel release source term

The disadvantages of Alternative 3a include the following:

- Increased peak design pressure
- Increased cost due to more difficult filter design and higher design pressure.

A2.5 ALTERNATIVE 3b FILTERED AND VENTED WITH RUPTURE PANELS AND EXPANDED VOLUME

Alternative 3b is shown in Figure 10, below.

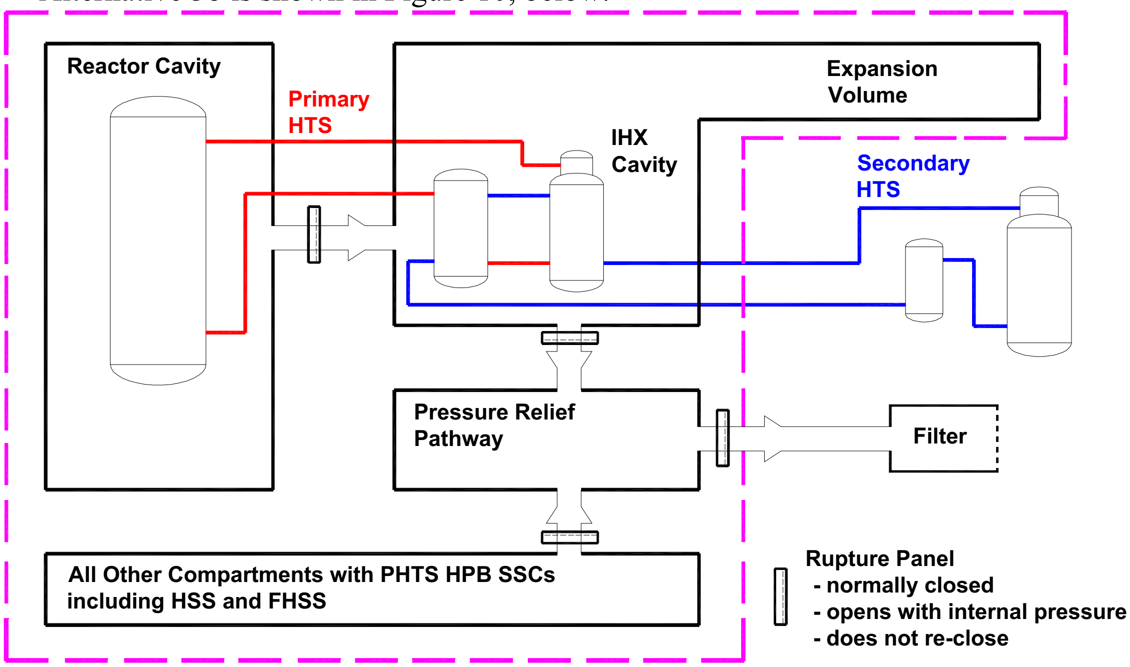

Figure 10 Reactor Building Alternative 3b Filtered and Vented With Rupture Panels and Expanded Volume

Alternative 3b is similar to Alternative 3a, except that additional expansion volume is assumed. Such expansion volume could be achieved by increasing the volume of the IHX vault, or by constructing a separate building and connecting it to the IHX vault by a tunnel or passage. Additional volume is expected to reduce the peak pressure for blow-down events. This would make it easier to develop a design for the exit point filter and would reduce the cost of the features needed to achieve the leak rate. The key disadvantage to Alternate 3b would be additional cost related to the increased building space.

A2.6 ALTERNATIVE 4a PRESSURE RETAINING WITH INTERNAL RUPTURE PANELS

Alternative 4a is shown in Figure 11, below.

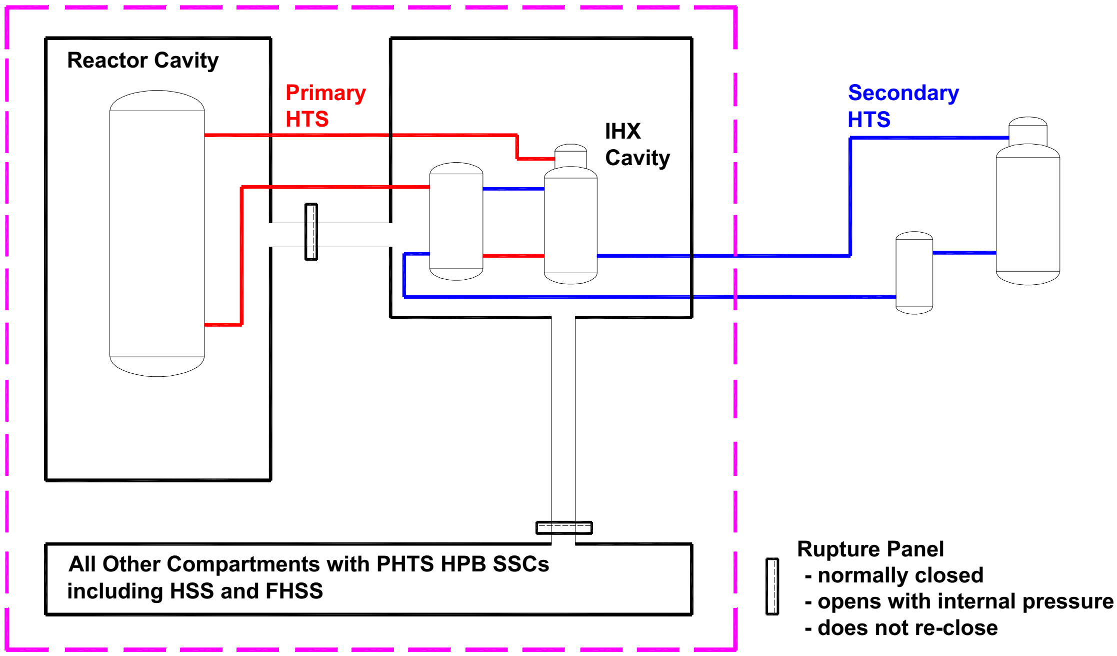

Figure 11 Reactor Building Alternative 4a Pressure Retaining With Internal Rupture Panels

Alternative 4a is based on the previous models, but does not have any engineered vent path. It is developed and analyzed to develop insight into the radiological retention performance that a pressure-retaining structure might have when subjected to a range of postulated NGNP accident events. The imposition of a pressure retaining building concept on the NGNP project would probably require a significant revision to the design approach for the reactor building. For analytical purposes, this alternative assumes that the pressure retaining features could be imposed on the pressurized zone within the reactor building. It is likely that the pressure retaining boundary location would be made larger, and located at the building outer boundary, making the NGNP reactor building conceptually similar to a dry containment normally applied to light water reactor designs. This case will also be analyzed.

The leak rate for the pressurized zone in Alternative 4a has been reduced to ~0.1-1 vol%/d. To achieve a leak rate this low, it would be necessary to provide nuclear grade airlock doors with inflatable seals and hard (welded) penetrations. It would also be necessary to add a steel or epoxy liner system.

The advantages of Alternative 4a include the following:

- The elimination of the vent path means all radioactive materials released from the HPB are retained within the reactor building
- The low leak rate reduces transport of radio-nuclides to the environment via building leakage.

The disadvantages of Alternative 4a include the following:

- Increased design pressure, with increased costs
- Very low allowable leak rate, with increased costs
- Higher design pressure may require curved walls and slabs
- Higher design pressure may require engineered penetrations, and may require alternative features to maintain passive heat rejection
- Increased inspection and testing required
- Pressure retaining boundary failure modes may have significant consequences (however they are low probability events)

A2.7 ALTERNATIVE 4b PRESSURE RETAINING WITH INTERNAL RUPTURE PANELS AND EXPANDED VOLUME

Alternative 4b is shown in Figure 12, below.

Alternative 4b is based on Alternative 4a, but includes an additional expansion volume. Such additional volume would reduce the design pressure. Lowering the design pressure should reduce the cost, but savings may be overcome by the cost of additional space.

Alternative designs from 1a to 4b have been identified so that the pressure transient response and radionuclide retention functions can be judged along an increasing dimension of building complexity, within the time and budget constraints of the study. These aspects are addressed in Section A3.

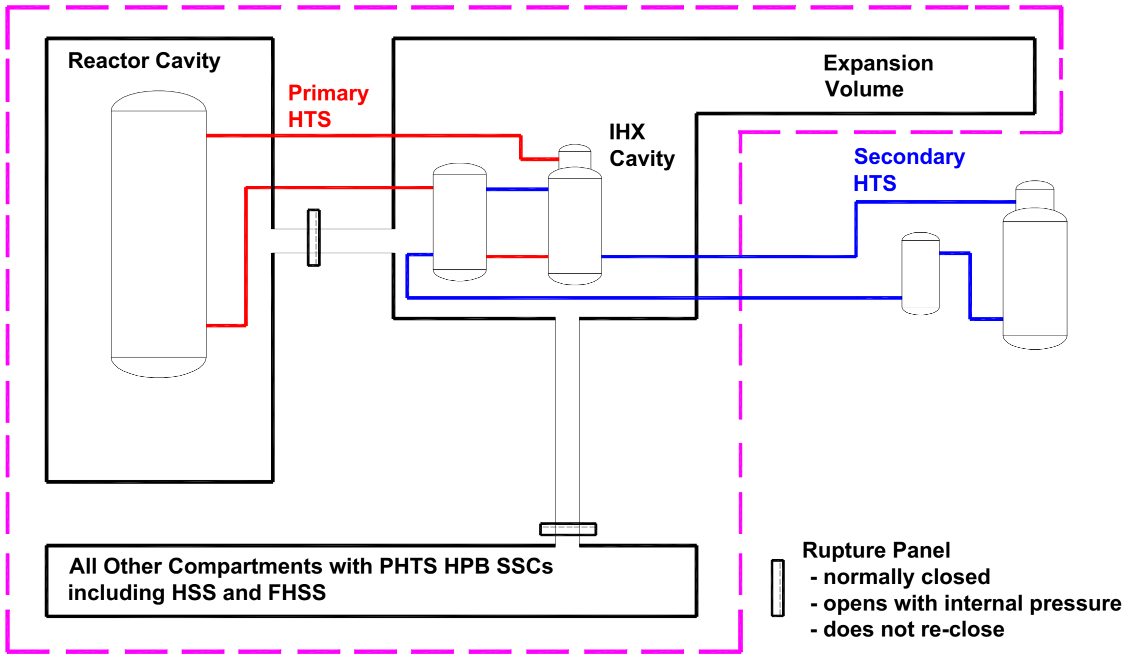

Figure 12 Reactor Building Alternative 4b Pressure Retaining With Internal Rupture Panels and Expanded Volume

A3 EVALUATION OF REACTOR BUILDING PRESSURE RESPONSE AND RADIOLOGICAL RETENTION CAPABILITY

A3.1 CHARACTERIZATION OF FISSION PRODUCT TRANSPORT

A depressurized loss of forced circulation (DLOFC) transient consists of three phases:

1. Depressurization Phase
2. Heat-up (Expansion)Phase
3. Cool-down (Contraction) Phase

The depressurization phase occurs when the helium coolant within the HPB blows down into the RB and depressurizes the PHTS. During this blow-down the circulating activity that is suspended in the helium atmosphere during normal operation is released to the RB. In addition, some of the activity deposited on the PHTS surfaces in the form of settled dust or plateout activity can be re-suspended and also released to the RB. This re-suspension release increases with increasing break size because a faster blow-down will exert larger liftoff forces on the deposited particles.

The second phase is the heat-up phase where the core fuel heats up after the blow-down is complete and the PHTS is depressurized. During the heat-up phase, a portion of the core reaches elevated temperatures where additional fission products are released primarily from already damaged fuel particles. These radio-nuclides are released in the hot core volume. Due to the temperature increase the helium in the hot volume expands into the cold helium volume by which it is surrounded. By this mechanism some of the radio-nuclides released into the hot volume migrate into the cold volume. The expansion of the hot helium volume into the cold helium volume pushes some of the cold helium volume through the cold break into the RB. The released activity in the RB can settle on the RB surfaces, remain suspended in the RB atmosphere, be trapped on the filters in the RB exhaust or leak out of the RB bypassing the filters.

Other potential radionuclide transport mechanisms between the hot volume, the cold volume and the RB are (1) Buoyancy driven natural convection, and (2) Diffusion. Both of these processes are assumed in this analysis to be insignificant. The helium transport path is very complex, as shown in Figure 13. Helium enters through the cold inlet pipe into the top of the reactor vessel. From there it flows down between the vessel wall and the core barrel to the lower plenum where it turns around and flows up through the side reflector channels. At the top it turns again 180º to flow down through the core to the lower core plenum and out through the hot core outlet pipe to the heat exchangers. The hot core outlet pipe is a concentric double pipe with the hot helium flowing on the inside and cold helium on the outside. Because of this double pipe arrangement, there are no hot helium pipe breaks in the DBE spectrum.

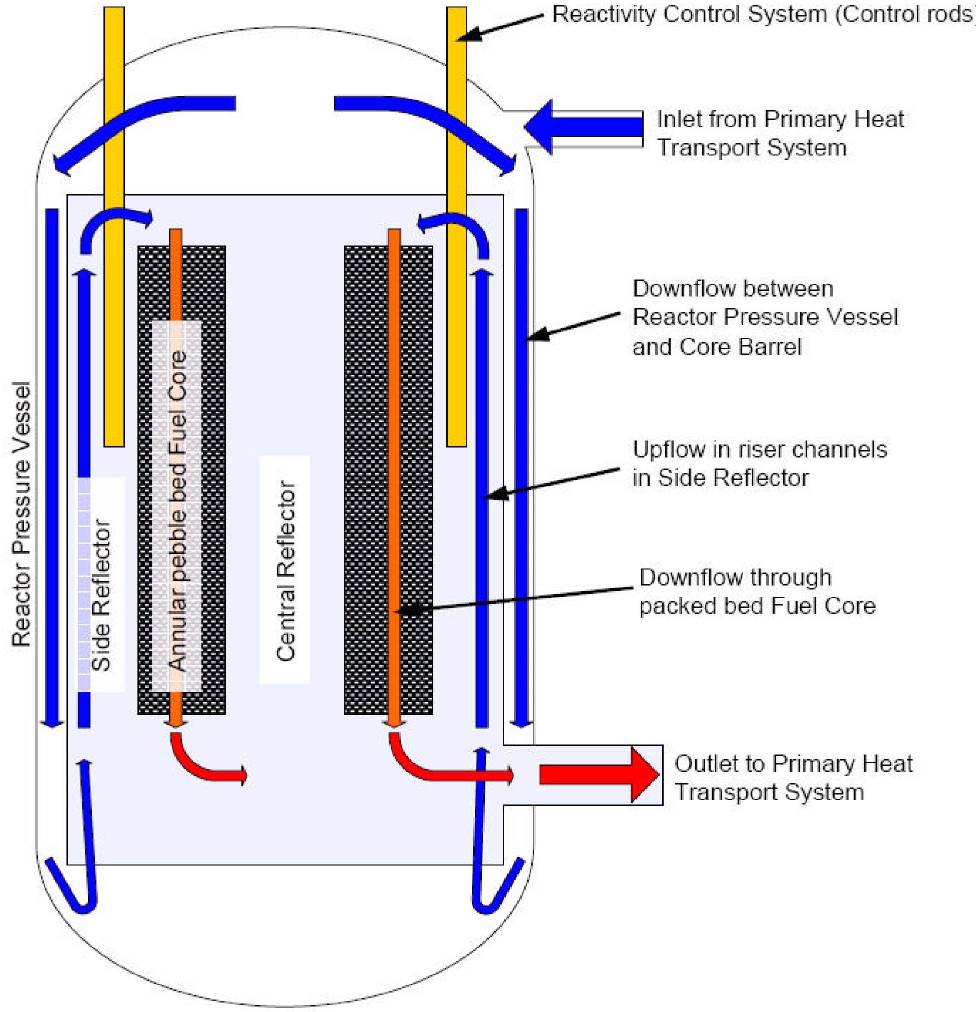

Figure 13 Helium Flow Path Through NGNP Reactor Pressure Vessel

The helium circulator has a check valve at the outlet that closes on loss of forced circulation and the pipe break for this analysis is assumed to be in the cold pipe between the circulator and the vessel. In order for a natural convection flow to develop, the hot helium, which wants to flow upward because of buoyancy, would have to reverse flow up through the core, down through the reflector and up again along the vessel, and out through the core inlet pipe to the break and out of the break. Because of the closed circulator check valve, the RB atmosphere would have to be drawn in through the same break and flow counter-current to the hot helium through the same flow path to the core in a streamline flow without mixing with the hot helium. This counter-current streamline flow is very unlikely to develop because there is no significant elevation difference between the core and the break location to provide a net buoyancy force to drive the flow. Even if there was a buoyancy driving force, mixing of the out-flowing hot helium and the in-flowing RB atmosphere would destroy the flow pattern and keep it from developing. Thirdly, the density of helium is 1/7th of the density of air at the same temperature which does not favor the development of buoyancy driven flow. Therefore, buoyancy driven flow has been assumed to have a negligible impact on the radionuclide transport to the RB.

The same complicated flow path also would make diffusion an ineffective means to transport radio-nuclides from the core to the RB and it also has been assumed to be negligible in this analysis. Double pipe breaks of the hot core outlet pipe and the cold shroud pipe and cold pipe breaks with a stuck open circulator exit check valve are BDBEs . For the PRA where BDBEs will be considered the possibility of a buoyancy driven flow may need to be revisited because in the double pipe break the flow pattern would be less complex and in the stuck open check valve case it would not need to be a counter-current flow.

The third phase is the contraction phase where the helium in the PHTS contracts because the fuel is cooling down and the helium temperature in the PHTS is decreasing. During this phase it is possible that air is drawn into the PHTS through the break, and if air comes in contact with hot graphite the graphite can oxidize to form carbon monoxide.

A3.2 TECHNICAL APPROACH

In order to assess the radionuclide retention and pressure capacity of the various Reactor Building (RB) alternatives for a DLOFC, a simplified 3 volume model representing the hot and cold parts of the PHTS and the RB vent volume (RBVV) was developed. The RBVV includes the free volume of the area that would be exposed to temperature and pressure loads following a depressurization event in the PHTS HPB. The model is an integrated model, capable of calculating the pressures and temperatures in each volume during the PHTS blow-down, heat-up and contraction phases of the transient, radionuclide transport between the volumes, radionuclide release from the RBVV and dose estimates at the site 425 m Exclusion Area Boundary (EAB). The heat-up and cool-down of the fuel and the time-dependent radionuclide release during the DLOFC are inputs to the model.

The PHTS is modeled with two volumes, one hot volume to account for the hotter portion of the PHTS, about 25%, and one cold volume to account for the cooler portion of the PHTS, about 75%, during steady state operation. The HPB breaks in this assessment represent breaks in the core inlet pipe (CIP) and are therefore simulated from the cold volume. The two volume model for the PHTS was adopted so that the effect of the time dependent transport of radio-nuclides released during the heat-up in the hot volume to the cold volume could be adequately reflected.

The third volume in the model is used to model the RBVV. The RBVV model is capable of simulating all of the proposed RB alternatives. Radio-nuclides are released from the RBVV directly to the environment. Deposition and holdup of the radio-nuclides that leak from the RBVV to the remaining RB volume are neglected.

The total air volume of the RB is approximately 100,000 m^3 per the NGNP Nuclear Heat Supply Building (NHSB) drawings, Refs. [18]-[21]. The RBVV volume is input to the model as a percentage of the total RB volume. Volumes for the RBVV of 10 to 20 percent of the total RB volume are typical and up to 100 percent of the total RB volume for cases with an added expansion volume. The volume of the 10 percent RBVV case is approximately equal to the volume of the IHX compartment and PRS relief shaft in the NGNP PCDR design.

Each volume in the model has a constant volume throughout the transient. Gas and radio-nuclides transferred between volumes are assumed to mix uniformly and instantly in the volume. The temperature of the hot PHTS volume is based on the heat-up of the fuel and any mixing of the hot and cold volumes during the transient. The temperatures of the cold PHTS and the RBVV are based solely on the mixing of the gases during the transient. Additional heat-up and/or cool-down of the cold volume and the RBVV due to conduction or convection effects are not accounted for in the model.

Insights from previous HTGR studies indicate that I-131 and Cs-137 are key radio-nuclides in determining off-site doses. The model calculates the transport of these two radio-nuclides from the PHTS to the RBVV and then to the environment. The total effective dose equivalent (TEDE) and the thyroid committed effective dose equivalent (CEDE) from inhalation are calculated for each isotope at the 425 m site boundary assuming a ground level release with no deposition accounted for in the environment. The total doses from including all radio-nuclides are then estimated by scaling the I-131 and Cs-137 doses.

The model accounts for the lift-off of plated-out activity and the re-suspension of dust activity in the PHTS during the initial blow-down but neglects the plate-out and resettling of nuclides in the PHTS following the blow-down. Settling in the RBVV is modeled, as is radioactive decay of the I-131 activity in all volumes and the environment. During the heat-up, radionuclides released from the fuel are added to the hot volume and transfer from the PHTS to the RBVV is based on thermal expansion. If the blow-down is complete, once the thermal contraction phase of the PHTS initiates, the release from the PHTS ceases as air ingress from the RBVV begins.

The model also addresses carbon monoxide (CO) formation as the result of air ingress during the cool-down phase of the transient. Two mols of CO are formed for each mol of oxygen that is transferred from the RBVV to the PHTS hot volume. The model only calculates the CO produced as a result of the air ingress from the RBVV into the PHTS due to thermal contraction. A separate analysis addresses air ingress for an assumed fuel inlet pipe rupture at the top of the reactor vessel.

The details of the analytical models and the model input are presented below.

1. All pressure differential flows are based on the isentropic choked and un-choked mass flow rate equations in the MHTGR PSID, Ref. [13]. The discharge coefficient is assumed to be 1.0 in all cases. The flow area for the PHTS leak is calculated by using the leak size as the diameter for a break with a circular area. The effective flow area for the leakage flow from the RBVV is calculated using the leakage rate per day and the corresponding pressure differential for the leakage rate. The effective flow areas for the re-closeable damper and filter damper are input.
2. At equal pressures, the mass flow is based on the ideal gas thermal expansion/contraction of the gas mixture.
3. Lift-off of plated out activity is based on the model in the MHTGR PSID, Ref. [13].
   a. 0.2% of deposited Cs-137 and I-131 lift-off for all shear force ratios less than 1.0
   b. 15% of deposited Cs-137 lift-off for all shear force ratios greater than 1.0 and less than 30.0
   c. 25% of deposited I-131 lift-off for all shear force ratios greater than 1.0 and less than 30.0
   d. Shear force ratios greater than 30.0 do not occur in the analyses.
4. The dust re-suspension model based on existing PBMR analyses [14].
   a. 0.02% of the Cs-137 dust is re-suspended for all shear force ratios less than 1.0
   b. 11% of the Cs-137 dust is re-suspended for all shear force ratios greater than 1.0 and less than 10.0
   c. 34% of the Cs-137 dust is re-suspended for all shear force ratios greater than 10.0
5. Settling of activity in the RBVV is based on the model in IAEA TECDOC-978, Ref. [15].
   a. Cs-137 deposition constant = 0.1hr^{-1}
   b. I-131 deposition constant = 0.3hr^{-1}
6. The TEDE calculation is based on the definition of TEDE in the NRC Reg. Guide 1.183, Ref. [16]. A simplified calculational methodology for the breathing rate, weather factor, and TEDE compared to that described in Ref [16] is used in the model and described in the items that follow. In the model, the TEDE is calculated based on the cumulative release for the 300 hour period following the initiation of the DLOFC.
   a. Dose conversion factors are based on EPA Federal Guidance Reports 11 and 12, Refs [22] and [23].
   b. A constant breathing rate of 2.3E-4 m^3/s based on the >24 hour breathing rate in NRC Reg. Guide 1.183, Ref. [16].
   c. A constant weather ? /Q factor of 2.3E-5s/m3 based on 10% of the ground release ? /Q at 425 m for the period of 24 to 96 hours in NRC Reg. Guide 1.4, Ref. [50].

This time period was selected because significant releases from the fuel do not start until >24 hours.

d. The total doses from all radio-nuclides are scaled based on "GT MHR Preliminary Safety Assessment Report, Table 4.4.3-3, Ref. [12].

   i. The total thyroid CEDE from all radio-nuclides is assumed to be a factor of 2 greater than the I-131 thyroid CEDE.

   ii. The total TEDE from all radio-nuclides is assumed to be a factor of 2.5 greater than the sum of the Cs-137 TEDE and the I-131 TEDE.

7. The calculation of the puff release from a gross RB failure for the pressure retaining RB alternative is based on the point in time when the activity in the RB results in the greatest dose. The product of the breathing rate and weather factor is assumed to be a factor of 18.5 higher than that used otherwise in the dose calculations to account for a sudden release. The factor of 18.5 is based on the breathing rates and 10% of the 0- to 8-hour weather factors in the NRC Reg. Guide 1.4, Ref. [50].

8. The time step during the calculation is 0.01 s for all times when the PHTS pressure is greater than the atmospheric pressure and 100 s for times when the PHTS pressure is equal to the atmospheric pressure.

9. The RBVV helium/air mixture properties are calculated at each time step.

10. The model inputs for each volume and for the environment are based on existing PBMR calculations for the 500 MWt NGNP design and are given in Table 8.

11. The time dependent average fuel temperature during the DLOFC is input to the model and shown in Figure 14. The source for this data is from existing PBMR calculations for the 500 MWt NGNP design. The average fuel temperature data was only provided for 0 to 72 hours. Based on results from existing calculations for a 500 MWt design with a different operating condition, the average fuel temperature is assumed to decrease linearly to about 750 ºC between 72 and 300 hours. The peak average fuel temperature of about 1100°C occurs at about 43 hours. During the transient, 28% of the PHTS hot volume is assumed to follow this temperature profile.

12. The characterization of the initial radiological inventories and the releases from the fuel during DLOFC are discussed in Sections A3.5 and A3.6, respectively.

13. The model input for the re-closeable damper, filter damper and filter is given in Section A3.4 for each alternative.

Table 8 NGNP Initial Conditions for Model

NGNP Plant Parameter	Value
Mass of PHTS cold volume, kg	2798.89
Volume of PHTS cold volume, m^3	376.43
Temperature of PHTS cold volume, °C	295.19
Mass of PHTS hot volume, kg	472.53
Volume of PHTS hot volume, m^3	117.23
Temperature of PHTS hot volume, °C	798.02
Pressure of RBVV, Pa	1.013×10^5
Temperature of RBVV, °C	20
Pressure of environment, Pa	1.013×10^5
Temperature of environment, °C	20
Steady state mass flow rate of PHTS, kg/s	160

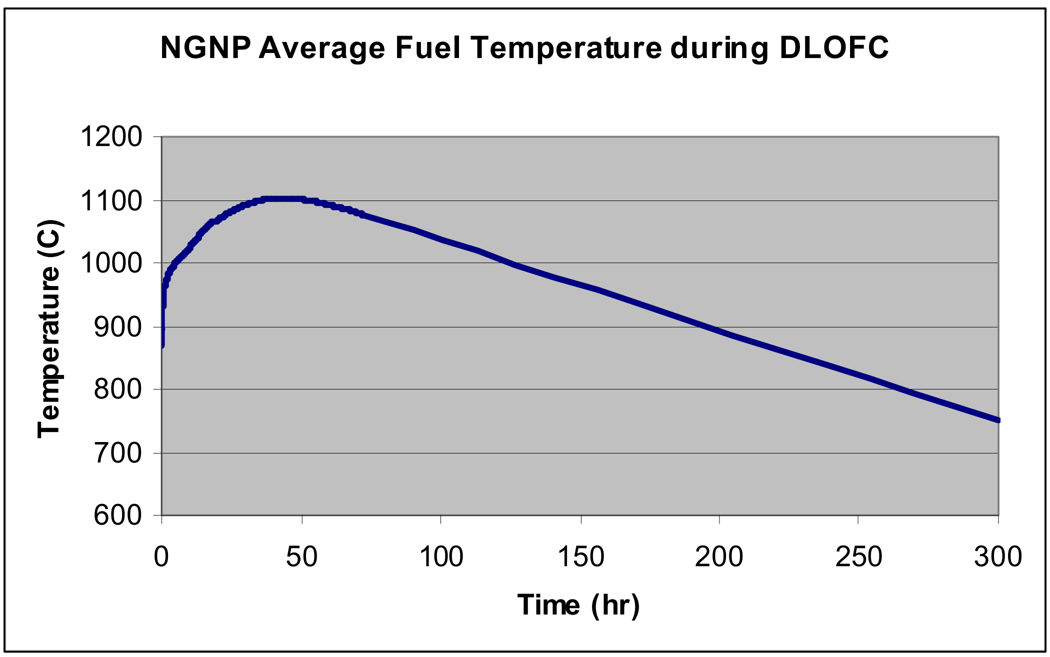

Figure 14 NGNP Average Fuel Temperature during DLOFC

A3.3 KEY ASSUMPTIONS

There is no integrated model available that is capable of modeling all of the aspects of the DLOFC including the blow-down, heat-up and contraction of the PHTS, the radionuclide release from the fuel, the transfer of radio-nuclides from the PHTS to the environment and the dose calculations for the NGNP design. Given the schedule and budget constraints for the project, a number of major assumptions were made in order to evaluate the RB alternatives.

1. Only 3 volumes are used to model the PHTS and RB.
2. Use of the DLOFC average fuel temperature to vary the hot volume temperature during the transient and the lack of additional heating or cooling of the PHTS cold volume or RBVV volume during the transient due to heat transport mechanisms other than mixing.
3. The radionuclide release from the fuel during the DLOFC is based on existing calculations for a 500 MWt PBMR design at different steady state operating conditions. The same radionuclide release profile is used for leaks of all sizes. These delayed releases are considered conservative for break sizes less than 100 mm due to the enhanced convection cooling of the core which lowers the peak temperatures and thereby lowers the release from the fuel during the heat-up.
4. The initial radionuclide inventories are taken from the NGNP Contamination Control Report, Ref. [10].
5. Use of simplified liftoff and dust re-suspension models based on the shear force ratio calculated by the model.
6. Mass transfer between the PHTS volume, the RBVV and the environment based on the pressure blow-down and on expansion/contraction during the heat-up and cool-down phase of the DLOFC.
7. No convection or buoyancy driven flow after depressurization.
    a. All DBE breaks in the PHTS outside the reactor vessel are breaks from the cold volume.
    b. The circulator discharge check valve closes at the end of the blow-down phase at the latest.
    c. The flow path for buoyancy driven convection from the core to the break location requires flow reversal and counter-current flow with multiple up-down paths.
    d. Flow patterns are complex and do not favor buoyancy driven convection.
8. Simplified dose calculational methodology as outlined in Section A3.1. All releases are treated as ground releases.

A3.4 BOUNDARY CONDITIONS

For the assessment of the RB pressure capacity, a design basis event involving a 1000 mm break (equivalent break size for a double-ended guillotine break (DEGB) of the SHTS HPB) is assumed due to the larger helium inventory of the SHTS relative to the PHTS. The lack of cooling for the SHTS based on the PCDR design introduces large uncertainties in the reliability of the hot gas pipe liner and insulation whose failure would directly lead to a major rupture of the SHTS HPB. Hence this break was classified as a design basis event, whereas breaks larger than 100 mm in the PHTS HPB are classified as beyond design basis events.

For the assessment of the RB radionuclide retention, the selected DBEs are limited to Depressurized Loss of Forced Cooling (DLOFC) cases for a range of break sizes in the core inlet pipe (CIP). Equivalent break sizes of 2, 3, 10, 100, 230, and 1000 mm are analyzed. The DEGB of the 710 mm CIP has an equivalent break size of 1000 mm.

Offsite doses for the DLOFC events at the 425 m site boundary are compared to the Top Level Regulatory Criteria of the Nuclear Regulatory Commission (NRC) and the Environmental Protection Agency (EPA) for assessing and regulating nuclear power plants in the US. These requirements are shown in Table 9. The TEDE is the sum of the CEDE from inhalation and the effective dose equivalent (EDE) from external exposure. The thyroid CEDE is the committed effective dose equivalent to the thyroid from inhalation. The TEDE and thyroid CEDE limits are based on the contributions from all radio-nuclides.

Table 9 Top Level Regulatory Criteria used to Evaluate Site Boundary Doses

Regulation and Reference	Application	Offsite Dose Limits
NRC 10CFR50.34 and NRC 10CFR50.67, Ref. [8]	Offsite dose limits during design basis events (DBEs)	TEDE < 25 rem/event
EPA Manual of Protective Action Guides and Protective Actions for Nuclear Incidents, Ref. [9]	Offsite dose limits during DBEs and beyond design basis events (BDBEs) at which sheltering is considered	TEDE < 1 rem/event Thyroid CEDE < 5 rem/event

A3.5 CHARACTERIZATION OF INITIAL RADIONUCLIDE INVENTORIES

The I-131 and Cs-137 source term inside the PHTS that is available for release at the beginning of the DLOFC is shown in Table 10 and is based on the NGNP Contamination Control Report, Ref. [10]. The plate-out and dust activities in Ref. [10] are based on the assumption of 60 years of continuous full power operation with an availability of 95 percent. No credit was taken for changes in the system conditions related to power operation and maintenance activities. For the model input, the Cs-137 plate-out and dust activities in each volume were reduced by 40 percent to be consistent with the total possible lifetime activity based on the Cs-137 steady state release rate given in Ref [10].

Table 10 Initial Release Source Terms for I-131 and Cs-137 Radio-nuclides

Source	Activity (Ci)
I-131 PHTS cold volume circulating	0.0
I-131 PHTS hot volume circulating	0.0
I-131 PHTS cold volume plateout	0.13
I-131 PHTS hot volume plateout	3.48
Cs-137 PHTS cold volume circulating	0.0
Cs-137 PHTS hot volume circulating	0.0
Cs-137 PHTS cold volume plateout	6.18
Cs-137 PHTS hot volume plateout	191
Cs-137 PHTS cold volume dust	4.67
Cs-137 PHTS hot volume dust	259

A3.6 MATRIX OF RADIOLOGICAL EVALUATION CASES

Using the three volume model described above, a matrix of cases was evaluated to determine the radiological consequences and the pressure response of the reactor building alternatives. For the DBEs a range of break sizes ranging form 2 mm to 100 mm equivalent diameter in the core inlet pipe were analyzed with reactor building vent volumes ranging from 10 to 100 percent of the nominal reactor building volume of 100,000 m^3. In all cases a total loss of forced circulation (LOFC) was assumed concurrent with the break in the Primary Heat Transport System (PHTS).

For the BDBE evaluation of margins and limiting reactor building pressures, two leak sizes were considered: a 230 mm equivalent break size, which corresponds to partial break in the 710 mm Core Inlet Pipe (CIP), and a 1000 mm equivalent break corresponding to a double ended guillotine break of the CIP. The same range of reactor building vent volumes (RBVV) from 10 to 100 percent was considered.

The reactor building alternatives considered were (1) no reactor building and (2) the reactor building alternatives 1a, 1b, 2, 3a, 3b, 4a and 4b described in Section A2.

In addition two special cases were considered: (1) Alternatives 4a (puff) and 4b (puff) are the same as Alternatives 4a and 4b with an assumed gross reactor building failure simulated by a puff release of the activity in the blow-down release from the reactor building at the worst point in time. This alternative was considered because in Alternative 4 the entire radionuclide inventory is retained in the RB in a pressurized state which would be available for a RB depressurization release in the event of a containment failure due to external causes (seismic, etc.) after the radionuclide release from the core has occurred. (2) A 1000 mm equivalent diameter SHTS break was considered as the limiting RB pressure response because the SHTS helium inventory is somewhat larger than the PHTS.

This case matrix analysis not only represents an evaluation of the previously described design alternatives, but the results can also be interpreted to represent a spectrum of scenarios in which the various RB design features (rupture panels, filters, pressure retaining features) are successful and unsuccessful in performing their respective safety functions such as would be developed in a full PRA. For example, the Alternative 1a results would also be applicable to Alternative 2 or 3a with failure of the filters and relief shaft isolation dampers.

For the analytical model the RB Alternative 2 shown in Figure 15 is used because it allows modeling of all the RB alternatives by varying the set-points of the re-closeable dampers and filter dampers. The different RB Alternatives are simulated with the inputs shown in Table 11.

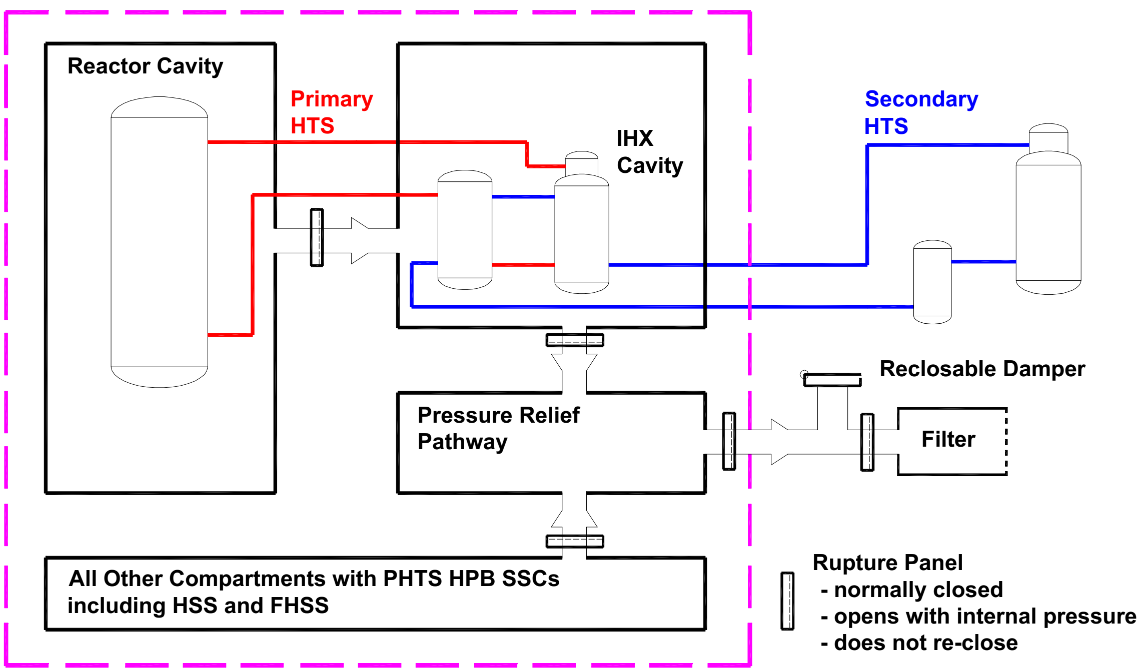

Figure 15 RB Alternative 2 Used as Basis for the RB Analytical Model

Table 11 Inputs Used for the RB Model to Simulate the Different RB Alternatives

RB Alter-native	Leak Rate [vol%/ day)	Leak Rate Pressure [bar-d]	RBVV [m^3]	Reclose Damper open pressure [bar-a]	Reclose Damper close pressure [bar-a]	Reclose Damper Flow area [m^2]	Filter Damper Open pressure [bar-a]	Filter damper Flow area [m^2]	Filter Decon. Factor Cs [%]	Filter Decon. Factor I [%]
1a	100	0.1	10,000 20,000	N/A	N/A	N/A	N/A	N/A	N/A	N/A
1b	100	0.1	10,000 20,000	1.113	N/A	N/A	N/A	N/A	N/A	N/A
2	50	0.1	10,000 20,000	1.213	1.113	3	[Note 1]	3	99	95
3a	50	0.1	10,000 20,000	N/A	N/A	3	1.213	3	99	95
3b	50	0.1	50,000 100,000	N/A	N/A	3	1.213	3	99	95
4a	1	10	10,000 20,000	N/A	N/A	N/A	N/A	N/A	N/A	N/A
4b	1	10	50,000 100,000	N/A	N/A	N/A	N/A	N/A	N/A	N/A

[Note 1] Filter damper opens when re-closable damper re-closes

A3.7 CHARACTERIZATION OF RELEASES FROM FUEL DURING DLOFC

The delayed radionuclide release from the fuel during the DLOFC is a function of the temperature during the transient. The calculations for the fuel release assume an atmospheric pressure at the beginning of the transient. This assumption is valid for a rapid depressurization cases. These delayed releases are considered conservative for break sizes less than 100 mm due to the enhanced convection cooling of the core which lowers the peak temperatures and thereby lowers the release from the fuel during the heat-up transient. As a result, releases from the fuel for breaks up to 100 mm are overstated in this analysis compared with a realistic assessment. However, this conservatism is acceptable for the purposes of this analysis: namely to compare alternative mitigation strategies for the reactor building design.

The temperature distribution profile during the transient is shown in Figure 16 and is based on existing calculations for a 500 MWt PBMR design at different steady state operating conditions than those for the NGNP [11]. Only a small amount of the fuel reaches temperatures above 1,700°C during a DLOFC transient, and only for a period of about 40 hours. The corresponding time dependent cumulative I-131 and Cs-137 releases from the fuel during this DLOFC are given in Figure 17 and are input for the model. The maximum I-131 release from the fuel during the transient is about 1,040 Ci and occurs at about 72 hours after which the decay of the source term is more than the quantity released. The maximum Cs-137 release from the fuel is 52 Ci, of which 90 percent is released by 72 hours. A variation from the PBMR DLOFC I-131 source term is that the model source term is used as the cumulative release and then decayed in the model. This assumption underestimates the I-131 maximum release somewhat but is considered reasonable in view of other model uncertainties.

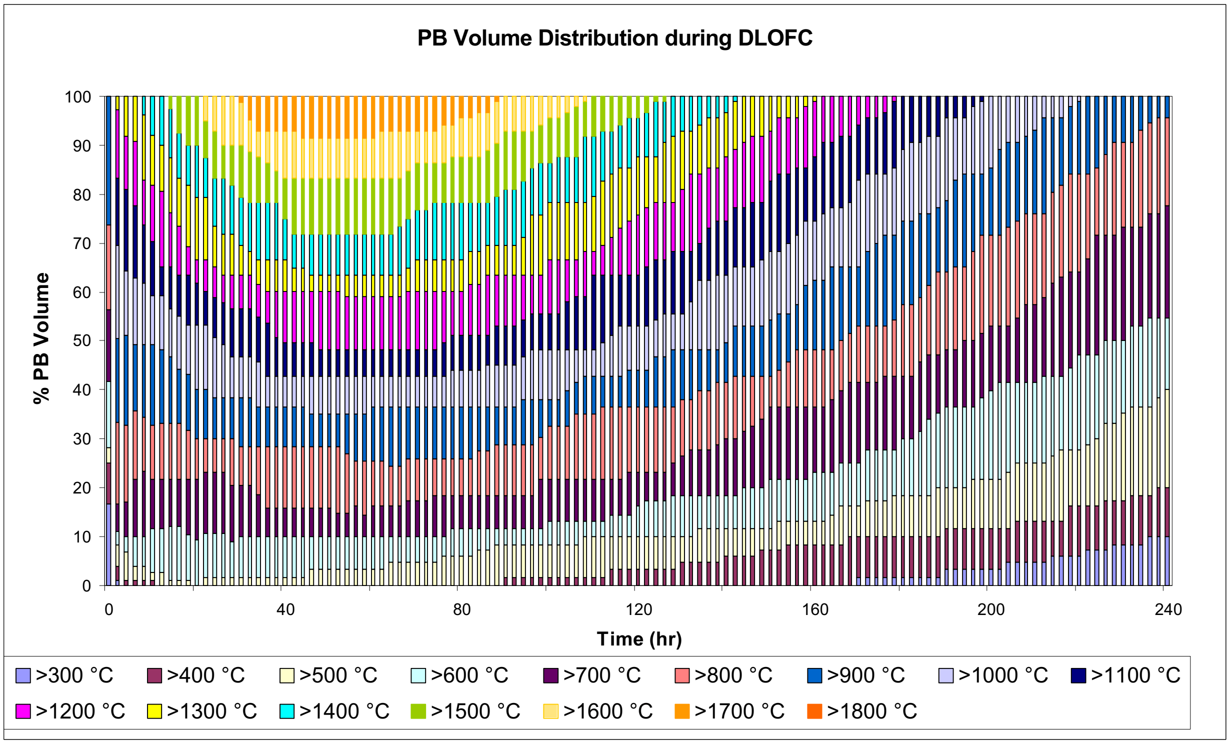

Figure 16 Core Fuel Time at Temperature Distribution during DLOFC

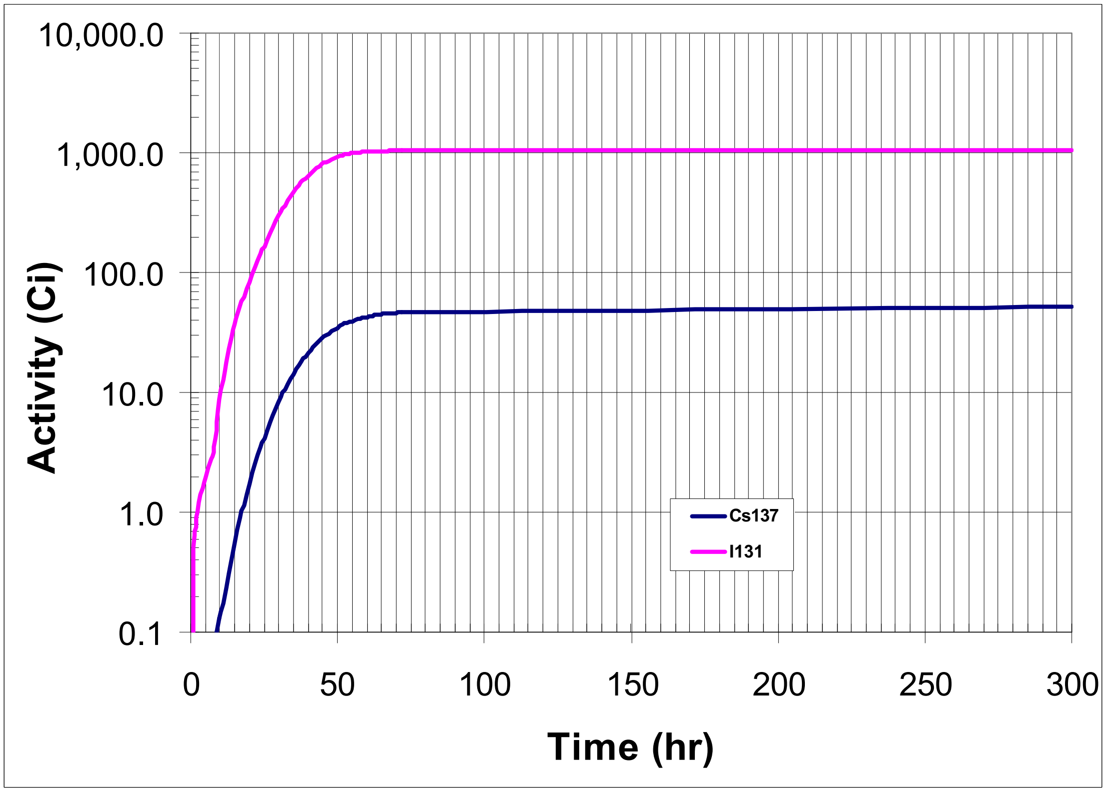

Figure 17 I-131 and Cs-137 Cumulative Activity Release from Fuel during DLOFC

A3.8 EVALUATION OF SITE BOUNDARY DOSES FOR THE NO REACTOR BUILDING CASE

The no RB case is an important case to consider because it sets the radionuclide mitigation required to be provided by any RB alternative. The release of radionuclide release in this case is directly from the HPB to the environment. The analysis of the break size range from 2 mm to 1000 mm showed that for all cases the 3 mm break size is the limiting break size. The reason for this is discussed in the following section.

The results for the no RB case areas are as follows:

TEDE at the site boundary (425 m):	0.4 rem
Thyroid dose at the site boundary (425 m):	10.0 rem

The no RB TEDE represents 1.6 percent of the 10CFR50.34 dose limit and 40 percent of the PAG dose limit for DBEs. No mitigation by a RB is needed to meet either TEDE limit.

The Thyroid dose is 2 times the PAG dose limit, and therefore some mitigation by a RB is required to meet the PAG dose limit. Note that meeting the PAG limit at the site boundary is an NGNP user requirement to avoid the need for an emergency planning zone beyond the site boundary and it is not a regulatory requirement.

A3.9 EVALUATION OF SITE BOUNDARY DOSES FOR REACTOR BUILDING ALTERNATIVES

As for the no RB case, the 3 mm break size is the dose limiting size for all RB alternatives. The reason for this is illustrated in Figure 18 for RB Alternative 1a. Figure 18 shows the TEDE for DLOFC as a function of break size. This figure is based on a selective, but limited, number of break size cases and is useful in explaining the trends in doses versus leak size; however, it may not represent the exact doses at all leak sizes in between those analyzed. The curve shows a minimum dose at a break size between 10 and 100 mm. At break sizes greater than 100 mm, the shear force ratio during the blow-down increase sufficiently that more of the radio-nuclides deposited in the PHTS are released from the PHTS and the dose increases accordingly. For break sizes smaller than about 10 mm, the PHTS is depressurizing during the time of significant release from the fuel providing a greater transport mechanism for the release from the PHTS to the RB. For the larger leak sizes the PHTS is depressurized before the heat-up release and the only mechanism available for release of the radio-nuclides from the PHTS to the RB is thermal expansion. For break sizes in the 3 mm to10 mm range, the smaller the break size, the more the blow-down overlaps the heat-up release and the more radio-nuclides are transported to the RB by the continuing blow-down. At break sizes less than 3 mm, the blow-down time is long enough that the decay of the I-131 while still inside the PHTS HPB becomes a significant factor in reducing the doses. Therefore, the 3 mm break size is the controlling break size for the DBE break size spectrum.

Table 12 shows the dose results for the 3 mm break size case for the RB alternatives with a 10% RB vent volume. The table gives the TEDE and the thyroid dose. The thyroid dose is more limiting when compared to the PAG limit than the TEDE for the 3 mm break size for all alternatives. The margin factor given in the last column is defined as the PAG thyroid dose limit (5,000 mrem) divided by the calculated dose. As shown in the table, the margin factor is significant, exceeding a factor of 10 in all cases.

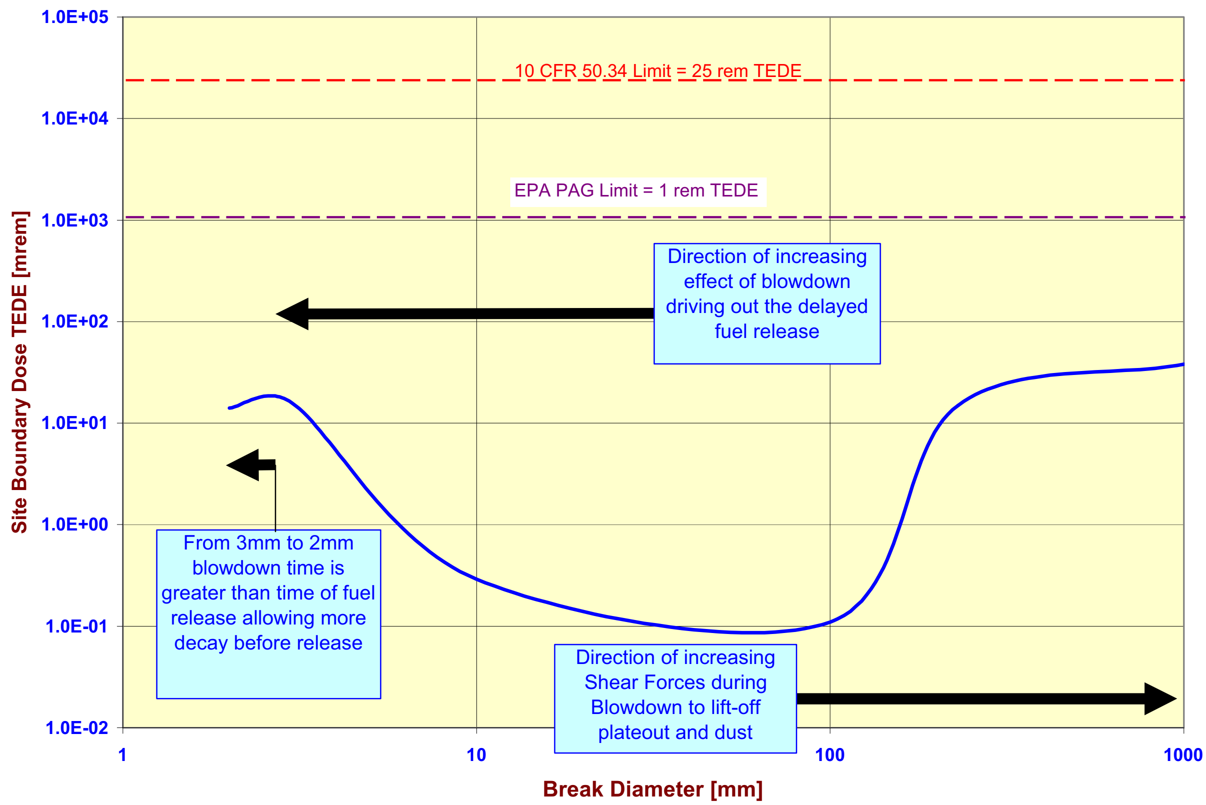

Figure 18 Dose Behavior vs. Break Size for RB Alternative 1a with 10% RB Vent Volume

Table 12 Dose Results for the RB Alternatives with a 10 % RB Vent Volume

RB Alternative	Leak Size [mm]	TEDE Dose [mrem]	Thyroid Dose [mrem]	Margin Factor
1a	3	16	400	12.5 (Th)
1b	3	16	400	12.5 (Th)
2	3	1	20	250 (Th)
3a	3	1	20	250 (Th)
4a	3	1.4	33	150 (Th)

Table 13 shows the dose results and margin factors again for the limiting 3 mm case for the large RB vent volume alternatives and for Alternatives 4a/b (puff) with the puff release simulating a gross RB failure for the pressure retaining RB alternative. For RB Alternatives 3a/3b, the margin factor increases in proportion to the RB vent volume. Since the maximum pressure during the transient is the same in these cases and controlled by the filter opening set-point, the amount of radionuclides transferred to the RB is the same for all the 3a/3b cases. The release from the RB decreases as the vent volume increases due to the lower concentration of the radio-nuclides in the larger volume. For Alternatives 4a/4b, the increased RB vent volume results only in a small increase of the margin factor. In Alternatives 4a/4b, there are competing effects as the vent volume increases. The release from the RB decreases as the vent volume increases as in the 3a/3b alternatives, but the amount of radio-nuclides that are transported to the RB is greater as the vent volume increases. The maximum RB pressure is different for all of the 4a/4b cases and decreases as the vent volume increases. Therefore, the largest vent volume case has the largest pressure decrease in the blow-down therefore it has the largest radionuclide transfer from the PHTS to the RB. For Alternative 4a/b (puff), the margin factor is less than one. For this alternative both the blow-down release and the heat-up release are stored within the RB and PHTS at an elevated pressure. When the gross RB failure for the pressure retaining RB alternative occurs and the RB blows down to atmospheric pressure, a significant portion of the stored radio-nuclides are released. This pressure driven release would not occur in the other RB alternatives because the RB is already at atmospheric pressure.

Table 13 Dose Results for the RB Alternatives with a large RB Vent Volumes

RB Alternative	RBVV [m^3]	Leak Size [mm]	TEDE Dose [mrem]	Thyroid Dose [mrem]	Margin Factor
3b	50,000	3	0.2	4	1250
4b	50,000	3	1.3	30	167
3b	100,000	3	0.1	2	2500
4b	100,000	3	1.1	25	200
4a (puff)	10,000	$3_{[note 1]}$	723	18,000	0.28
4b (puff)	100,000	$3_{[note 1]}$	943	23,000	0.22

[Note 1]: The accumulation of source term in the building prior to the gross building failure and release is from a 3 mm break of the PHTS

Table 14 lists more detailed results for RB Alternative 1a. In addition to the doses the table gives the maximum RB vent volume pressure, the time to depressurize and the time when the air ingress starts.

Table 14 Detailed Results for RB Alternative 1a

RB Alternative	RBVV [m^3]	Leak Size [mm]	TEDE Dose [mrem]	TEDE [mrem]	Max RBVV Pressure [bar]	Time to Depressurize to 1atm	Start of Air Ingress
1a	10,000	1000	7.7	38	4.3	16.2 sec	43 hr
1a	10,000	230	6	15	1.8	63 sec	43 hr
1a	10,000	100	0.1	0.1	1.125	334 sec	43 hr
1a	10,000	10	4.9	0.3	1.013	9.3 hr	43 hr
1a	10,000	3	398	16	1.013	102 hr	102 hr
1a	10,000	2	326	14	1.013	230 hr	230 hr
1a	20,000	1000	6.5	32	2.8	28.6 sec	43 hr
1a	20,000	100	0.07	0.08	1.115	334 sec	43 hr
1a	20,000	3	206	8.5	102	102 hr	102 hr

Some insights from the Alternative 1a analysis are as follows:

- The initial I-131 inventory for release in the blow-down is small, the major I-131 activity source is the activity release during heat-up.
- The maximum I-131 release from the fuel during the transient occurs at about 72 hours after which the decay of the source term is more than the quantity released. The maximum Cs-137 release occurs at 300 hrs of which 90% is released by 72 hours. 1% of the total release from the fuel is reached at 11 hours for I-131 and at 15 hours for Cs-137.
- For the 10 mm break size the depressurization is complete at 9.3 hours. For break sizes greater than 10 mm the depressurization is complete before a significant release from the fuel occurs.
- For break sizes greater than 10 mm the doses are low because the release from the fuel is stagnant in the hot volume without a driving force to the environment except for expansion until the cool-down starts.
- For break sizes greater than 10 mm the air ingress into the PHTS starts at 43 hours, the time of PHTS cool-down and is not perturbed by the blow-down. For break sizes less than 10 mm the blow-down extends into the cool-down phase and delays the air ingress to 102 hours for a 3 mm leak and to 230 hours for a 2 mm leak.
- For break sizes less than 10 mm the extended blow-down provides a significant convective mechanism to drive radio-nuclides out of the HPB during the time of the heat-up release from the fuel which is the major release phase.

- The major doses result for break sizes less than 10 mm because of the extended blow-down transport of radio-nuclides to the RB vent volume.
- The slow blow-down rates allow major retention of radio-nuclides in the RB vent volume.
- Doubling the size of the RB vent volume reduces the maximum dose by almost a factor of two.

Figure 19 and Figure 20 show the time dependent release of I-131 for Alternative 1a with a 10% RB vent volume for a 3 mm break and a 2 mm break, respectively. These figures show the release from the fuel, the activity retained in the HPB and in the RBVV and the activity released from the RB. Due to the long blow-down duration most of the activity released from the fuel is retained in the RB vent volume due to the lack of a thermo hydraulic driving force for its release.

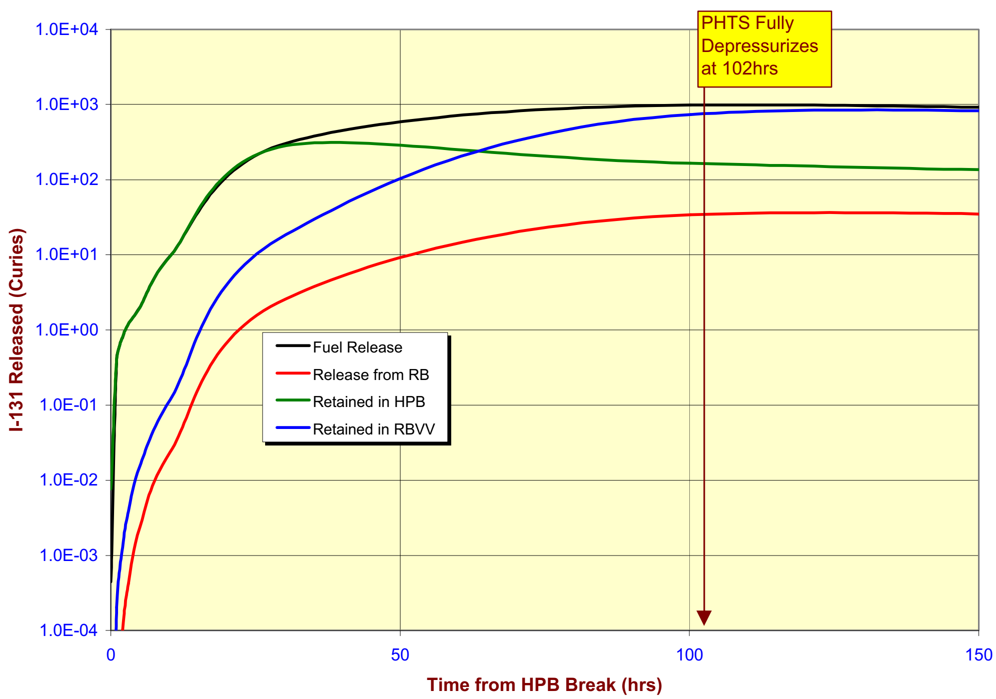

Figure 19 Time dependent release of I-131 for a 3 mm Break in Alternative 1a with a 10% RB Vent Volume

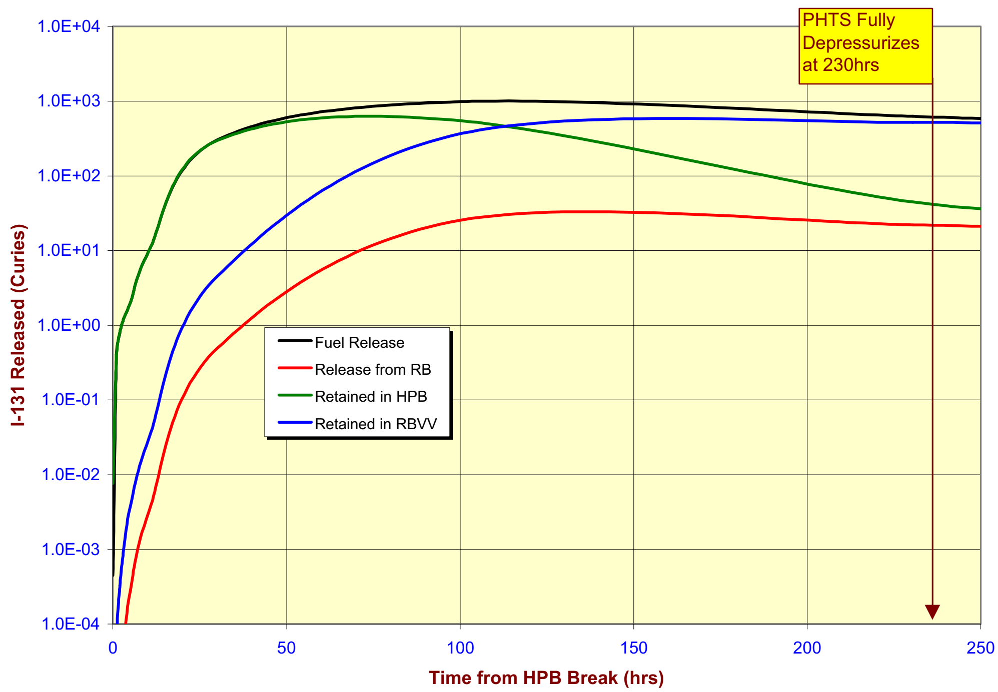

Figure 20 Time dependent release of I-131 for a 2 mm Break in Alternative 1a with a 10% RB Vent Volume

Figure 21 compares the I-131 release for the 2 mm Break and the 3 mm Break from the two previous figures, but on a linear scale. It shows that the I-131 release in the 3 mm break is significantly higher because the extended blow-down time in the 2 mm break allows more of the I-131 to decay.

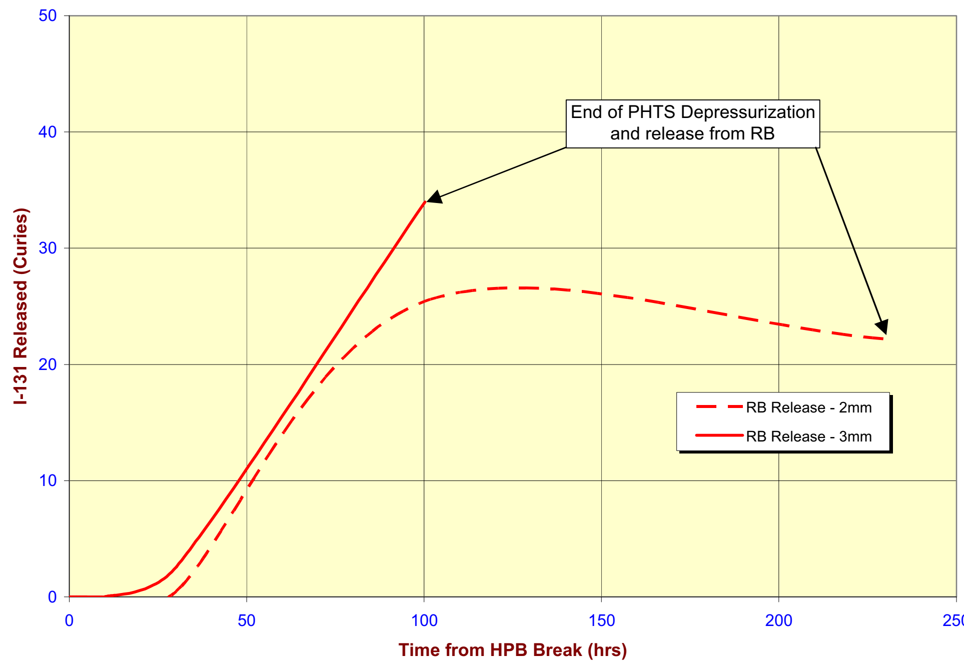

Figure 21 Comparison of the I-131 Release for the 2 mm Break and the 3 mm Break for Alternative 1a

Figure 22 compares the TEDE as a function of the break size for the different RB alternatives. All four RB alternatives clearly meet the TEDE limits. In the DBE domain, for breaks less than 100 mm, the Alternatives 2 and 3a are clearly superior to Alternatives 1a and 4a. In the BDBE domain, for breaks greater than 100 mm, the Alternatives 3a and 4a are clearly superior to Alternatives 1a and 2a. Overall, alternative 3a provides the best radiological protection over the entire break size spectrum for the alternatives shown in Fig 22. Not shown is Alternative 3b, which has a larger assumed RBVV and its radiological retention performance is the best of all the alternatives that were analyzed in this study.

Figure 23 compares the thyroid dose for the different RB alternatives with a 10 % RB vent volume for the 3 mm break size. As discussed earlier the no RB case exceeds the thyroid PAG limit by a factor of two. All RB design alternatives are well below the thyroid dose limit except for alternative 4a with the gross RB failure for the pressure retaining RB (Alternative 4c), which exceeds the PAG thyroid dose limit by a factor of four. Design Alternatives 2 and 3a provide the most margin to the PAG thyroid dose limit. The thyroid dose for Alternative 4c, representing gross RB failure for the pressure retaining RB, exceeds that for the no reactor building case because the release from alternative 4c is a prompt release compared to a slow release for the no RB case. The product of the breathing rate and weather factor is a factor of 18.5 greater for the prompt release.

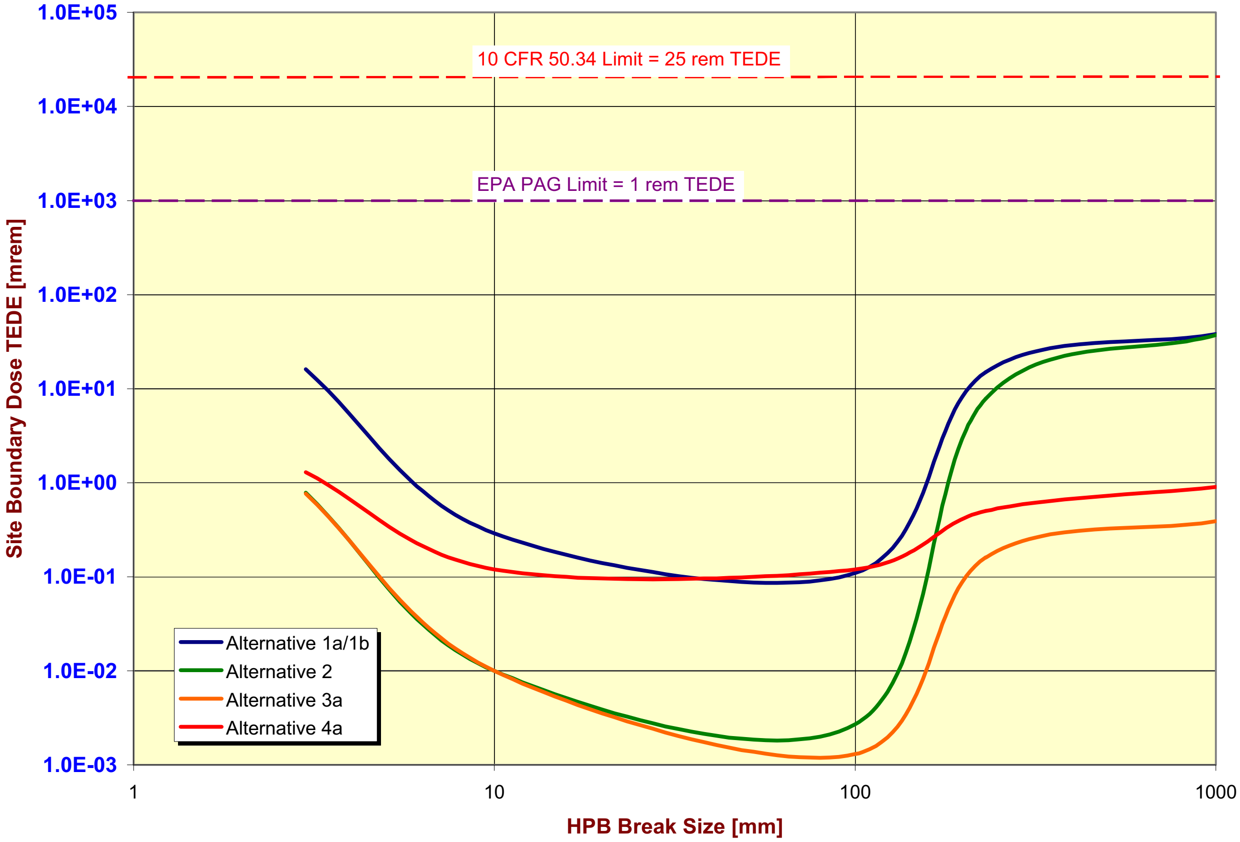

Figure 22 Dose vs. Break Size for the RB Alternatives with 10% RB Vent Volume

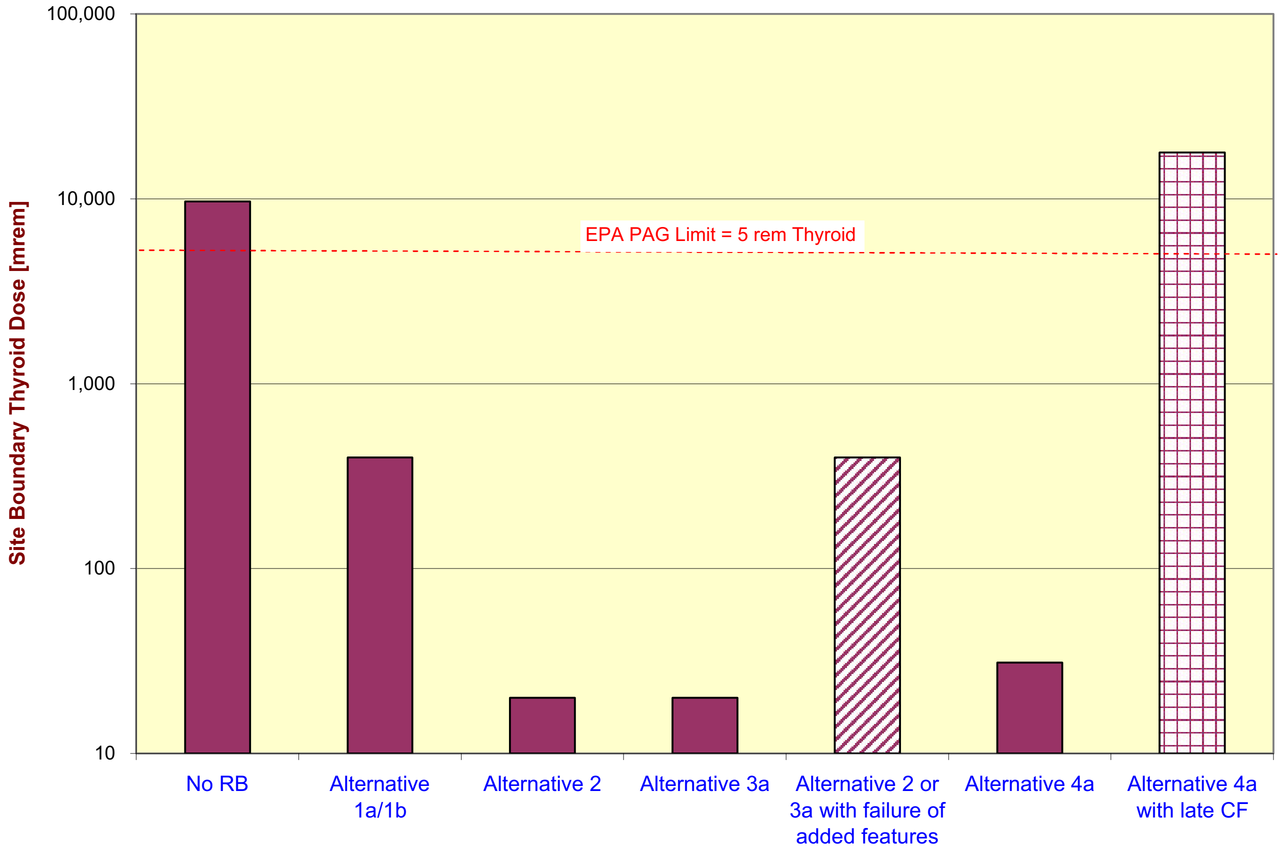

Figure 23 Thyroid Dose for the alternative Designs with a 10% RB Vent Volume for the 3 mm Break Size

A3.10 SUMMARY OF RESULTS FOR REACTOR BUILDING PRESSURE RESPONSE

For DBEs the RB vent volume peak pressure is controlled by the 100 mm leak and by the RB vent volume leak rate. Table 15 summarizes the results for the RB peak pressure for the different RB alternatives and RB vent volumes.

For RB alternatives 1a and 1b (leak rate 100 volume %/day at 0.1 bard), and for RB alternatives 3a and 3b (leak rate 50%/d at 0.1 bard) the RB vent volume peak pressure is below 0.125 bard.

For RB alternative 2 (leak rate 50%/d at 0.1 bard) the RB vent volume peak pressure is controlled by the opening pressure of the reclosable damper (0.2 bard).

For RB alternative 4a and 4 b (1%/d at 10 bar), the peak pressure ranges from 1.4 bar to 5.1 bar, decreasing with increasing RB vent volume. For this alternative there is a significant tradeoff between design pressure and volume.

For beyond design basis break sizes, the RB would experience high RB vent volume pressures, up to 4.3 bar for Configurations 1a, 2 and 3a with a 10% RB vent volume in a 1000 mm cold leg HPB break DLOFC. Therefore, with a low RB vent volume design pressure to meet the DBE needs, the RB would be likely to fail on overpressure for the BDBE break sizes and therefore the release might behave more like a No RB Configuration.

Table 15 Design Basis Peak Pressures for RB Design Alternatives

RB Alternative	RBVV m^3	Max RBVV Pressure [bar]
1a	10,000	1.125
1a	20,000	1.115
1b	10,000	1.138
1b	20,000	1.131
2	10,000	1.213[Note 1]
2	20,000	1.213[Note 1]
3a	10,000	1.125
3a	20,000	1.116
3b	50,000	1.097
3b	100,000	1.079
4a	10,000	5.1
4a	20,000	3.1
4b	50,000	1.9
4b	100,000	1.4

Notes: 1. Re-closeable damper opening pressure

The more limiting condition for the RBVV design pressure would be the maximum size break of the SHTS pipe of 1000 mm, a DBA. Due to the larger secondary side helium inventory, the peal pressures in the RBVV are higher than for the 1000 mm PHTS breaks. The SHTS has a total helium inventory mass of 3423 kg, which is 5% higher than the PHTS mass.

The peak pressures for the 1000 mm SHTS hot leg break were analyzed for Alternative 1a, but the 100 mm results are the same for all design alternatives, except for design alternative 4. The results for design alternative 1a are (as shown in Table 16):

Table 16 Peak Pressures for the 1000 mm SHTS Hot Leg Break

RBVV (% of RB Volume)	Peak RBVV Pressure (bar)	Time of Peak Pressure(s)
10	5.1	1.9
20	3.3	2.5
50	2.0	3.2
100	1.5	3.5

These peak pressures are about 25% higher than the corresponding 1000 mm PHTS breaks, and the peak pressure occurs at only a few seconds.

A3.11 AIR INGRESS

Air ingress is of interest because at elevated temperatures graphite can oxidize in the presence of air (oxygen) and form carbon monoxide, a flammable gas and the reaction is exothermic. The reaction of interest is:

$$C + \frac{1}{2} O_2 \Rightarrow CO + 110.5 \text{ kJ/mol-CO}$$

Two potential air ingress paths were considered in this study to determine whether air ingress could influence the RB requirements. (1) air ingress and CO formation during the cool-down/contraction phase of a cold leg HPB break DLOFC, and (2) air ingress through a fuel inlet pipe break at the top of the reactor vessel.

(1) Air ingress and CO formation during the cool-down phase of a cold leg HPB break DLOFC

Two air transport mechanisms were considered in this study: (a) buoyancy driven counter-current flow, and (b) Primary system contraction during the cool-down phase.

(a) Buoyancy Driven Counter-Current Flow

Air can theoretically be drawn into the reactor vessel by a buoyancy driven flow. Hot helium in the core wants to rise and push cold helium out through the break and draw the cold air-helium mixture from the RB into the vessel through the break. The closed circulator discharge check valve forces hot helium and cold air to flow in counter-current streamline flow past each other without mixing.

The flow path is very complex. For breaks downstream of the circulator discharge check valve, the helium has to flow from the hot core up to the core inlet plenum, then down through the outer reflector to the cold lower plenum, then up along the core barrel to the inlet plenum at the top, then out through the cold inlet pipe to the break before the check valve (see Figure 13). The cold helium-air mixture from the RB must be drawn in through the break and flow counter-current to the hot helium without mixing.

For breaks upstream of check valve, the hot helium has to flow against the buoyancy forces down from the hot core to the hot lower core plenum and out through the hot outlet pipe to the break between the heat exchangers or between the second heat exchanger and the circulator. The cold helium-air mixture must be drawn in through the break and flow counter-current to the hot core without mixing.

Streamlined counter-current flow without mixing in these complex geometries are very unlikely. The buoyancy force in hot helium is 1/7th of same temperature buoyancy in air because of the low He density, making the development of a buoyancy driven flow even more unlikely. This transport mechanism for air was assumed to be negligible in this study.

(b) Primary System Contraction during the Cool-down Phase

Air ingress does not start until 43 hours, and even longer for breaks < 10 mm. The contraction of the helium at atmospheric pressure in the PHTS draws the cold air-helium mixture from the RB through the break into the cold PHTS volume where it mixes with helium in the cold PHTS volume. Contraction of the hot PHTS volume draws the air-helium mixture in the cold volume into hot volume. Oxygen reaching the hot core can react with hot graphite. Calculations with the 3 volume model show that less than 2 mols of CO, a negligible amount, is produced by 300 hours due to contraction effects alone. Since cooling mechanisms other than gas mixing in the volumes is neglected in the model, the CO produced by contraction may be underestimated. Since the CO quantities are so small, it is estimated that the additional cooling will also yield small quantities.

(2) Air Ingress through a Fuel Inlet Pipe Break at Top of Reactor Vessel

There are three fuel tubes that rise in a vertical pipe from the core to the top of the reactor vessel (one fuel tube per 120 º core sector). The fuel tubes are 65 mm diameter with ribs to guide the 60 mm diameter fuel pebbles. A double ended guillotine break at the top of the vessel with a concurrent total loss of forced circulation is postulated. The blow-down of first the hot helium in the core and then the cold PHTS volume through the core will tend to cool the core down. The PHTS is depressurized at 1000 seconds. After the blow-down hot helium in the core will rise through the fuel tube and draw the cold air-helium mixture from the RB into the core. This is a reasonably optimal configuration for counter-current buoyancy driven flow to develop with hot gas at the bottom and cold gas at the top connected by a straight vertical tube.

Two equations must be satisfied:

1. Force Balance: Buoyancy force = Friction losses

2. Volume Flow balance: Volumetric flow rate of helium out = air in

The set of equations was solved by a double iteration on the friction factors for helium and for air under the following conservative assumptions:

- At the end of the blow-down it is assumed that the helium temperature in the core and the air-helium mixture in the RB are both at 1000 ºC.
- Assume pure air the in RB. In reality the RB atmosphere will be mostly helium because the air has been displaced in the blow-down. This assumption is conservative and maximizes the buoyancy.
- Assume 8 ribs of 2 mm height in the fuel tube.
- Assume all air entering the core reacts with graphite to form CO.

The analysis yielded the following results:

- The helium up-flow is laminar and occupies 86 percent of fuel tube flow area
- The air down-flow is turbulent and occupies 14 percent of fuel tube flow area
- The volumetric up-flow = the volumetric down-flow = 0.00151 m3/s
- The air mass flow = the air ingress rate = 4.18E-04 kg/s
- The air inflow would replace the helium in the core and upper and lower plenum in 10.3 hours
- The CO formation rate = 0.0283 Mol CO/s
- The reaction energy = 3.13 KW
- The decay heat at 3 hours = 5 MW

Conclusions:

- The air ingress through a ruptured fuel tube is small.
- The CO reaction energy from air ingress through a ruptured fuel tube is significantly larger than that from air ingress due to the contraction mechanism.
- The CO reaction energy from air ingress through a ruptured fuel tube is negligible compared to the decay heat.The analysis is conservative (pure air in RB, equal temperatures, all O2 reacts to form CO).

Therefore, both air ingress mechanisms yield a negligible rate of graphite oxidation and have no impact on the RB requirements.

A3.12 CONCLUSIONS AND RECOMMENDATIONS FOR FURTHER STUDY

The analysis of dose consequences for the RB alternatives provides the following results and conclusions:

- Some mitigation by a RB is required to meet the thyroid PAG dose limit.
- All evaluated RB alternatives provide ample margin to meet 10CFR50.34 and PAGs dose criteria evaluated in this study.
- Alternative 3a provides the most effective radiological retention for DLOFC across the break spectrum for the RBVV = 10 percent cases; the doses for Alternative 3b are somewhat lower than 3a due to the increased volume and provides the best radionuclide retention capability of all the alternatives evaluated in this study.
- Alternative 2 provides comparable radiological retention to Alternative 3a and 3b for DLOFC with design basis break sizes up to 100 mm break size.
- Alternatives 1a and 1b provide an indication of the consequences of events where design features such as filters and re-closable dampers added in Alternatives 2, 3a and 3b fail (BDBEs)
- Alternatives 4a and 4b provide less effective retention than Alternatives 3a and 3b across the full break spectrum and less than Alternative 2 across the DBE spectrum because the RB is pressurized to drive out radio-nuclides for the duration of dose calculation despite a relatively low leak rate
- Alternative 4c was defined to simulate the consequences of an assumed delayed failure of the RB (a BDBE assumption). The doses from Alternative 4c exceed the dose limits for all the cases analyzed including the case with no reactor building, because the RB failure results in a large prompt release from the pressurized RB, compared to the slower releases from a depressurized RB in the other alternatives.
- The limiting RB pressure is from the 1000 mm DBA break of the SHTS piping. The 10 percent RBVV design would need to withstand a peak pressure of 5.1 bar. The peak pressure is reduced to 3.3 bar, 2.2 bar, and 1.5 bar if the RBVV is increased to 20, 50, and 100 percent of the total RB volume, respectively. In the conceptual design, strategies to improve the reliability of the SHTS HPB should be considered to reduce the reactor building design pressure and/or volume of the vented area.
- The RB pressure capacity discussed in the previous bullet would also maintain RB integrity for the BDBEs and for design alternative 4.
- The chosen RB alternative should be optimized for minimum cost by considering the tradeoffs between the design pressure, the RB vent volume, the RB vent volume leak rate, the re-closeable damper opening pressure and the filter damper opening pressure.

The conclusions regarding the radiological retention capability of the evaluated RB alternatives are subject to limitations due to:

- The lack of design maturity and an associated full scope PRA model for the NGNP.
- The need to evaluate different HPB break locations and a fuller set of licensing basis events.
- The need to consider the impact of natural convection on core temperatures during small leaks (2 to 10 mm) for break locations and configurations not addressed in this study.
- The lack of a fully integrated mechanistic source term model.
- The need to consider the quantitative failure probabilities of various design features as well as the RB structural capability to withstand loads from a full set of licensing basis events.
- The lack of a full uncertainty analysis for the source term and consequence modeling.

These limitations should be addressed in the Conceptual and Preliminary Design Stages of the NGNP.

A detailed mechanistic code with integrated PHTS and RB models, in place of the simplified 3 volume model, will be required for the next phase of the analysis.

A4 INTEGRATED EVALUATION OF REACTOR BUILDING DESIGN CONCEPT ALTERNATIVES

A4.1 TECHNICAL APPROACH TO EVALUATION

The focus of this part of the study was on design strategies to perform pressure relief, radiological retention, and control of air-ingress functions. A total of 7 reactor building concepts was defined in Section A2 including those listed in Table 7. These concepts include variations of unfiltered vented concepts, filtered-vented concepts, and pressure retaining concepts. The parameters that were varied in the evaluation of these concepts included the leak rate for the area of the building that would be exposed to blow-down loads and fission product release pathways (Reactor Building vented area), use of rupture panels vs. open vented areas to manage the pressure relief function, use of filters to mitigate the delayed fuel release, blow-down phase of the release, or both, use of isolation devices to protect the normal operation HVAC system and to isolate the pressure relief shaft following blow-down, and alternative volumes of the vented area.

The above alternatives were evaluated using a qualitative multivariate decision analysis approach based on engineering judgment of the Westinghouse project team supported by radiological release calculations, design pressure calculations for the selected LBEs, as well as a preliminary assessment of the relative costs of the different alternatives. The evaluation criteria and weighting factors used in this evaluation are summarized in Table 17. For each criterion, the alternative ranked the best according to that criterion was given a score of 10 and the remaining alternatives given scores from 1 through 10 based on how they compared to the top ranking alternative for that criterion.

Table 17 Evaluation Criteria and Relative Weights

Criteria	Relative Weight
Normal Operation Requirements	10%
Investment Protection Requirement	5%
Safety Functional Requirements	35%
HPB leaks/breaks	20%
Seismic	5%
Hydrogen/process hazards	10%
Security /aircraft crash	10%
Capital and Operating Cost	25%
Licensability	15%
Total	**100%**

A4.2 EVALUATION OF REACTOR BUILDING ALTERNATIVES

The pressure transient and radionuclide retention capability of alternatives 1a through 4b have been evaluated using a judgment based scoring system. This process has followed these basic steps:

- Each alternative is presumed to be capable of meeting all non-safety functional requirements (mostly geometry and strength)
- Approximate pressure transients are calculated for several break sizes and locations in Section A3
- Engineered features (rupture panels, filters, etc) and possible changes in building geometry and strength requirements are estimated for each alternative
- Radionuclide retention by inherent and engineered features is estimated for each alternative, and site boundary doses are estimated in Section A3
- Cases are evaluated in Section A3 in which reactor building design features successfully perform their functions as well as cases to simulate the effects of various failure modes that may be identified in a future NGNP PRA
- Capital costs are estimated for each alternative with a focus on the relative costs of the features that differentiate the alternatives

The areas in which judgments are created are normal operating requirements, investment protection requirements, safety requirements, security and aircraft crash resistance requirements, costs, and licensability. Safety performance is judged by combining judgments on the alternative's pressure transient performance, its radionuclide retention performance, its seismic response, and its ability to withstand external hazard events related to the NGNP hydrogen process design. Cost performance is a combination of judgments related to capital cost and operating cost. Licensability is judged based on expert opinion regarding the expected difficulties that the concept may encounter in the licensing arena. An arbitrary score of 10 is applied to the alternative judged to best meet the requirements for each area. Other alternatives are given scores less than ten, based on the extent to which their ability to meet the requirements is less. The capabilities that are important when judging alternatives against requirements for normal operation are good operator access during operation and maintenance, minimum operator exposure to radiation and other hazards, and minimum normal operating release of radio-nuclides to the offsite public.

A4.2.1 Evaluation of Normal Operation Criterion

In the area of normal operation, the alternatives were judged as follows:

Alternative 1a	Adequate access and control of exposure; May not provide control of air activation products, and may require large HVAC flow.	Resulting score: 9
Alternative 1b	Adequate access and control of exposure; vent path normally isolated from environment improves control of air activation products and does not require large HVAC flow.	Resulting score: 10
Alternative 2	Same as 1b.	Resulting score: 10
Alternative 3a	Reduced leak rate requires airlocks and other impediments to operator access, otherwise, similar to 2.	Resulting score: 9
Alternative 3b	Same as 3a.	Resulting score: 8
Alternative 4a and 4b	Pressure retention requires additional features that impede operator access.	Resulting score: 3

A4.2.2 Evaluation of Investment Protection Criterion

In the area of investment protection, important attributes are low forced outage rate and durations, low risk of events that can cause damage to plant systems, structures, and components (SSCs), low risk of events that can result in significant plant outage time, and acceptably low risk of plant write-off. In this area, the alternatives were scored lower if they are judged to have an increase in forced outages and an increase in downtime. Those with more stringent leakage requirements were assumed to have an increase in outage durations and to take more time to recover from radionuclide releases.

Alternative 1a	Easy access, but more susceptible to external factors causing outages.	Resulting score: 9
Alternative 1b	Easy access, less susceptible to external factors causing outages.	Resulting score: 10
Alternative 2	Access hindered, but shorter recovery from events.	Resulting score: 8
Alternative 3a	Assumed to have an increase in outage durations (hindered accessibility) and to take more time to recover from radionuclide releases.	Resulting score: 6

Alternative 3b	Assumed to have an increase in outage durations (hindered accessibility) and to take more time to recover from radionuclide releases.	Resulting score: 6
Alternative 4a and 4b	Assumed to have an increase in outage durations (lack of accessibility) and to take more time to recover from radionuclide releases.	Resulting score: 4

A4.2.3 Evaluation of Safety Criterion

A4.2.3.1 Evaluation of Helium Pressure Boundary Break Response

When judging alternatives for their safety response to HPB breaks, the over-arching goal is to maintain the geometry of the reactor and its passive heat rejection system (RCCS). This requires that the building be design to reliably resist the pressure transient loads.

Alternative 1a	A vented design that results in survivable pressure transient with high reliability.	Resulting score: 10
Alternative 1b	Similar to 1a, but has the addition of rupture panels, which increase pressure transient slightly. This addition will help to isolate and protect HVAC.	Resulting score: 9
Alternative 2	Has higher pressure transients, but with additional cost, could be designed to resist the expected load therefore scored the same as 1b.	Resulting score: 9
Alternative 3a	Has higher pressure transients, but with additional cost, could be designed to resist the expected load therefore scored the same as 1b.	Resulting score: 9
Alternative 3b	Has higher pressure transients, but with additional cost, could be designed to resist the expected load therefore scored the same as 1b.	Resulting score: 9
Alternative 4a and 4b	Has higher pressure transients, but with additional cost, could be designed to resist the expected load therefore scored the same as 1b.	Resulting score: 9

A4.2.3.2 Evaluation of Radiological Retention Criterion

When judging alternatives for their response to radionuclide retention, calculations have been performed that provide an estimated building response over a spectrum of postulated HPB break events and these are summarized in Section A3. Scores for each alternative are correlated to offsite doses over these events.

Alternative 1a	Given lower scores because it indicates higher doses (although still below design goals).	Resulting score: 7
Alternative 1b	Given lower scores because it indicates higher doses (although still below design goals).	Resulting score: 7
Alternative 2	Given lower scores because it indicates higher doses (although still below design goals).	Resulting score: 8
Alternative 3a	Receive highest scores based on the results of the radiological release study. It should be noted that has possible event sequences in which the filters do not operate.	Resulting score: 10
Alternative 3b	Receive highest scores based on the results of the radiological release study. It should be noted that has possible event sequences in which the filters do not operate.	Resulting score: 10
Alternative 4a and 4b	Have a low score because some events (small helium breaks) result in the release and transport of both prompt and delayed fuel source terms driven by pent-up non-condensable helium retained in the low leakage building.	Resulting score: 5

A4.2.3.3 Evaluation of Seismic Capability

When judging alternatives for their response to seismic requirements, it is assumed that all alternatives are capable of resisting seismic forces. Design alternatives are less desirable if they require that heavy systems or components must be placed at high elevations in the building. All alternatives designed for seismic with margin and must respond with capability for beyond SSE events. Those with fewer filters and dampers are graded higher. Those with higher leak tightness more susceptible to seismic-induced cracks and leakage and are graded lower. This criterion is impacted by degree of embedment, which is discussed in later sections of the report.

Alternative 1a	Fewest filters and dampers.	Resulting score: 10
Alternative 1b	Fewest filters and dampers.	Resulting score: 10
Alternative 2	Similar to 1b but has filters and damper on top of building.	Resulting score: 9
Alternative 3a	Has larger filters mounted on top of building.	Resulting score: 8

Alternative 3b	Has larger filters mounted on top of building.	Resulting score: 8
Alternative 4a and 4b	Susceptible to seismic-induced cracks and leakage.	Resulting score: 6

A4.2.4 Evaluation of Process Hazards Criterion

When judging alternatives for their response to process hazard design goals, important criteria include the ability to protect safety related components and functions from shock or pressure waves, or toxicity of chemical hazards. This area is also impacted by the degree of building embedment. The alternatives were judged as follows:

Alternative 1a	External pressure loading due to hydrogen explosion could be greater than pressure transient loading; Open vent path may weaken resistance to hydrogen explosion.	Resulting score: 5
Alternative 1b	External pressure loading due to hydrogen explosion could be greater than pressure transient loading; Open vent path may weaken resistance to hydrogen explosion.	Resulting score: 5
Alternative 2	Superior to open alternatives 1a and 1b.	Resulting score: 8
Alternative 3a	More robust building means hazards not likely to control or impact design.	Resulting score: 9
Alternative 3b	More robust building means hazards not likely to control or impact design.	Resulting score: 9
Alternative 4a and 4b	Even more robust building, without vent or filters, offers the greatest resistance to hydrogen event hazards.	Resulting score: 10

A4.2.5 Evaluation of Physical Security and Aircraft Crash Criteria

When judging alternatives for their response to security threats or aircraft crash hazards, the goals are to prevent malevolent intervention from impacting plant safety or operation and to protect RB internals from being impacted by airplane crash, including fuel fires. This area may be impacted by the degree of embedment; however, the impact is probably the same for all alternatives.

Alternative 1a	Judged poorer than all the other alternatives because of the open vent path.	Resulting score: 5
Alternative 1b	Judged poorer than all the other alternatives because of the open vent path.	Resulting score: 5

Alternative 2	No directly open path for ingress of fluid or debris.	Resulting score: 10
Alternative 3a	No directly open path for ingress of fluid or debris.	Resulting score: 10
Alternative 3b	No directly open path for ingress of fluid or debris.	Resulting score: 10
Alternative 4a and 4b	No directly open path for ingress of fluid or debris.	Resulting score: 10

A4.2.6 Evaluation of Capital and Operating Costs

Alternatives are judged on estimated capital cost based on rough estimates from other project data. The effect due to operating cost is correlated with leak-tightness and number of active components. Costs are clearly impacted by the degree of embedment. In the area of cost, the alternatives were judged as follows:

Alternative 1a	Minimal features and loads.	Resulting score: 10
Alternative 1b	Modest additional features.	Resulting score: 9
Alternative 2	Additional damper and filter, not significant cost drivers.	Resulting score: 8
Alternative 3a	Significant cost penalty for reduced leak rate. Structural design probably not controlled by pressure loads.	Resulting score: 5
Alternative 3b	Similar to 3a, but with additional cost for expansion volume.	Resulting score: 4
Alternative 4a	Large cost increment for low leak rate, pressure retaining capability.	Resulting score: 2
Alternative 4b	Similar to 4a, but with modest reduction in cost due to lower pressure load; additional cost for expansion volume.	Resulting score: 1

A4.2.7 Evaluation of Licensability Criterion

When judging alternatives for their response to licensability goals it is important that they meet statutory limits for LBEs at site boundary. It is a project design goal that the NGNP not require and evacuation drills. This goal means that the designs must also meet EPA protective action guidelines (PAGs) for LBEs for no evacuation at site boundary. This assessment is highly subjective, and is linked to the margins of the safety criteria and the ability to credibly present and defend a non-LWR safety design approach. Evaluation of the licensability of alternatives is as follows:

Alternative 1a	Given a low licensability due to the difficulty in proving that design features such as dampers and filters are not cost effective and that the design is consistent with defense-in-depth principles.	Resulting score: 4
Alternative 1b	Given a low licensability due to the difficulty in proving that design features such as dampers and filters are not cost effective and that the design is consistent with defense-in-depth principles.	Resulting score: 4
Alternative 2	Vented options are ranked high based on their dose margins.	Resulting score: 9
Alternative 3a	Vented options are ranked high based on their dose margins.	Resulting score: 10
Alternative 3b	Vented options are ranked high based on their dose margins.	Resulting score: 10
Alternative 4a and 4b	Have the potential for the highest consequence sequences.	Resulting score: 6

A4.2.8 Summary of Evaluation Results

The scoring of the Reactor Building alternatives against each individual criterion, without the weighting factors, is shown in Table 18, and the combined effects of the scores and the weighting factors is shown in Table 19. As seen in the results of the integrated evaluation, all of the vented options scored higher than either of the two pressure retaining options when all of the factors were considered in an integrated fashion. Alternative 2 followed by Alternative 3a were the highest ranking alternatives and should be considered for further evaluation in the Conceptual Design phase of the NGNP. Because the volume of the vented area of the Reactor Building is a parameter that must be optimized against many factors not addressed in this evaluation, Alternative 3b is also a good candidate for further evaluation. After understanding the importance of the assumed reactor building leak rates in the radiological evaluation in Section A3, the use of additional features such as reclosable dampers should also be considered across all alternates during conceptual design.

It is noted that all of the options considered in this evaluation could be applied to any level of reactor embedment. It is also expected that with full embedment, a somewhat lower leak rate of the Reactor Building vent area might be achieved, however lower leak rates could also be achieved via engineered features on the building such as seals and special doors without any special level of embedment.

Table 18 Evaluation Scores for Individual Criteria

Alternative	1a	1b	2	3a	3b	4a	4b
Normal Operating Requirements	9	10	10	9	8	3	3
Investment Protection Requirements	9	10	8	6	6	4	4
Safety Requirements							
HPB Breaks - Pressure Response	10	9	9	9	9	9	9
HPB Breaks - Dose Response	7	7	8	10	10	5	5
Seismic Response	10	10	9	8	8	6	6
Hydrogen/Process Hazard Response	5	5	8	9	9	10	10
Security / Aircraft Crash Response	5	5	10	10	10	10	10
Capital and Operating Cost	10	9	8	5	4	2	1
Licensability	4	4	9	10	10	6	6

Table 19 Total Evaluation Scores for Reactor Building Alternatives

Alternative	weight	1a	1b	2	3a	3b	4a	4b
Normal Operating Requirements	10	90	100	100	90	80	30	30
Investment Protection Requirements	5	45	50	40	30	30	20	20
Safety Requirements								
HPB Breaks - Pressure Response	12	120	108	108	108	108	108	108
HPB Breaks - Dose Response	8	56	56	64	80	80	40	40
Seismic Response	5	50	50	45	40	40	30	30
Hydrogen/Process Hazard Response	10	50	50	80	90	90	100	100
Security / Aircraft Crash Response	10	50	50	100	100	100	100	100
Capital and Operating Cost	25	250	225	200	125	100	50	25
Licensability	15	60	60	135	150	150	90	90
TOTAL	**100**	**771**	**749**	**872**	**813**	**778**	**568**	**543**

A4.2.9 Evaluation Summary of Alternative Reactor Building Concepts

In general the vented options that were considered (1a, 1b, 2, 3a, and 3b) were found to be superior to the pressure retaining options (4a and 4b) based on the following considerations:

- Greater compatibility with a non-condensable and inert primary coolant
- Venting of the primary coolant inventory to atmosphere with or without filtration eliminates a driving force for subsequent fission product transport of the delayed fuel release source term
- When used with filtration (2, 3a, and 3b) provides more effective retention of radio-nuclides for the design basis event spectrum up to 100 mm. Alternatives 3a and 3b provide superior retention for beyond design basis event break sizes up to 1,000 mm as well
- Lower capital and operating costs
- Easier and less costly to engineer interfaces with RCCS, SHTS, FHSS, HSS, and other NHSS and auxiliary systems

The highest rating of integrated evaluation for alternatives examined was Alt 2 (Partially filtered and vented with rupture panels) followed by Alt 3a (Fully filter and vented with rupture panels).

- Both alternatives (2 and 3a) provide superior radionuclide retention capability for design basis HPB breaks with DLOFC than the pressure retaining alternatives (4a and 4b); Alt 3b closely followed by 3a is superior to all evaluated alternatives across the entire HPB break spectrum including AOOs, DBEs, and BDBEs.
- Alts 2, 3a, and 3b are expected to have greater licensability than either of the open vented options (1a and 1b) due to their superior capability to mitigate releases and air ingress.
- Another alternative for future study is a vented building with a passive re-closable damper without a filter. This is expected to have delayed fuel release retention capabilities approaching that of Alt 2 due to the capability to achieve a lower leak rate.
- Results of the radionuclide retention study show that all the evaluated alternatives provide sufficient margins to offsite dose limits based on inherent and passive safety characteristics of the PBMR NGNP.
- Added engineered features such as filters and re-closable dampers add additional margins.
- This study confirms that radiological retention is not a required safety function but rather a supportive safety function for the NGNP reactor building according to how these terms are defined in the NHNP risk informed and performance based licensing approach.

Having said that it is noted that the required safety functions of the reactor building that involve the structural protection of the reactor and its inherent and passive safety characteristics also serve to maintain the fundamental safety function of controlling radionuclide releases. It supports this function primarily by keeping the radio-nuclides inside the coated particle fuel and PHTS HPB and secondarily by retaining radio-nuclides that may be released from the fuel and HPB.

- In order to support the PBMR NGNP capabilities for defense-in-depth, it is recommended that a design goal be set for a radiological retention capability of a factor of 10 reduction in releases from the RB relative to that released from the HPB for I-131 and Cs-137 for DBE and BDBE HPB breaks.

A5 OPEN ISSUES AND ADDITIONAL ENGINEERING STUDIES

Conclusions regarding radiological retention capability of evaluated RB options are subject to limitations due to:

- Lack of design details and associated full scope PRA model
- Need to evaluate different HPB break locations and a fuller set of licensing basis events
- Need to consider the impact of natural convection on core temperatures during small leaks (2-10 mm)
- Lack of a fully integrated mechanistic source term model
- Need to consider the failure probabilities of various design features as well as RB structural capability to withstand loads from a full set of licensing basis events
- Lack of a full uncertainty analysis in the source term and consequence modeling

These limitations should be addressed in the Conceptual and Preliminary Design Stages of the NGNP supported by the NGNP PRA

Key challenges were identified for reactor building design that need to be addressed in the conceptual design stage. These challenges include:

- Need for optimization of RV vented volume dimensions vs performance and cost
- Unknowns regarding needed protection against hydrogen process hazards
- Unknowns regarding needed protection against physical security threats
- Systems interactions issues associated with SHTS piping penetration RB walls
- Key requirement for the RB is to provide physical separation of the NHSS from events and hazards associated with the HPS, PCS, and BOP facilities
- SHTS piping provides structural linkage between RB and adjacent buildings

There is a need to investigate further the possible “systems interactions” involving:

- Faults in HPS or PCS propagating into RB
- External events for which RB is protected but other buildings are not causing adverse interactions
- Need to provide high confidence of no adverse interactions for design basis events

- May lead to identification multiple large HPB breaks for BDBEs during the Conceptual Design PRA
- Key challenge in the next phase of the design
- Issue may complicate the approach to embedment

Pending a more thorough design iteration, DDNs will be formulated leading to potential technology development primarily in the fuel, reactor, and HPB.

PART B: EVALUATION OF REACTOR EMBEDMENT

This portion of the study develops requirements and criteria for determining the degree of embedment of the reactor. This study of the PBMR reactor considers the interaction among factors that influence the depth of the embedment. These factors include cost, design basis threats, seismic effects, and hazards resistance. The results of this study will be used to characterize the interactions of these factors on embedment depths for commercial application of this technology. References from relevant sections of the Electric Power Research Institute (EPRI) Advanced Light Water Reactor Utility Requirements Document are reviewed for applicability in this study description.

In addition, this study evaluates the potential differences embedment could make for transferring heat to the environment including heat flow through walls to ground or to the air during severe beyond accident conditions.

This study assesses contributing features with respect to siting of the NGNP at INL and a commercial facility sited at other potential locations within the US under the jurisdiction of the US NRC.

This portion of the study is conducted as indicated in the following flow chart (Figure 24).

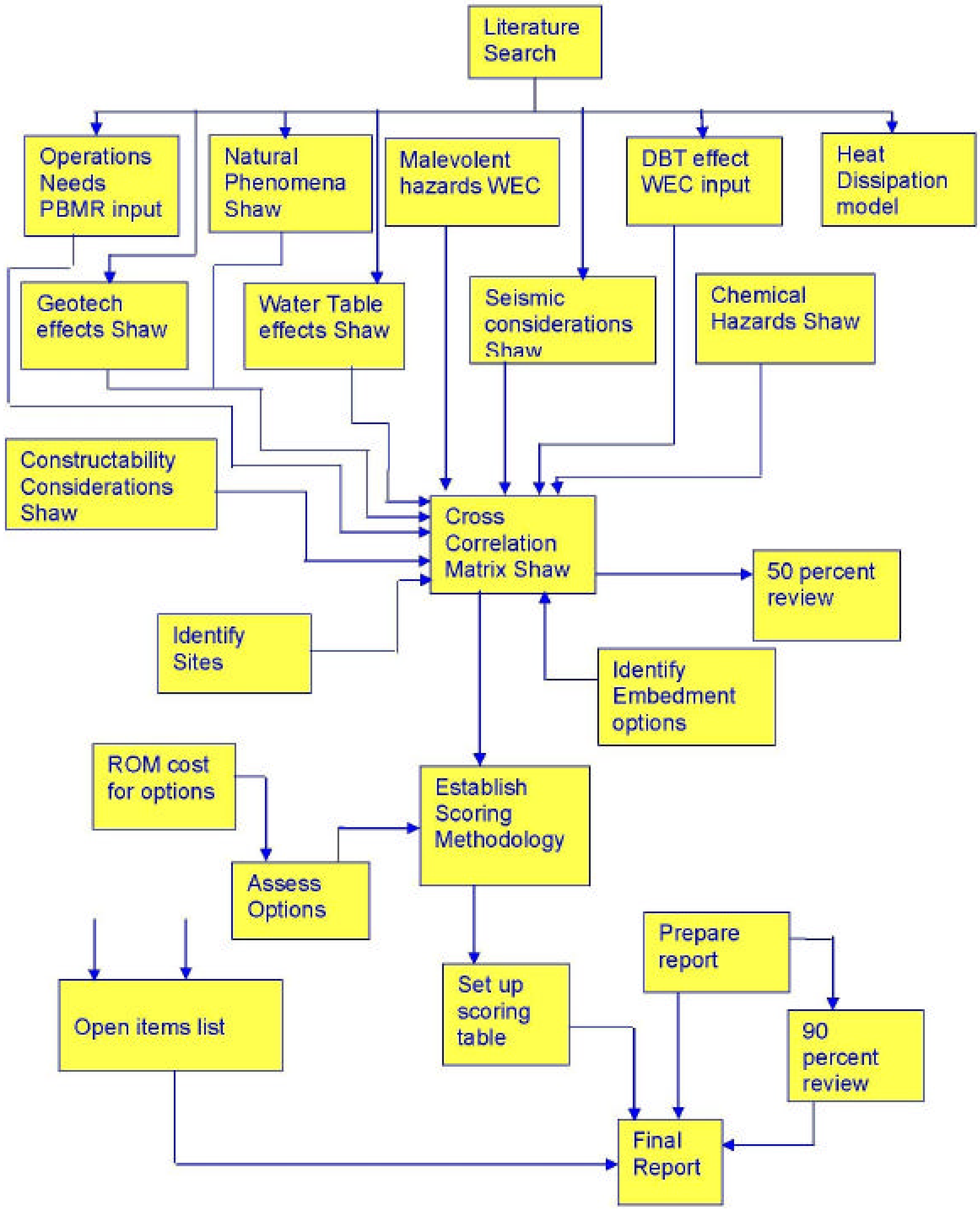

Figure 24 Flow Chart of Part B: Evaluation of Reactor Embedment

B1 LIST OF ISSUES RELEVANT TO REACTOR EMBEDMENT

The following factors are considered in this section of the report. Reference information provided in Ref. [8] has been reviewed along with other regulatory sources and discussed as required.

- Operational Needs including equipment layout
- Heat Dissipation to Environment
- Water Table Effects
- Geotechnical Constraints and Foundation Performance
- Construction Considerations
- Cost Consideration
- Malevolent Hazards
- Natural Phenomenon Hazards
- Natural Geological Hazards
- Chemical Releases, Explosions, and Manmade Hazards

B1.1 OPERATIONAL NEEDS

The Reactor and the reactor building layout should consider the following with respect to embedment for safe reliable and efficient operation and maintenance of the Nuclear Heat Supply Facility.

- Maintenance access must be provided to all major equipment.
- Safe personnel access and egress paths (elevators, corridors and stairwells) must be provided.
- Enable placement of system components as needed relative to the elevation of the reactor itself, to meet thermal hydraulic and other operational needs.
- Protection must be provided of safety related components from external hazards and hazards due to internal SSC failures.
- The PBMR design does not require routine access to the reactor for refueling activities. (The need to access the reactor head for refueling drives other HTGR configurations to favor full embedment.)
- The Pressure Relief System (PRS) is anticipated to discharge from a vent stack located on the top of the building. This need could benefit with a building roof elevated above

ground. It is not anticipated that an elevated release point requiring a tall stack will be needed to meet off site dose limits. Analysis in the section A assumes ground level releases for all cases. Therefore there is no particular benefit with a taller building.

- Optimized travel distances for routine operator access and egress. Travel distances can have an impact on life cycle costs as discussed in section B1.6.

The following information has been provided by the PBMR design team and reflects current work in progress in development of the reactor building layout for a single module design for a Brayton Cycle Electrical Machine (the DPP) and is provided for discussion purposes only. It is intended that during the conceptual design phase of the NGNP a reactor building layout will be developed using the functional requirements identified in this study report as a basis (see part A). The degree of embedment does not significantly affect plant operations. The Reactor building layout currently being considered includes access to the Reactor Building from the Auxiliary Building. This is either via a main equipment hoist lobby or at the lowest level in the building. However, the access to the reactor building via the main equipment elevator lobby can be at any level. Each level has a lobby and direct access to the elevator. It is expected the equipment maintenance elevators and hoist spaces to transport major components to these access points can be readily configured regardless of level of embedment.

Another key feature that leads to the conclusion that the level of embedment is not significant to the design for operations is that the refueling system uses continuous circulation of the fuel spheres and does not require routine overhead access to the reactor as is the case with other HTGR design using prismatic block core structures. This is the main reason why other HTGR designs have gone to the embedded configuration.

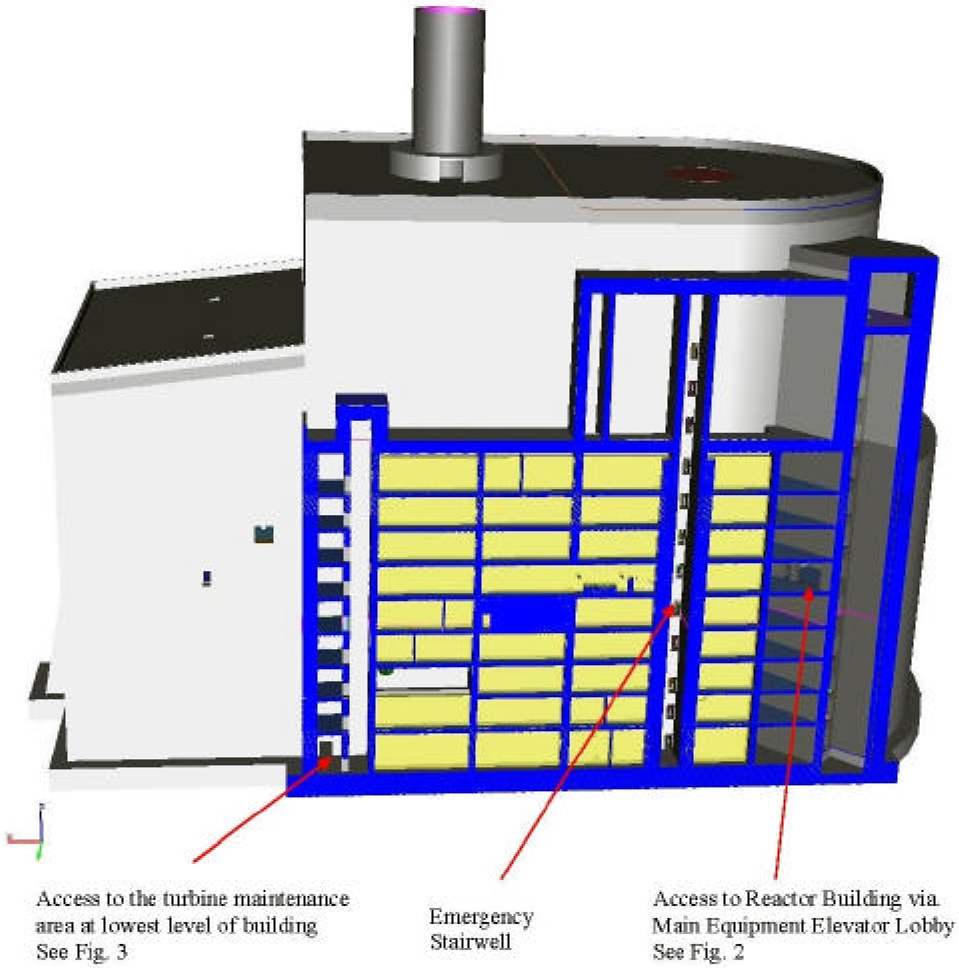

Figure 25 Access to Reactor Building

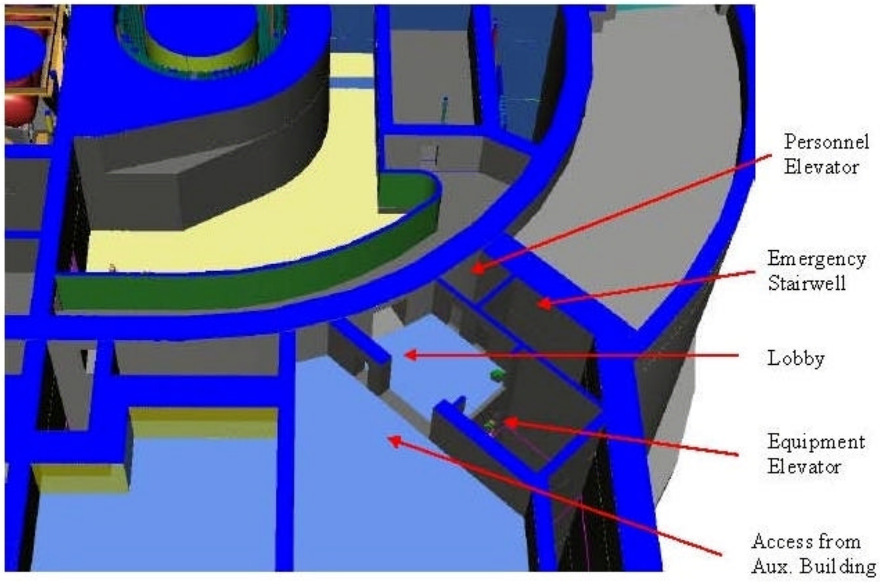

Figure 26 Access to Reactor Building at Main Equipment Elevator

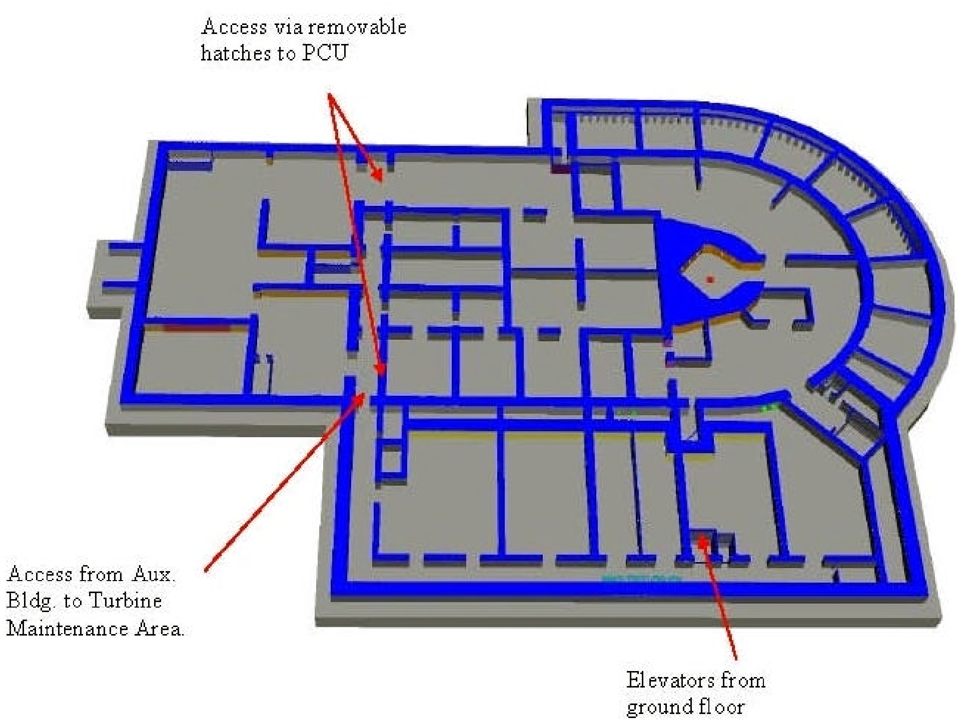

Figure 27 Access to Reactor Building at Lowest Level in the Building

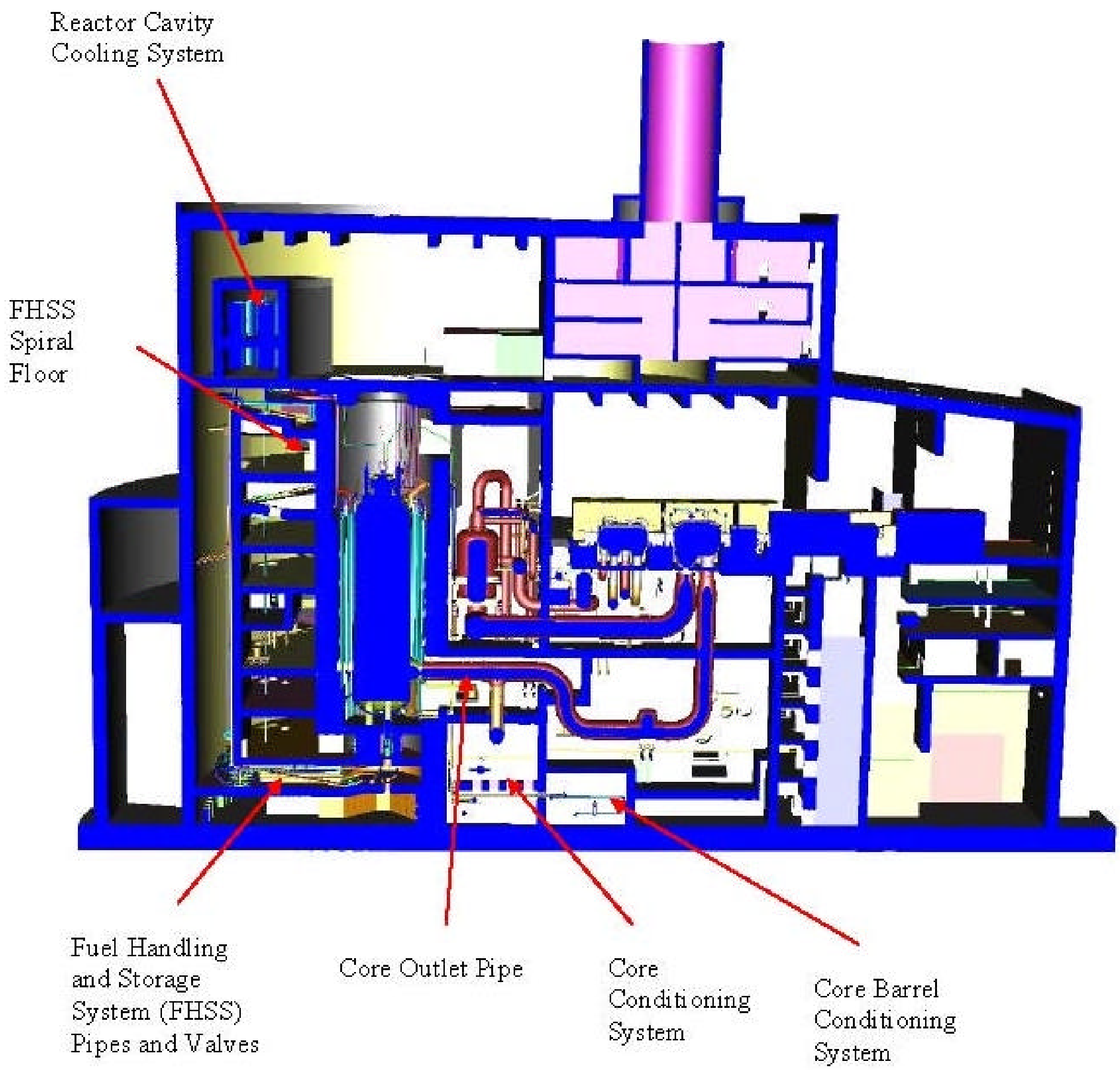

Figure 28 Elevation Requirements of Major SSC With Respect to the Reactor

The following key systems have specific location requirements with respect to the Reactor:

- The Reactivity Control System, Reserve Shutdown System and Core Unloading Device are an integral part of the Reactor Unit Assembly.
- The Fuel Handling and Storage System (FHSS) valves and piping.
- The Reactor Cavity Cooling System storage tanks must be higher than the Reactor to facilitate the passive cooling function. The RCCS stand pipes must be at the same elevation as the RPV within the reactor cavity.

- The In Core Delivery System must be above the reactor to facilitate top entry into the core.
- The Core Conditioning System (CCS) is located below the Core Outlet Pipe (COP). Positioning of the CCS heat exchanger circuit should be below the Reactor due to reduce the probability of water leakage into the core. The CCS is located as close to the COP as possible to reduce the Hot Pipe length.
- The HSS Tanks and Fuel Storage Tanks are located as low in the building as possible due to their size and weight. In the DPP an advantage of placing them below ground level is the additional protection from external events.

Table 20 identifies the major systems to be housed in the Reactor Building, the relative safety classification and requirements for location with respect to the Reactor and other systems. The following safety classifications were taken from Reference [2] for input to this table. The safety classification are assumed for the purposes of this study only in order to gain an understanding of the portion of the Reactor building that will house SSCs requiring protection from internal and external hazards. Table 20is not to be interpreted as a formal position on the SSC safety classification. Safety classification will be developed during conceptual and preliminary design based on the RI-PB safety analysis and licensing approach.

Safety-Related SSCs (SR):

This category is for SSCs relied on to perform required safety functions to mitigate the public consequences of Design Basis Events (DBEs) to comply with the dose limits of 10 CFR §50.34[8]

This category is also for SSCs relied on to perform required safety functions to prevent the frequency of Beyond Design Basis Accidents (BDBEs) with consequences greater than the 10 CFR §50.34[8] dose limits from increasing into the DBE region.

Non-Safety-Related with Special Treatment (NSR-ST):

This category is for SSCs relied on to perform safety functions to mitigate the consequences of Anticipated Operational Occurrences (AOOs) to comply with the offsite dose limits of 10 CFR Part 20[79]

This category is also for SSCs relied on to perform safety functions to prevent the frequency of DBEs with consequences greater than the 10 CFR Part 20 [79] offsite dose limits from increasing into the AOO region.

Non-Safety-Related with No Special Treatment (NSR)

This category is for all SSCs not included in either of the above two categories.

The basic conclusion drawn from Table 20 is that the required reactor building foot print is likely to be larger than just the reactor cavity and citadel structure at all elevations including the lower elevation and therefore the foot-print to be embedded is quite large. Figure 29 below provides the initial layout of the NGNP under consideration which is based on the NGNP PCDR baseline.

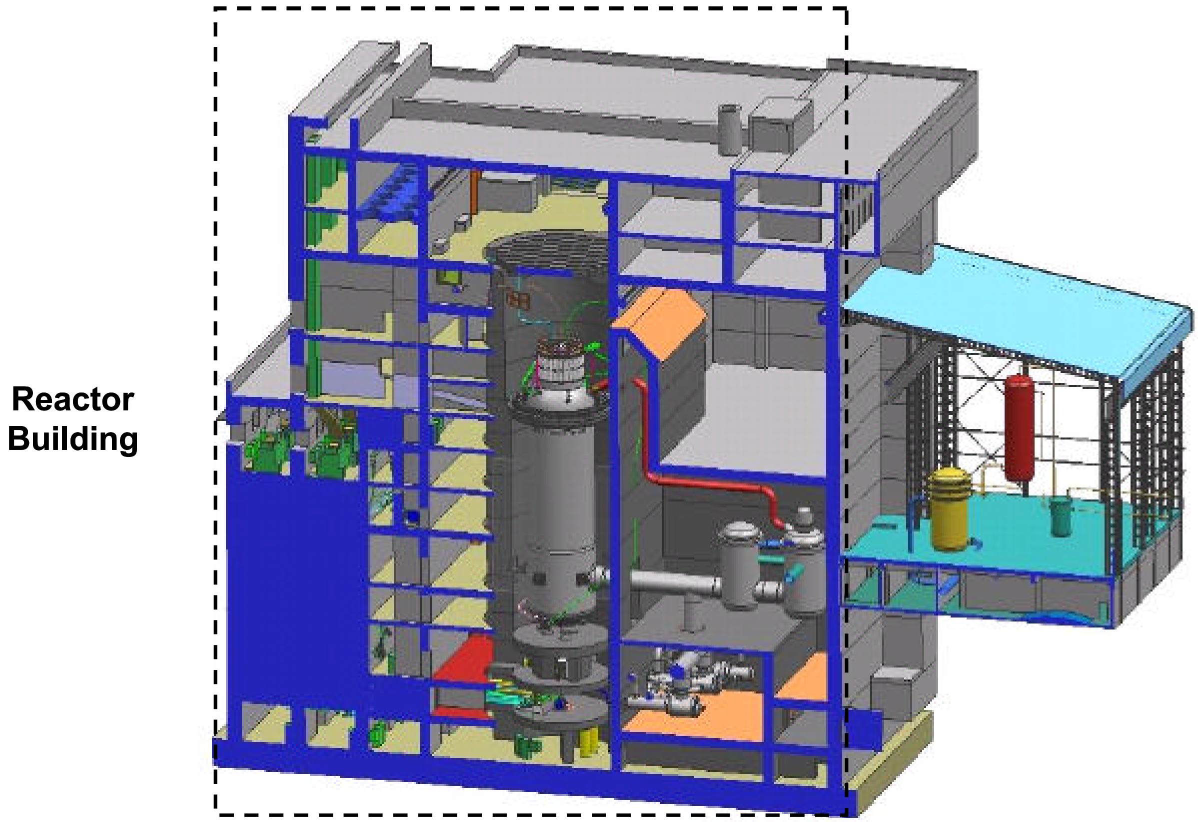

Figure 29 NGNP Reactor Building Layout

Table 20 Reactor Building Systems

Nuclear Heat Supply System (NHSS)	Fixed elevation relative to RPV	NGNP Safety Class	Remarks
• Reactor Unit			
○ Core Barrel Assembly	YES	SR	Reactor Internals
○ Core Structures Ceramics	YES	SR	Reactor Internals
○ Reactivity Control System	YES	SR	Installed on top of RPV Lid
○ Reserve Shutdown System	YES	SR	Installed on top of RPV Lid
○ Reactor Pressure Vessel	YES	SR	Vessel itself
○ In-core Delivery System	YES	NSR-ST	Reactor Internals
• Core Conditioning System			
○ CCS Blower	NO	NSR-ST	Location close to bottom of Reactor Vessel is desirable to minimize helium piping
○ CCS Heat Exchanger	YES	NSR-ST	Location below Reactor Vessel prevents possible gravity water leakage to reactor
○ CCS Valves	NO	NSR-ST	
○ CCS Piping	NO	SR	Arrangement can be optimized to suit embedment but constrained by COP and CCS HX locations
• Reactor Cavity Cooling System			
○ RCCS Header Tank Vessels	YES	SR	Location above reactor vessel required for Natural circulation

Nuclear Heat Supply System (NHSS)	**Fixed elevation relative to RPV**	**NGNP Safety Class**	**Remarks**
○ RCCS Inlet and outlet manifolds	YES	SR	Location required for Natural circulation
○ Oval stand pipes	YES	SR	Location required for Natural circulation. Oval stand pipes on same elevation as PRV to protect reactor cavity concrete.
• Fuel Handling and Storage System			
○ Sphere Storage Subsystem	NO	NSR-ST	
○ Sphere Circulation Subsystem	YES	NSR-ST	Design development extensive It is desirable to maintain DPP geometry
○ Sphere Replenishment Subsystem	YES	NSR-ST	Design development extensive It is desirable to maintain DPP geometry
○ Fuel Handling Control Subsystem	NO	NSR-ST	Arrangement can be optimized to suit embedment
○ Circulating Gas Subsystems	NO	NSR-ST	Arrangement can be optimized to suit embedment
○ Sphere Decommissioning Subsystem	NO	NSR-ST	Arrangement can be optimized to suit embedment
○ Auxiliary Gas Subsystem	NO	NSR-ST	Arrangement can be optimized to suit embedment
○ High-level Waste Handling Subsystem	NO	NSR-ST	Arrangement can be optimized to suit embedment
• Helium Services System			
○ Inventory Control System	NO	NSR-ST	Arrangement can be optimized to suit embedment

Nuclear Heat Supply System (NHSS)	Fixed elevation relative to RPV	NGNP Safety Class	Remarks
○ Helium Purification System	NO	NSR-ST	Arrangement can be optimized to suit embedment
○ Helium Make-Up System	NO	NSR-ST	Arrangement can be optimized to suit embedment
• NHSS Control & Instrumentation System			
○ Operational Control System	NO	NSR-ST	Arrangement can be optimized to suit embedment
○ Reactor Protection System	NO	SR	Arrangement can be optimized to suit embedment
○ Equipment Protection System	NO	NSR-ST	Arrangement can be optimized to suit embedment
• NHSS Cooling Water System			
○ Auxiliary Component Cooling Water System	NO	NSR-ST	Arrangement can be optimized to suit embedment
○ Equipment Protection Cooling Circuit	NO	NSR-ST	Arrangement can be optimized to suit embedment
• NHSS Electrical System	NO	NSR-ST	Arrangement can be optimized to suit embedment
• NHSS HVAC System	NO	NSR-ST	Arrangement can be optimized to suit embedment
• Primary Loop Initial Cleanup System	NO	NSR-ST	Arrangement can be optimized to suit embedment
• NHSB Pressure Relief System	NO	SR	Arrangement can be optimized to suit

Nuclear Heat Supply System (NHSS)	Fixed elevation relative to RPV	NGNP Safety Class	Remarks
			embedment
• Heat Transport System (HTS)			
○ Primary Heat Transport System (PHTS)			
• PHTS Circulator	NO	NSR-ST	Need to minimize Helium Piping Lengths
• PHTS Valve	NO	NSR-ST	Need to minimize Helium Piping Lengths
• Intermediate Heat Exchanger (IHX)			
• IHX Core	NO	NSR-ST	To be determined during Conceptual Design.
• Internal Ducts, Supports and Insulation	NO	NSR-ST	To be determined during Conceptual Design.
• IHX Vessel, Including External Supports and Insulation	NO	SR	To be determined during Conceptual Design.
• Piping (primary circuit from Reactor to IHX and between IHX vessels - both hot and cold legs)			
• Pressure Boundary Piping, Including External Supports and Insulation	YES	SR	Need to minimize Helium Piping Lengths
• Piping Internal Ducts, Supports and Insulation	YES	NSR-ST	Need to minimize Helium Piping Lengths
• PHTS Pressure Relief System	NO	SR	
○ Secondary Heat Transport System (SHTS)			

Nuclear Heat Supply System (NHSS)	Fixed elevation relative to RPV	NGNP Safety Class	Remarks
• SHTS Circulator	NO	NSR-ST	Need to minimize Helium Piping Lengths
• Helium Isolation Valves (if required)	NO	NSR-ST	Need to minimize Helium Piping Lengths
• Piping (secondary circuit, hot and cold legs, plus PCHX to SG)			Need to minimize Helium Piping Lengths
• Pressure Boundary Piping, Including External Supports and Insulation	NO	NSR-ST	Need to minimize Helium Piping Lengths
• Piping Internal Ducts, Supports and Insulation	NO	NSR-ST	Need to minimize Helium Piping Lengths
• SHTS Flow Coupling and Mixer	NO	NSR-ST	Need to minimize Helium Piping Lengths
• SHTS Pressure Relief System	NO	SR	

B1.2 HEAT DISSIPATION TO THE ENVIRONMENT

The ability of the building to transfer heat from the reactor through the reactor cavity to the environment either to ground or to ambient air is considered. First it is important to understand that the transfer of heat from the reactor is normally through the Heat Transport System (HTS) to the Power Conversion System (PCS), with a small portion of waste heat going to the RCCS. When the HTS and PCS are not available the CCS is used to remove heat. In a design basis event where the CCS is no longer available, the RCCS would be the heat transfer pathway to control the temperature of the reactor cavity. Heat transfer through the reactor cavity wall to the environment would only occur in a very unlikely beyond design basis event where the RCCS is emptied of water and no longer effective to remove heat.

Prior work by PBMR has led the team to expect that the thermal capacity of the Reactor Citadel walls is so high that the external medium will not influence the internal concrete temperatures significantly.

As an additional check, PBMR has completed a scoping analysis and documented it in reference [28]. The basis for this analysis was a BDBE which combined a DLOFC and a loss of RCCS water on initiation of the event. The objective of this calculation is to determine if there is a significant difference between heat transfer from the reactor citadel to air, soil or clay. The analysis considers heat transfer to air, sandy soil (low moisture content and low thermal conductivity) and clay (high moisture content and high thermal conductivity) with a high water content. Figure 30 shows a schematic view of this heat transfer model of the reactor cavity.

The maximum and average concrete temperatures at a selected height are plotted below in Figure 31 and Figure 32 to see the difference for the various media. The results also include a reference calculation with a fixed boundary condition of 40ºC on the outer surface of the reactor cavity. The maximum concrete temperature is the inside temperature, while the average is a mathematical average across the width but at a specific height. It can be seen that the maximum temperature is almost identical, while the average temperature only differs slightly. The average concrete temperature is only slightly cooler with clay surrounding the cavity. Clay has a high moisture content and a resultant high thermal conductivity. Sand surrounding the cavity does not show an improvement above air.

It must be stressed that the transient results shown the figures are not physically realistic, as the RPV and concrete temperatures exceed their design values early on in the transient and the embedment solution will not influence the ability of the citadel to perform its function of structural support to the reactor (NHSB-2.2.1).

The conclusion from this is that heat transfer should not be a driver in the decision to embed the reactor.

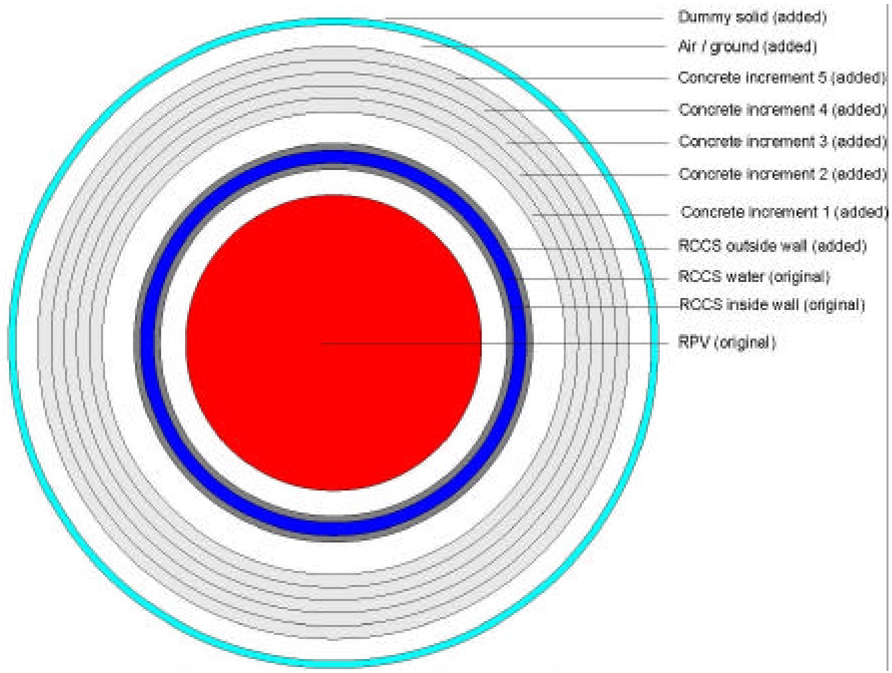

Figure 30 Top View of Reactor Cavity Schematically Showing the Added Layers

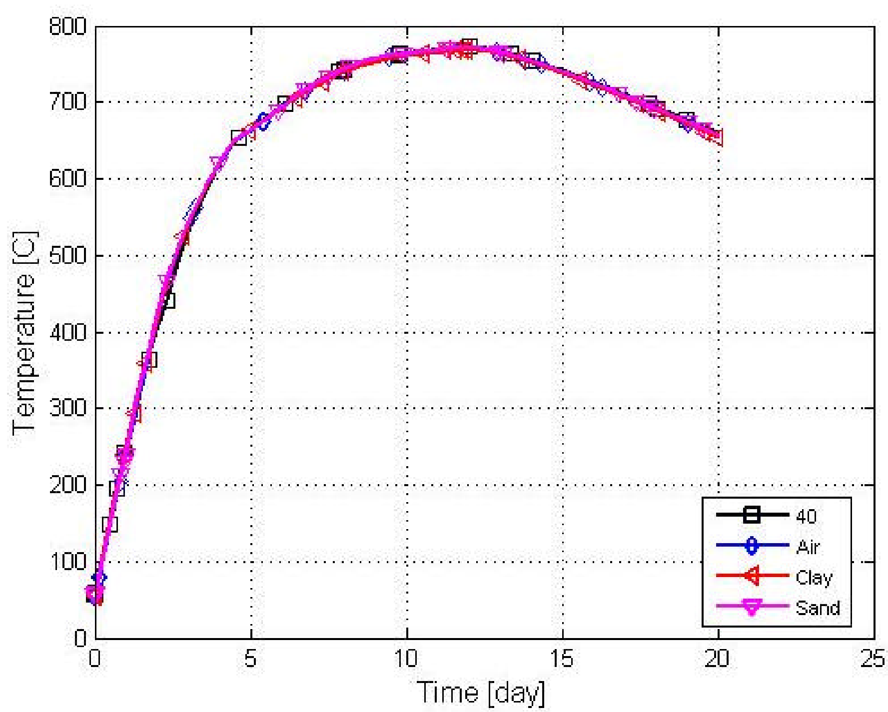

Figure 31 Maximum Concrete Temperature at a Height Corresponding to the Middle of the Pebble Bed

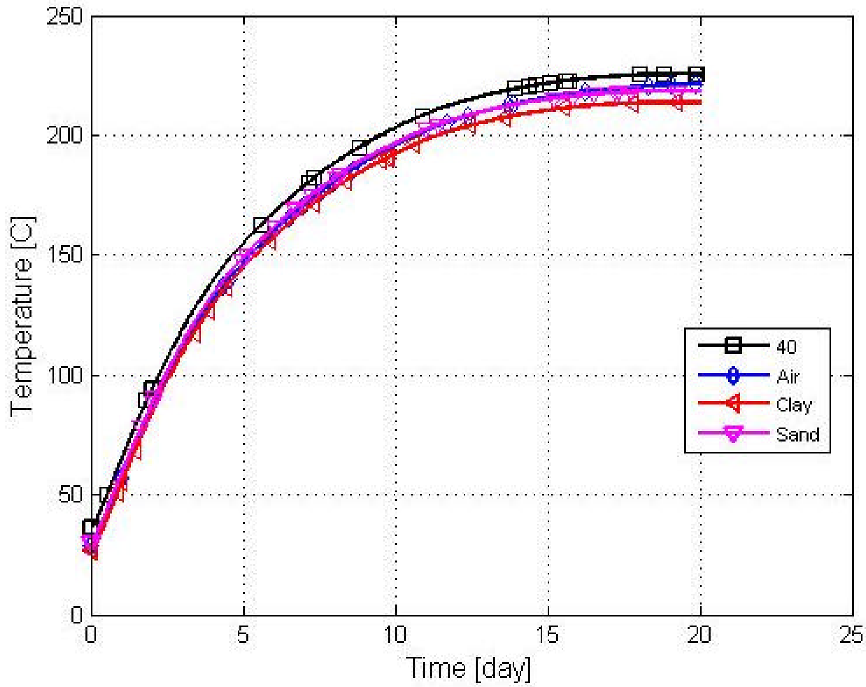

Figure 32 Average Concrete Temperature at a Height Corresponding to the Middle of the Pebble Bed

B1.3 WATER TABLE EFFECTS

Excluding arid regions, the design water table at most sites will likely be less than 15 m below the ground surface.

A structure founded below groundwater level must be designed to resist flotation, and to withstand static water pressures on the submerged portion of the exterior. Additionally, the submerged portion of the exterior must be waterproofed to prevent seepage into the structure.

During construction, the groundwater level must be temporarily lowered to permit construction in the dry.

Groundwater withdrawal permits must be obtained before groundwater can be removed and disposed. This requires making estimates of groundwater withdrawal rates and of potential effects on existing groundwater users.

At INL, the approximate depth to groundwater ranges from 60 m in the northern part to 275 m in the southern part. Therefore, groundwater would likely only be a design consideration in the northern part of INL, and then only for a structure that is fully-embedded or nearly fully-embedded.

B1.3.1 Flotation

The hydrostatic uplift force on a submerged body equals the volume of water displaced times the unit weight of the displaced water. For a structure having a regular exterior cross-section, the design hydrostatic uplift force increases in direct proportion to the depth of the structure below the design groundwater level.

Typically, resistance to flotation is provided by the weight of the structure and the weight of soil overlying any extension of the foundation beyond the exterior wall. Although additional resistance could be provided by anchoring the foundation into the underlying soil or rock, for a nuclear plant the anchors would have to be designed as a safety-related system. We are not aware of any such system at a nuclear facility.

Hydrostatic uplift reduces the normal force between the base of the structure and the underlying material, reducing the sliding resistance at the base. This is usually not a significant design issue for an embedded structure, because significant resistance to sliding is developed as the side of the structure moves against the adjacent soil.

B1.3.2 Water Pressure on Embedded Walls

The submerged portion of a structure must be designed for static lateral water pressure, which at a given depth Y below the design groundwater level, equals:

$$s_w = ?_w Y$$

Where: s_w = static water pressure
$?_w$ = unit weight of water
Y = depth below the design groundwater level

The static water pressure diagram has a linear distribution, with zero pressure at the design groundwater level and a maximum value at the base of the structure.

B1.3.3 Permanent Dewatering Systems

If it were necessary to reduce subsurface water loads on the structure, the ground water level could be lowered using a permanent dewatering system. The system might consist of pumped wells located outside the footprint of the structure, and associated discharge piping. Wells can be effective if the underlying soils are permeable, e.g. sand or gravel.

Alternatively, a zone of freely draining material could be placed beneath the base of the structure. Perforated collection pipes in the drainage layer would conduct the water to sumps, from which it would be pumped through discharge piping.

In either case, monitoring wells would be required to verify that the groundwater level is maintained at or below the design elevation.

Problems with using a permanent dewatering system for the NGNP project are:

- The system would have to be safety-related. This would likely require redundant pumps and power supply, and seismically designed discharge piping. This would substantially increase the cost of designing, constructing and operating the system.
- The potential for water sources other than groundwater would need to be considered, e.g. failure of water piping in the area.
- It might be difficult to demonstrate conclusively that the freely draining material beneath the structure would not become clogged during the life of the system, due to factors such as soil infiltration, mineral deposition or bacterial growth.

B1.3.4 Waterproofing

Waterproofing will be required on the outside of the submerged portion of the structure. Basic types of waterproofing systems are (Postma and Walker, 2006):

- Cementitious systems: These contain Portland cement with water and sand combined with an active waterproofing agent. These systems include metallic, crystalline, chemical additive and acrylic modified systems.
- Fluid-applied systems: These include urethanes, rubbers, plastics, and modified asphalts. Fluid membranes are applied as a liquid and cure to form a monolithic seamless sheet.
- Sheet-membrane systems: These include thermoplastics, vulcanized rubbers, and rubberized asphalts.
- Bentonite clays: Natural clay known as bentonite acts as waterproofing by swelling when exposed to water, thus becoming impervious to water. The bentonite is sandwiched between two layers of geo-textile or paper, and is furnished in panels or sheets.

For this study, fluid-applied systems will be considered, because they are easy to construct. They can be applied to vertical and horizontal surfaces, and can be installed in relatively tight quarters, such as in a narrow annular space around a foundation wall.
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B1.3.5 Fluid-Applied Waterproofing Membrane

A fluid-applied membrane is sprayed or rolled onto the surface to be waterproofed, and later solidifies. The fluid may be applied in one or more layers to form a membrane having a required dry thickness. After the fluid is applied the covered area is inspected for proper thickness, and for pinholes, blisters or other voids in the membrane. If a defect is detected, additional material is applied over the defect and surrounding area until a monolithic layer of the specified minimum thickness is formed.

An example of a fluid-applied waterproofing system is the water-borne asphalt emulsion manufactured by CETCO Liquid Boot Company. This product is also used as a gas vapor membrane.

To waterproof the bottom of the structural mat, a 75 mm thick mud mat would be placed and allowed to set. A base geotextile would be placed over the mud mat, followed by the asphalt emulsion membrane (e.g., 2 mm dry thickness). After the membrane has cured and been checked for proper thickness and absence of flaws, a protection board or protection mat would be placed over the membrane. This would be followed by another 75 mm mud mat, and then by the structural mat. Membrane installation on vertical surfaces (mat and embedded foundation wall) would be similar, except that no mud mats would be placed.

B1.3.6 Construction Dewatering

For excavations below the groundwater table, the groundwater level must be lowered some distance (say 1 m) below the bottom of the excavation, in order to provide a reasonably firm working surface. In sands or gravels, deep wells would likely be used where a wide excavation must be made or where the depth of excavation below the groundwater table is more than 9 to 12 m, or where artesian pressure in a deep aquifer beneath an excavation must be reduced. The required rate of groundwater withdrawal, and therefore the number and spacing of wells, depends on the depth that the water surface has to be lowered, the area of the excavation to be dewatered, the permeability of soil, the presence of nearby sources of recharge (such as a stream), and the distance from the wells to the excavation.

In fine-grained silts with low permeability, gravity alone will not drain the soil, because capillary forces hold the water in the soil voids. In such case, a vacuum dewatering system can be used. The system consists of wells with the screen and riser pipe surrounded with a free-draining sand filter extending to within a few feet of the surface. The remainder of the hole is filled with impervious soil. By maintaining a vacuum in the well screen and filter, the hydraulic gradient producing flow toward the well is increased. In order to dewater this type of soil properly, it us usually necessary to install the wells fairly close together.

Preliminary estimates of soil permeability can be made from correlations with the soil grain size, and results of small-scale field permeability tests in boreholes. Such estimates can be used to identify suitable dewatering options. A field pumping test is typically performed to provide values for final design of the system.

Lowering the groundwater level at the excavation lowers the groundwater level in the surrounding area. This effect diminishes with increasing distance from the dewatering wells, and eventually disappears where the distance is sufficiently great. Effects on existing groundwater users within the zone of influence of the dewatering system have to be evaluated, along with potential effects on existing structures within this zone. The latter is necessary because lowering the groundwater level increases the vertical effective stress on the soil. If a structure within the affected area is underlain by a layer of highly compressible soil, lowering the water level could lead to unacceptable settlement of the structure. If such a situation were identified, consideration would be given to recharging water to reduce or prevent lowering of the water table at the structure.

Groundwater withdrawal must start some time before the excavation reaches the normal groundwater level, in order to maintain the groundwater the required distance below the current bottom of the excavation.

A permit is typically required to remove groundwater during construction. Also, the quality of the water to be discharged must meet regulatory requirements.

Dewatering is typically performed by a specialty contractor. Dewatering can be costly, because the dewatering system must be operated continuously (24 hours per day, 7 days per week). Three shifts of operators will likely be required.

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B1.4 GEOTECHNICAL CONSTRAINTS AND FOUNDATION PERFORMANCE

B1.4.1 Temporary Excavation Support

B1.4.1.1 Excavations in Soil

Open cuts with sloped sides are feasible to very large depths, provided there is sufficient space to accommodate the wide excavation and provided the excavation has an acceptable cost. Typical excavation slopes in sand range from approximately 1.75 H: 1V to 2H: 1V. Flatter slopes are required in weaker soil. Equipment access to the excavation is by ramp having a maximum slope of approximately 10 percent. With an open cut, the top of slope is some distance from the structure to be constructed, requiring longer crane reach.

Vertical cuts require various types of excavation support, depending on the excavation depth. For shallow excavations in soil, a cantilever wall of steel sheet piles or pipe piles can be used, or a soil nail wall can be constructed. Soil nailing is a method of reinforcing existing soil by installing threaded steel bars into the cut slope as wall construction proceeds from top down. The bars are grouted in place to create a stable mass of earth. The excavated face is then covered by shotcrete, which is reinforced with wire mesh and attached to each bar by a plate.

For deep excavations, a braced secant pile wall or braced concrete diaphragm wall can be used. Bracing can be external or internal. External bracing consists of earth or rock anchors attached to horizontal beams at vertical intervals along the inside (structure side) of the wall. Internal bracing for a circular wall consists of horizontal beams at vertical intervals, with the beams acting as compression rings.

Factors affecting the design of the bracing system include:

- Soil strength and stiffness (affect the lateral soil stress on wall)
- Groundwater level during construction (determines water pressure on wall)
- Adjacent surcharge loads, such as heavy cranes
- Construction method and sequence

Note that deep cuts may cause some lateral movement of the adjacent soil. It is necessary to evaluate the potential effect on any adjacent structures or underground utilities.

B1.4.1.2 Excavations in Rock

Excavation in rock would be performed using drill and blast. Controlled blasting would be performed to limit damage to the rock and to minimize breakage beyond the excavation contract pay limit. Vibration monitoring would be performed during blasting to verify that design velocities are not exceeded. Loose rock would be removed from the rock surface. The exposed rock surface would be mapped geologically to verify that no active fault passes through the excavation.

Anchor bolting would be required to prevent sliding or toppling of rock into the excavation. In addition, a protective layer of shotcrete would be placed over the rock surface. The shotcrete would be applied over welded wire fabric that is attached to the rock by short rock anchors or dowels.

- Major factors affecting design of the rock slopes include:
- Rock strength. There are more problems with weak rock.
- Orientation of discontinuities in the rock. There are more problems where slope discontinuities such as joints and shear planes slope downward into the excavation.
- Spacing of discontinuities. More support is needed for close spacing
- Shear strength along discontinuities. There are more problems with low strength.

- Presence of water in the discontinuities. Water pressure adds to the force to be resisted by the rock support system.
- The location and thickness of major soil beds between basalt flows at INL. The presence of a thick soil bed within the excavation depth increases requirements for excavation support.

B1.4.2 Lateral Earth Pressures

B1.4.2.1 Static Lateral Earth Pressure

For deeply embedded structures with rigid walls, the static earth pressures at any depth equals the at-rest lateral earth pressure, which is calculated as:

$$s_{h} = K_o\, s'_{v} = K_o\, [(?_{moist})\,(h_{moist}) + (?_{sub})\,(h_{sub})]$$

Where: K_o = at-rest lateral earth pressure coefficient of the soil

$?_{moist}$ = moist unit weight of soil

h_{moist} = thickness of moist soil above the depth in question

$?_{sub}$ = submerged unit weight of soil

h_{sub} = thickness of submerged soil above the depth in question

In general, the static lateral earth pressure increases with the depth to the groundwater level. It increases at a slower rate below the groundwater table because the submerged unit weight of the soil is less than the moist unit weight..

The lateral earth pressure is increased by surcharge loading near the excavation, with the effect determined by the geometry and intensity of the loading. For surcharge of great extent, the surcharge pressure is generally taken as K_o q, where q is the applied surcharge pressure.

B1.4.2.2 Dynamic Lateral Earth Pressure

Dynamic lateral earth pressures on embedded structures are determined by soil-structure interaction (SSI) analysis, taking into account the following:

- Relative motion between the ground and the embedded portion of the structure
- Modulus of lateral subgrade reaction, which depends on the stiffness of the structure wall and of the ground.

B1.4.3 Backfill Around Embedded Structure

Where there is room, fill typically consists of structural fill compacted to 95% of ASTM D1557 maximum dry density. Vibratory compactors are used, with the lift thickness and number of compactor coverages per lift depending on the compactor used. Light compactors are used near walls. Backfill height is kept relatively uniform around the structure to prevent unbalanced loading on the structure.

Where tight access makes placing compacted backfill difficult or uneconomical, controlled density fill can be used. This is a fluid mix of cement, sand, fly ash and water that flows easily into place, and then hardens to form a material that is as stiff as or stiffer than compacted granular fill. Controlled density fill needs no compaction. A layer is placed and allowed to set before another layer is placed.

B1.4.4 Allowable Bearing Pressure

The allowable bearing pressure is governed by the lower of the following: the pressure that has an adequate factor of safety against bearing capacity failure, and the pressure that causes the maximum allowable settlement.

Bearing capacity is generally not a significant concern for nuclear structures founded on large mat foundations, because such structures are typically founded on competent soil or on rock. Embedment increases the downward soil pressure adjacent to the base of the structure. This pressure resists a bearing capacity failure. Therefore, increasing the embedment increases the allowable bearing pressure.

Settlement estimates are performed to estimate the amount that the mat-supported structure will settle. In general, allowable settlement of mat-supported structures is in the order of 2 inches.

B1.4.5 Modulus of Subgrade Reaction

The modulus of vertical subgrade reaction K_v is used to characterize the stiffness of the supporting medium in designing mats and slabs on grade using the Winkler method. K_v is defined as:

$$K_v = q / d$$

Where: q = applied stress at the soil surface
d = settlement

K_v is expressed in units of $(F/L^2)/L$ which is the same as F/L^3.

The magnitude of K_v depends on the stiffness of the soil, the width, shape and depth of the loaded area, and the position on the mat or slab. Therefore, K_v is not a fundamental soil property.

If all other conditions are the same, K_v increases with embedment depth, because settlements are smaller for the same applied stress. This occurs because the change in stress in the soil due to the applied q is a smaller percentage of the initial stress (Coduto, 2001). However, where the soils are competent, embedment would have only a small impact on structural design.

B1.4.6 Sliding Stability

The factor of safety against sliding is the sum of the forces resisting sliding divided by the sum of the forces causing sliding.

The resisting forces are the base shear resistance and the passive soil pressure that develops as the structure is driven into the soil on the side opposite the driving force. The latter depends on the amount that the structure moves into the surrounding soil.

The base shear resistance equals the normal force times the coefficient of sliding friction. The latter value is typically controlled by the coefficient of friction of the waterproofing system, which is dependent on the membrane material, its hardness and/or surface roughness, the type of material on either side of the membrane, the applied normal stress, the rate of loading, and whether the materials are dry or saturated. Such values are determined by shear box tests performed according to ASTM D 5321, using project specific materials and conditions.

Hydrostatic uplift reduces the normal force between the base of the structure and the underlying material, reducing the sliding resistance at the base. This is usually not a significant design issue for an embedded structure, because significant passive resistance is developed as the side of the structure moves against the adjacent soil.

B1.4.7 Structure Overturning

Wind or seismic forces exert lateral forces on a structure, creating an overturning moment. Resisting moments are developed by the weight of the structure (minus the force due to any hydrostatic pressure on the base of the mat) and by passive earth pressure acting on the side of the structure that moves against the adjacent soil. Therefore, embedment increases the factor of safety against overturning.

B1.4.8 Structure Settlement

Embedding a structure decreases the amount of settlement that the structure will undergo. This is due to the fact that the net vertical stress imposed on the underlying material equals the average bearing stress minus the effective stress that was exerted by the excavated material. This net vertical stress is always less than that imposed by a structure founded at grade.

B1.4.9 Input to Soil-Structure Interaction Analysis

Strain-compatible values of shear modulus and damping for the subsurface materials at the site are required for input to analysis of soil-structure interaction. To develop these values, borings are drilled at the site to determine the site stratigraphy. Cross-hole and/or down-hole seismic velocity surveys are performed in selected boreholes. These surveys provide a profile of shear wave velocity vs. depth for the soil and any rock adjacent to the boreholes in which the survey was made. These results are used to generate profiles of low-strain shear modulus vs. depth. Relationships for shear modulus vs. shear strain, and damping vs. shear strain, are established by laboratory test, or are adopted from the literature.

The shear modulus profiles are analyzed for response to shaking by a suite of time-histories generated by earthquakes of magnitude similar to the design earthquake, that were recorded on bedrock, and that have been scaled to the design bedrock acceleration value. The site response analysis is performed using computer codes such as ProShake (EduPro, 1999). The analyses provide strain-compatible values of shear modulus and damping for each profile. Values corresponding to the lower and upper bound profiles are then used in analyses of soil-structure interaction. These analyses apply the input motion at the base of the structure.

The process described above would be the same for an embedded structure and for a structure founded at the ground surface. The borings for the embedded structure would likely extend to somewhat greater depth, increasing the boring cost.

As discussed in Section B1.9, embedding a structure decreases the input motion at the base of the structure. Certain limits apply. For example, NRC Standard Review Plan 3.7.2 requires the horizontal component of the acceleration at foundation depth to be no less than 60 percent of that at finished grade in the free field.

B1.5 CONSTRUCTION CONSIDERATIONS

Construction complexity will be assessed. The effect of the embedment depth on constructability using modern techniques such as modularization is discussed. The distance below grade at which a structure is located has a significant impact on the complexity of the construction process. It affects not only the construction of the foundations but the superstructure and interior components as well. Geotechnical conditions vary greatly from site to site and range from significant depths of soils to solid rock at or near the surface. Each geotechnical condition presents its own unique challenges when constructing the foundation.

To embed a structure on a site where solid rock is at or near the surface would start with drilling and blasting so that the rock can be removed to the depth of the bottom of the foundation. This is a costly and dangerous process that can take a long time to complete. In addition, there is potential for collateral damage to existing structures nearby due to vibrations and shock waves set up by the blasting. While the likelihood of significant flooding as the result of ground water in solid rock is small, pumping systems are still required to remove water that may seep in through cracks or that is deposited by rains.

On soil sites, the major issues include dealing with ground water and stability of the soils. The techniques for dealing with ground water generally involve some means of pumping water away from the excavation. The various dewatering techniques generally involve continuous pumping and are well proven. Disposal of the water can be a costly process if there isn't a place to receive the water in the immediate vicinity.

In general, protection of earthen side walls from cave-in must be addressed for any excavation deeper than about 1 meter. The US Department of Labor regulations found in 29 CFR 1926 (OSHA) require that side walls be benched or sloped at 1.5 (horizontal) to 1 (vertical), unless engineered protective means are employed such as sheet piling or other bank stability or bank restraint systems.

Deep embedment of a structure without sheet piling or a similar restraint system would require that significant amounts of soil be removed and stockpiled. For a deep excavation, the amount of soil to be excavated and returned to provide slope stability can easily exceed the volume of soil required to be excavated to contain the structure. Additional amounts of soil would also have to be removed and returned for the roads necessary to drive heavy equipment into the excavation.

Depth of embedment can have a significant affect on the ability to use certain modern construction techniques such as modularization. Modularization entails assembling smaller plant components including interconnecting piping and electrical services onto a larger assembly or module at a shop located away from the congested construction area then installing the completed assembly in the plant. This reduces the installation time required at the congested jobsite.

With a deeply embedded structure, modules in the lower elevations would have to be installed very early and generally before floors are installed. This requires modules to be designed, constructed and delivered to the jobsite very early in the construction schedule which may not be practical. This also requires that the modules and their components be protected from significant hazards including weather and construction operations above including concrete placement since these modules would have to be placed in the structure before concrete floors are installed. It should be noted however that modules can incorporate structural elements including leave-in-place steel forms which can reduce the amounts of rebar that is required.

Even without modularization, deep embedment will pose certain construction challenges. Working below grade requires significant ventilation and continuous air monitoring to assure worker safety. Some welding processes require the use of inert gasses such as argon for a shielding medium. Argon gas is extremely dangerous in confined spaces since it displaces oxygen. Argon is also heavier than air and tends to find its way to the lowest spaces in the building volume thus presenting a potential hazard to workers in those areas. Above ground structures can be use construction openings to promote ventilation, whereas below ground installations do not have this option.

B1.6 COST CONSIDERATION

B1.6.1 Capital Cost

A cost analysis which included key elements of relative capital costs for the reactor building was performed [69]. The reactor building is assumed to be 65.8 m high measured from the top of concrete mat to the roof. Two assumed configurations of structures with approximately equal volumes were considered.

1. A circular structure with a 56 m outside diameter.
2. A square structure with 50 m long exterior sides

Two foundation conditions were assumed to bound the potential site geological conditions. The first condition assumed a rock site with a water table below the bottom of the reactor building mat under full embedment of the structure. This condition is comparable to the condition expected at the INL site. The second condition is a deep competent soil site with a high water table. This condition would be anticipated along the southeastern coast and Gulf coast of the United States.

The cost benefit analysis was based on the following assumptions:

1. Internal pressures from pipe ruptures would not govern the thickness of the exterior wall of the structure.
2. Plant layout fits within the confines of the assumed building sizes.

3. Reactor Building internals will remain essentially the same regardless of the embedment depth.
4. Below grade structures would include a concrete exterior

The key elements considered in the cost analysis [69]were the thickness of the exterior wall above and below grade, the cost of excavation, including excavation support or stabilization systems, and backfill. Use of dewatering systems was also included for soil sites.

Both structure configurations were assumed to have 1.5 m thick walls above ground to protect against malevolent hazards. While this thickness is somewhat in excess of current nuclear plant designs, the current nuclear plant designs look at these cases as a “beyond design basis event” as the result of an independent pressure boundary or containment inside of the exterior wall. The NGNP reactor building is still in the schematic design phase and might not have a pressure boundary or containment. If it does have a pressure boundary, the pressure boundary will likely be present in a localized area of the reactor building. The use of a potentially thicker above ground exterior wall will result in favoring a deeper embedment. The assumed wall thickness is anticipated to be sufficient to cover all natural generated phenomena such as hurricane, tornado and earthquake.

The below grade wall thickness was established by applying a uniform lateral pressure at any given depth, (i.e., sum of soil and water pressures). For the rock site, the soil pressure was assumed based on a unit weight of rock of 2720 kg/m3 and an at-rest lateral pressure coefficient Ko of 0.3. For the deep soil site, the soil pressure was assumed based on a unit weight of 2165 kg/m3 and a Ko of 0.5. The unit weight of water was assumed at 1000 kg/m3. Both lateral soil and water pressures were assumed to increase linearly with depth. The circular wall thickness was estimated by assuming the uniform pressure was resisted by the exterior wall acting as a compression ring. The straight wall was assumed to span vertically between five major floor levels, which were assumed to act as diaphragms. Under these assumptions, the thickness of the exterior wall increased with the depth of embedment. In general, the circular wall was thinner than the straight wall associated with the square structure. In the cost comparison, the cost of the circular wall was increased to account for the increased difficulty of placing forms and reinforcement in this wall system. Also, the cost of placing a cubic meter of concrete was increased with depth of placement.

The rock excavation was assumed to require blasting. The rock face was assumed to taper at 4 V to 1 H between benches (terrace levels). Benches 3 m wide were assumed at 9 m vertical intervals. Rock anchors were assumed to be required to keep the rock face stable. Also, a thin layer of shotcrete placed over welded wire mesh was assumed to be applied to the rock face to stabilize the rock surface.

The soil excavation was assumed to occur within the confines of a 1 m thick diaphragm wall of reinforced concrete. This wall was assumed to be constructed similar to a slurry wall placed to the required depth. As the excavation progresses, walers (horizontal beams) are placed on the inside face of the wall to resist soil and water pressures. The walers become larger and more closely spaced as the depth increases. While the study provided a comparison at full embedment, it is unlikely that the fully-embedded reactor building will be stable due to the uplift pressure generated by water at this depth. In addition, constructing the diaphragm wall to a full embedment depth would at best very costly and difficult. Other methods of excavation support would need to be explored.

The cost analysis [69] looked at embedment depths from 9 m to 65.8 m in approximately 9 m intervals.

The relative results of this cost benefit analysis are shown in Table 21and Figure 33. Costs identified consist of major elements that would be expected to vary depending on the selected embedment depth. The major elements of cost include excavation, dewatering, and foundations, as well as above and below grade exterior concrete walls. The cost analysis shows that regardless of the site conditions, the cost of the embedded structure increases with depth of embedment.

Table 21 Cost Analysis

(all in k $'s)

Embedded Depth (m)	Circular Exterior Wall, Rock Below Grade	Circular Exterior Wall, Soil Below Grade	Square Exterior Wall, Rock Below Grade	Square Exterior Wall, Soil Below Grade
9	$30,517	$22,033	$32,227	$24,191
18	$45,028	$31,009	$48,614	$36,178
27	$70,022	$45,787	$77,481	$57,625
36	$112,342	$68,202	$125,304	$87,761
45	$180,872	$100,685	$202,204	$133,368
54	$287,502	$146,137	$326,660	$197,189
65.8	$451,773	$213,090	$513,311	$286,741

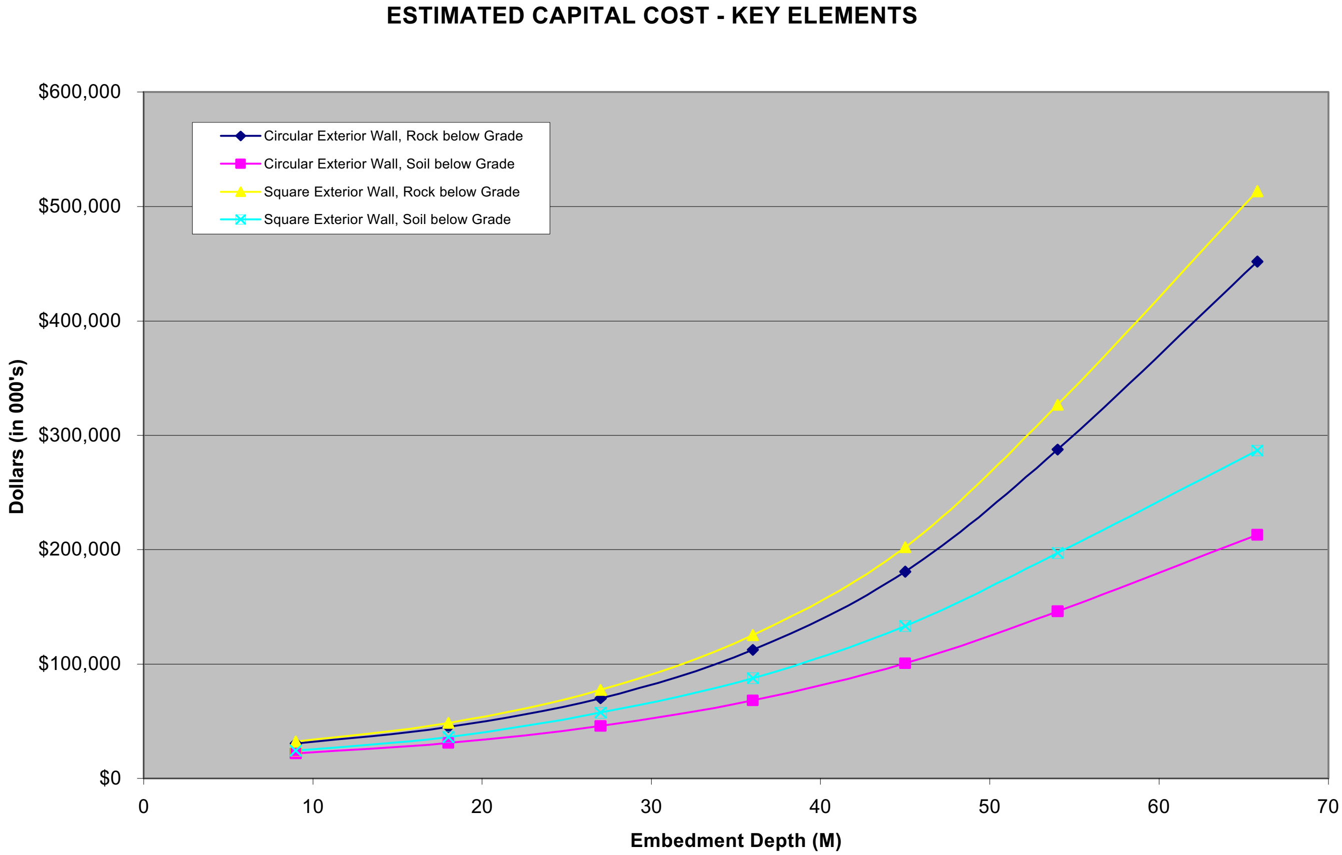

Figure 33 Estimated Capital Cost of Key Elements

The relative first cost option is just one of many attributes to be considered in selecting the depth of embedment. For example, as the amount of the safety related equipment located below grade increases, the reactor building becomes less susceptible to emerging threats. Tables 26 and 33 show the attribute scoring criteria and the embedment depth ranking summary, respectively.

B1.6.2 Life Cycle Cost

The above discussion is directed towards key elements of the capital cost that might vary with the selection of embedment depth. It recognized the there may be variation in life cycle cost for operations as well. Section B1.1 indicates that a partial embedment might result in a slight reduction in travel time to reach locations in the building to perform inspections and maintenance. This is difficult to quantify at this point without more details on the layout of equipment in the building. It is recommended that operations and maintenance needs be revisited during conceptual design to verify that the partial embedment alternative is indeed favorable with respect to access and travel times.

It could be speculated that full or partial embedment might reduce HVAC system loads and associated energy costs with reduction of solar and transmission heat loads. HVAC system sizing and performance requirements are not usually driven significantly by external conditions. Concrete structures have substantial thermal inertial to dampen effects of external temperatures. Also, HVAC flows are driven more by internal loads, fresh air, and contamination control needs. Again, it is recommended that this concern be reviewed in more detail during conceptual design.

B1.7 MALEVOLENT HAZARDS

Following the attack of September 11, 2001, the NRC issued a number of interim compensatory measures (ICMs) to better define the security requirements for operating nuclear plants. The bulk of these ICMs are Safeguards Information. The requirements in the ICMs cover a variety of security topics including plant physical design, access controls, guard force management and fitness for duty. The ICMs were developed to be consistent with the overall response to potential threats as coordinated by the United States Department of Homeland Security. This overall response strategy assigned different parts of the overall threat spectrum to different agencies such as the Department of Defense, the Department of Homeland Security, the Federal Bureau of Investigation, the NRC and others. The resultant DBT for nuclear power plants is discussed above in Subsection A1.3.7. To ensure a more comprehensive approach to security at nuclear power plants the NRC asked licensees to assess their response and mitigation design features and response strategies to threats and consequences beyond the DBT. Examples of these scenarios are large fires and explosions and the loss of all spent fuel cooling.

Orders have been issued to organizations with active Design Certification applications to assess their design's resistance to the crash of a large commercial aircraft. The orders included a force/time curve to be used as input for the assessment. Success is based upon the assessment showing that core cooling or containment integrity and spent fuel cooling or spent fuel containment integrity is maintained. For plants without a containment, a more elaborate justification of no significant radiation release is required. Assessment guidelines are included in the "Methodology for Performing Aircraft Impact Assessments for new plant Designs soon to be issued as NEI 08-13 [26]

For the reactor building embedment, there is a trade off to be made for this assessment. If the reactor building is totally underground, it presents no target for aircraft crash and no further assessment is needed other than for crash induced vibration effects on in-plant equipment. If the reactor building has portions above grade, then a beyond design basis assessment must be made in accordance with Reference 9 to show no loss of core or spent fuel cooling or containment integrity. To date, plant designs with reactor buildings incorporating robust structural design for seismic and DBT resistance can generally show acceptable aircraft crash assessment results. Also in general, these designs must be reviewed to ensure that protection is provided from all directions.

In addition, event mitigation features and licensee strategies must be provided to NRC during the Combined License application phase of plant deployment in accordance with proposed Rule 10CFR50.54(hh)2 [8].

Large fires are defined as those greater than those established in a conventional design basis fire hazards analysis. These large fires than develop with large quantities of fuel or flammable chemicals, can affect large portions of the plant including multiple fire zones, well in excess of those having their origin from on-site sources. No explicit initiation mechanism is assumed for these fires. Historically, one source of these large fires is the release of jet fuel from a large commercial aircraft crash. This has led to the need to include jet fuel as part of the large fire assessment. Assessments involve ensuring that the capability to shut down and cool down the plant and spent fuel facility is not lost nor is there significant radiation release as a result of this large beyond design basis fire. Success is based upon having readily available mitigation equipment.

No explicit mechanism is assumed for loss of all spent fuel cooling. The assumed starting point (for LWRs) is instantaneous loss of spent fuel pool water. This portion of the beyond design basis requirement should not apply to the reactor building since if houses no spent fuel in bulk. In addition, when the spent fuel storage facility is assessed for loss of cooling, new scenarios will be required since PBMR NGNP does not use water for cooling.

Rules of thumb and guidelines that, if followed, will make the reactor plant more resistant to beyond DBT are:

1. Minimize or eliminate the effective target area for aircraft crash
2. In addition to general structural robustness of the reactor building for seismic, enhance the penetration resistance of reactor building structures
3. Maximize the inherent and passive fuel cooling and radiation containment features of the plant
4. Maximize separation of redundant safety function initiation features
5. Maximize separation of safety and defense in depth system features
6. Ensure that alternate sources and means of fuel cooling are available following a beyond design basis threat event

B1.8 NATURAL PHENOMENON HAZARDS

B 1.8.1 Introduction

In general, it is proposed that the design of the NGNP Plant for natural phenomena events be carried out in conformance with the EPRI – Advanced Light Water Reactor Utility Requirements Document Revision 8 dated March, 1999 (EPRI – URD)[24]. This document should be updated to reflect the latest codes and standards as almost ten years have passed since the last revision to this document.

In general, it is the intent that the NGNP Plant would be constructed in the United States in most areas east of the Rocky Mountains. The following discussion reflects this assumption.

B1.8.2 Tornado Hazards

The current EPRI-URD calls for a design wind load of 49 meters per second (m/s) adjusted to a 100 year mean recurrence interval through the use of an importance factor of 1.11. Today, it is likely that the building code used by the local authority will be the International Building Code. The International Building Code provides a significant amount of wind design data and criteria but states that wind loads shall be determined in accordance with Section 6 of ASCE 7 - Minimum Design Loads for Buildings and Other Structures [61]. Based on the current ASCE 7 standard, the 49 m/s basic design speed would preclude the plant from being built along many sections of the eastern and gulf coast of the United States. A design wind speed of 65 m/s (3 second gust) 10 m above grade with a design exposure C adjusted by a 1.15 factor for safety related structures and 1.0 for non-safety related structures appears more appropriate for the NGNP plant design.

Following is a comparison of the EPRI-URD design parameters for tornado design versus NRC Regulatory Guide 1.76 dated March 2007 [62]:

Parameter	EPRI-URD	RG-1.76
Max Tornado Wind Speed	134 m/s	103 m/s
Max Rotational Speed	107 m/s	82 m/s
Max Translational Speed	27 m/s	21 m/s
Radius of Max Rotational Speed	45.7 m	45.7 m
Maximum Pressure Drop	138 mb	83 mb
Rate of Pressure Drop	83 mb/sc	37 mb/s

The EPRI-URD data is more conservative than RG 1.76 [62] in terms of pressure drop and wind velocity; therefore, it is proposed to use RG 1.76 [62] criteria. This criteria addresses tornado wind and pressure drop anywhere in the continental United States.

RG 1.76 [62] specifies the following tornado generated missiles be considered in the design of safety related nuclear structures:

Schedule 40 Pipe (0.168 m dia. x 4.58 m long)	130 kg	41 m/s
Automobile (5 m x 2 m x 1.3 m)	1810 kg	41 m/s
Solid Steel Sphere (2.54 cm diameter)	0.0669 kg	8 m/s

The automobile is not considered in heights greater than 9.14 m above grade.

B1.8.2 Flooding Hazards

The site shall be chosen so that the flood level including the flood from a potential dam break be kept at a minimum 0.3 m below existing grade.

B1.9 NATURAL GEOLOGICAL HAZARDS

Natural geologic hazards include seismic-related hazards, non-tectonic site deformation, and volcanic hazards. A brief listing is given below based on references [46] and [47]

B1.9.1 Tectonic and Seismic Hazards

Seismic-related hazards include site earthquake ground shaking, tectonic site deformation (fault rupture and associated tectonic surface deformation, failure induced by high tectonic stresses), ground failure induced by ground shaking including liquefaction, differential compaction and land-sliding, and earthquake-induced flooding. For sites adjacent to large bodies of water, hazards include tsunami and seiche.

In general, it is proposed that seismic design be performed in accordance with EPRI-URD [24]. Highlights of the design requirements are discussed below as well as changes imposed by the latest design codes and or standards. It is expected that a seismic Basis of Design document will be prepared early in conceptual design to capture this methodology.

B1.9.1.1 Seismic Classification

The seismic classification system will be consistent with Nuclear Regulatory Guide 1.29 [70]. EPRI-URD [24] has added Seismic Category II to address non-seismic items/structures whose collapse could jeopardize the loss of function of Safety Class components or structures. Seismic Category II requires that only structural integrity be maintained, not functional integrity. Consequently, the seismic classifications are as follows:

- Seismic Category I – This classification includes all structures, systems and components whose safety class is SC-1, SC-2, or SC-3. Seismic category I shall also include spent fuel storage pool structures including all fuel racks.
- Seismic Category II – This classification applies to all plant structures, systems and components which perform no nuclear safety function but whose failure could degrade SC-1, SC-2, and SC-3 structures, systems and/or components.
- Non-seismic Category III– This classification includes all structures that do not fall into Seismic Category I or Seismic Category II structures.

B1.9.1.2 OBE Design Basis

EPRI-URD [24] has eliminated the operating design earthquake from its load cases. The Nuclear Regulatory Commission (NRC) has essentially agreed to this change (NRC Policy Issue I.M., "Elimination of OBE); however, there continues to be discussions between EPRI and the NRC concerning the number of one-half Safe Shut-Down Earthquake (SSE) cycles to be used in the evaluation of equipment and components in regard to fatigue and seismic performance.

B1.9.1.3 Ground Motion Characteristics

EPRI proposes to use a SSE comprising a single ground motion spectrum conforming to Regulatory Guide 1.60 [63] anchored to 0.3g peak ground acceleration and applied at the free-field soil surface except at site where rock extends above the nuclear island founding level where the peak ground acceleration is applied at the top of rock. This peak ground acceleration will allow the NGNP to be constructed at most sites in the continental United States east of the Rocky Mountains.

The design response spectra shall be in accordance with Regulatory Guide 1.60 – "Design Response Spectra for Seismic Design of Nuclear Power Plants" [63] with a time history to envelop the design spectra.

The project will initially use a deterministic seismic approach that will provide an enveloping ARS for a standard NGNP design. Specific site criteria, when available, will be checked against this enveloping ARS.

Once a site is selected, the vibratory ground motion characteristics will be evaluated in accordance with the requirements set forth in Section 100.23, "Geologic and Seismic Siting Criteria," of Title 10, Part 100 [72], of the Code of Federal Regulations (10 CFR 100), "Reactor Site Criteria." [57].[54] Regulatory Guide 1.165 "Identification and Characterization of Seismic Sources and Determination of Safe Shutdown Earthquake Ground Motion" [71] provides Seismic Hazard Analysis (PSHA) for sites in different parts of the U.S. An acceptable method for determining the annual probability of exceeding the SSE, defined as the reference probability, is described in Appendix B of Regulatory Guide 1.165 [71]. The development of seismic hazard results will be based on a site-specific PSHA and will consider site amplification effects.

The NRC also provides an alternative approach to satisfy 10 CFR 100.23 [72] and Appendix S, "Earthquake Engineering Criteria for Nuclear Power Plants" to 10 CFR Part 50 [76]. This method, detailed in Regulatory Guide 1.208 "A Performance-based Approach to Define the Site-Specific Earthquake Ground Motion" [73], provides guidance for development of the site specific ground motion response spectrum (GMRS). The performance based approach combines ground motion hazard with equipment/structure response to establish risk-consistent GMRS. It differs from the hazard-consistent ground shaking that would be determined from the hazard reference probability described in Regulatory Guide 1.165 [71].

The methodology for developing the GMRS is consistent with ASCE/SEI Standard 43-05, "Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities" [74]. The method is based on the use of site-specific mean seismic hazard. The ASCE/SEI Standard 43-05 graded approach is designed to meet quantitative safety performance goals for a range of nuclear facilities.

B1.9.1.4 Analysis & Design

Design of Seismic Category I structures shall follow the ground motion characteristics specified above, the analytical techniques specified in ASCE Standard 4-98 [41] and meet all quality assurance requirements of 10 CFR 50 [54], Appendix B. ASCE Standard 4-98 has been accepted by the NRC as meeting the requirements for seismic analysis with some exceptions. Design of Seismic Category I concrete structures shall be in accordance with ACI 349-06 [64]. Design of Seismic Category I steel structures shall be in accordance with AISC N690L-03 [65]. This version of the AISC code for safety related steel structures for nuclear facilities is a LRFD version of the code which is in line with the concrete design code for nuclear facilities.

Design of seismic category II structures shall utilize the same analysis and design codes as those used for seismic category I structures; however, these structures may be constructed to the requirements of the non-nuclear building codes. In general, these requirements will invoke the International Building Code (IBC-06) [66] or the version specified by the local building authority.

Design of Seismic Category III structures shall follow the analytical requirements of IBC-06 [66]. IBC-06 specifies that ACI-318-05 [67] by used for the design and construction of concrete structures and that AISC – ASD/LRFD Steel Construction manual, 13th Edition [77] be used for steel design and construction. As an alternate, the use of ASCE Standard 7 for seismic analysis will be explored.

ASCE 43-05 [74] is a seismic design criteria for structures, systems and components in nuclear facilitates and is intended for use in conjunction with the design and analysis references listed above. ASCE 43-05 is similar to DOE-STD-1020-2002 [75]. ASCE 43-05 has been accepted by RG 1.208 [73] (DG 1146) [78] for defining site specific design-basis earthquake response spectrum. ASCE 43-05 [74]may be used to set different levels of seismic input for structures, systems and components that have different failure consequences.

B1.9.1.5 Soil Structure Interaction

An additional consideration of this study is to evaluate the effects of structure embedment on the seismic input ground motion at the base of the structure. In general, embedding the structure reduces the input motion at the base of the structure; thereby, reducing the seismic load. However, there are limits to this effect. Based on published literature references [38], [40] and [38][41], it is reasonably conservative to assume that embedment will not reduce the input base ground motion to less than approximately 60% of the surface ground motion. For the assumed reactor building, this limit is reached at a depth of approximately 30 m. In addition, ASCE 4 [41] states that the top 6 m or one-half of the embedment depth, whichever is less, must be neglected in the soil structure interaction formulation. Based on the above, it is estimated that little or no reduction will be realized at embedment exceeding 36 m or about 50% embedment of the reactor building. It is not unrealistic to assume that the input motion at the base of the structure varies linearly from 100% at grade to 60% at 36 m.

An example of this beneficial effect is illustrated in Figure 34 and Table 22 below.

There are many factors that contribute to amplification of the peak ground acceleration. The building stiffness, aspect ratio, and layer thickness of the soil along with earthquake characteristics significantly impact the amplification factor.

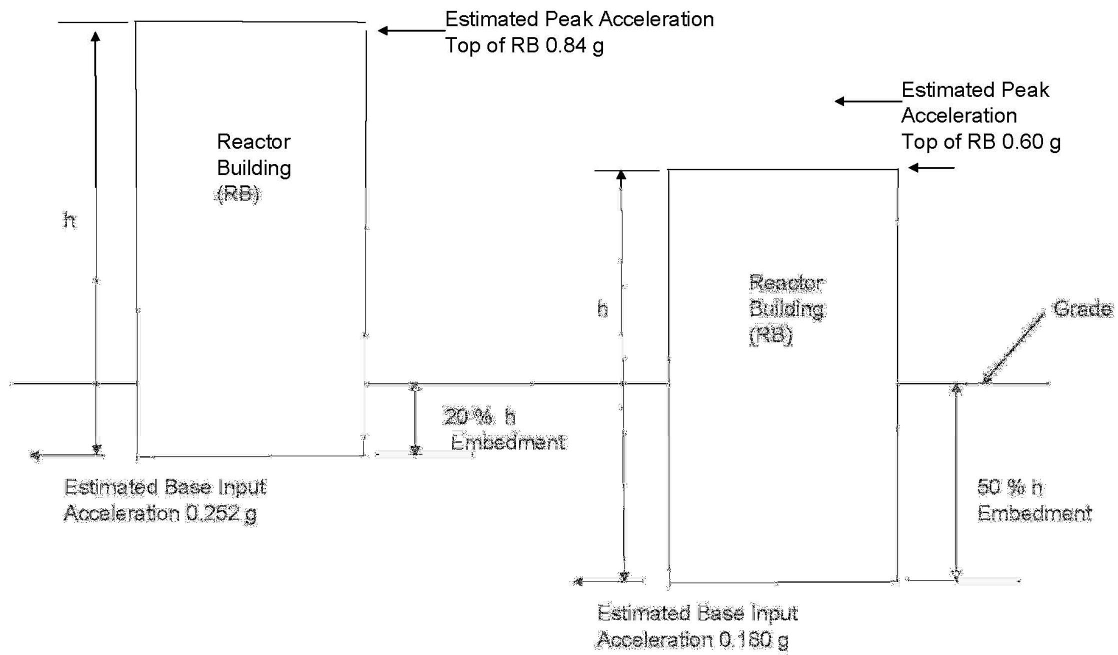

Figure 34 Reactor Building Seismic Accelerations

Table 22 Hypothetical Reactor Building Seismic Accelerations

REACTOR BUILDING EMBEDMENT PERCENTANGE	ESTIMATED INPUT ACCELERATION AT BASE OF REACTOR BUILDING (g's)	ESTIMATED PEAK ACCELERATION TOP OF REACTOR BUILDING
0	0.300	1.0 (ASSUMED)
10	0.276	0.92
20	0.252	0.84
30	0.228	0.76
40	0.204	0.68
50	0.180	0.60
100	0.180	0.60

In the above table for each embedment depth, column 2 represents the adjusted base acceleration from a peak ground level value of 0.3g. Column 3 is based on an assumed value of 1.0 corresponding to 0.3 g peak ground level acceleration. This value would ultimately be determined based on site specific seismic conditions and characteristics of the structure. The amplified acceleration at the top of the building varies proportionately with the acceleration at the base.

B1.9.2 Non-Tectonic Surface Deformation

Non-tectonic phenomena that can relate to surface deformation at a site include glacially-induced faulting, growth faulting, landslides or other mass-wasting phenomena, collapse or subsidence in areas due to underground voids such as found in karstic limestone terrain, subsidence due to excessive withdrawal of fluid (such as oil and gas), chemical weathering, and induced seismicity and fault movement caused by reservoir impoundment and fluid injection and removal.

B1.9.3 Volcanic Hazards

Potential volcanic hazards may include: lava flows, ballistic projections, ash falls, pyroclastic flows and debris avalanches, lahars (mudflow composed of pyroclastic material and water that flows down from a volcano, typically along a river valley) and flooding, seismic activity, ground deformation, atmospheric affects, and acid rains and gases. These phenomena are restricted to limited areas in the western United States.

B1.9.4 Effects on Embedment Study

The hazards listed above have to be evaluated and addressed in the licensing documents for a nuclear facility. For the purpose of this study, however, only those phenomena that differently affect embedded and non-embedded structures need be considered. These phenomena are seismic hazard and liquefaction, both of which are reduced by partially embedding the structure.

As discussed in Section B1.6, embedding a structure decreases the input motion at the base of the structure, thereby reducing the dynamic forces on the structure. Certain limits apply to this benefit. For example, NRC Standard Review Plan 3.7.2 requires the horizontal component of the acceleration at foundation depth to be no less than 60 percent of that at finished grade in the free field.

Liquefaction refers to generation of high excess pore water pressure as a loose, saturated granular soil is forced to assume a denser configuration. The excess pore water pressure reduces the shear strength of the soil. This can lead to loss of foundation support, or a flow slide on sloping ground. Embedment reduces the potential for seismically-induced liquefaction. One reason is that the ground motion decreases with depth below the ground surface, reducing the energy available to shake the loose soil into a denser configuration. Below depths of 15 to 20 m, liquefaction is generally not an issue for moderate earthquakes. If liquefaction were an issue, the soil could be densified using standard techniques, prior to building the structure.

B1.10 CHEMICAL RELEASE, EXPLOSIONS, MANMADE HAZARDS

Consideration needs to be given to the hazards of chemicals stored and or transported in range of the Reactor, the potential effects, and mitigating features that could be utilized. This type of concern does often materialize with operating facilities located near active industrial areas.

B1.10.1 Toxic Chemical Releases

Appendix A, “General Design Criteria for Nuclear Power Plants”, of 10 CFR Part 50, “Domestic Licensing of Production and Utilization Facilities”[54] requires that the design of structures, systems, and components (SSCs) important to safety be able to accommodate the effects of and be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. Criterion 19 of Appendix A requires that a control room from which actions can be taken to operate the facility safely under normal conditions and to maintain it in a safe condition during accidents. Releases of onsite and offsite hazardous chemicals, which infiltrate the control room, can result in the control room becoming uninhabitable.

Regulatory Guide 1.78, "Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release" [35], describes assumptions acceptable to the NRC staff for use in assessing the habitability of the control room during and after a postulated external release of hazardous chemicals from mobile or stationary sources, offsite or onsite, including storage tanks, pipelines, tank trucks, railroad cars, and barges. The regulatory guide directs that all hazardous chemicals present onsite in weights greater than 100 pounds within 0.3 mile of the control room or chemicals present within a 5-mile radius of the plant above certain threshold quantities be considered in a control room evaluation. The consideration of mobile sources within 5 miles of the plant is also dependent on the shipment frequency. The evaluation is performed to determine the potential toxic effects on control room operators from accidental chemical releases reaching the control room fresh air intake.

The regulatory guidance states that two types of chemical accidents should be considered for each source of hazardous chemicals: maximum concentration accidents; and maximum concentration-duration accidents. A maximum concentration accident is one that results in a short-term puff or instantaneous release of a large quantity of hazardous chemicals. The typical assumption is the failure of the largest single container resulting in the instantaneous release of the total contents of the tank or vessel. Depending on the boiling point of the chemical, the vapor release may be the result of just puddle evaporation or vaporization or may involve an instantaneous 3-dimensional gaseous puff release due to flashing followed by puddle vaporization. A maximum concentration-duration accident is one that results in a long-term, low-leakage-rate release. For a maximum concentration-duration accident, the continuous release of hazardous chemicals from the largest safety relief valve on a stationary, mobile, or onsite source should be considered.

According to Regulatory Guide 1.78 [35], the atmospheric transport of a released hazardous chemical should be calculated using a dispersion or diffusion model that permits temporal as well as spatial variations in release terms and concentrations. Atmospheric dispersion models can be used for dispersion calculations as long as these models are capable of calculating spatial and temporal variations in release terms and concentrations, simulating building wake effects, and simulating near-field effects. The NRC uses a computer code, HABIT, for control room habitability evaluation. The HABIT code is described in NUREG/CR-6210, "Computer Codes for Evaluation of Control Room Habitability (HABIT)". This code has two modules, EXTRAN and CHEM, for calculation of chemical concentration and exposure, respectively. The model in EXTRAN, a Gaussian plume or puff dispersion model, allows longitudinal, lateral, and vertical dispersions. The model also allows for the effect of wakes and for additional dispersion in the vertical direction when the distance between the release point and the control room is small. Other atmospheric dispersion models (e.g., ARCON96, Ramsdell and Simonen, 1987) with similar capabilities may be used for dispersion calculations.

The atmospheric dispersion model provides a time history in the control room of the accidentally released toxic chemical concentration based on the assumed meteorological conditions and control room design parameters (i.e., control room free volume, normal and emergency ventilation rates, and time to isolate the control room). The meteorological condition assumed in such an analysis is the 5-percentile worst case dispersion condition at the site to provide a conservative assessment. The calculated chemical concentrations in the control room are then compared to health effects data (i.e., toxicity limit) appropriate to determine potential control room operator incapacitation. Regulatory Guide 1.78 recommends the use of the Immediately Dangerous to Life or Health (IDLH) concentration provided by the National Institute for Occupational Safety and Health (NIOSH, 1997)[30]. The IDLH is an atmospheric concentration of any toxic, corrosive or asphyxiate substance that poses an immediate threat to life or would cause irreversible or delayed adverse health effects or would interfere with an individual's ability to escape from a dangerous atmosphere based on a 30-minute exposure.

According to NUREG/CR-6944, "Next Generation Nuclear Plant Phenomena Identification and Ranking Tables (PIRTs)", Volume 6 [34], certain processes for the production of nuclear hydrogen may have large inventories of toxic chemicals. One example is the sulfur iodine process that uses the toxic chemicals hydrogen iodine (HI) and iodine (I_2). Under the appropriate accident conditions, a toxic gas release could be postulated. In addition, many corrosive chemicals are toxic but, in most cases, their corrosive characteristics dominate the hazard. Corrosive chemicals are a hazard to both equipment and operators whereas toxic chemicals are primarily a hazard to the control room operator. Ultimately, the impact on the control room operators will be dependent on the quantity of chemical released, the release phenomenology, meteorological conditions, and the toxicity of the chemical.

One method of mitigating potential adverse effects of accidental toxic chemical releases on control room operators is the use of instrumentation to detect the presence of the chemical at the control room air intake whereby the operators can take protective actions such as donning breathing apparatus or the control room may be automatically isolated upon detection. A protective design feature may include dual control room air intakes that are sufficiently separated so as to prevent a toxic plume from affecting both intakes simultaneously.

B1.10.2 Asphyxiant Gas Releases

Another potential source of concern is the use of gases at industrial facilities that are generally considered to be non-toxic but that could adversely affect human health and equipment operability by displacing sufficient atmospheric oxygen. One example includes pressurized helium fluid in the intermediate heat transfer loop whereby a helium leak in a confined space could create an environment incapable of sustaining life and in which personnel would be incapacitated within seconds. Other potential asphyxiants that could be found in a chemical process includes nitrogen and carbon dioxide. Displacing air with an inert gas also has the capability of incapacitating equipment. For example, if emergency diesel generators are needed for backup power in the event of a loss of offsite power, they may not be able to operate due to an inert gas plume at the air intake reducing oxygen content. Potential mitigation measures could include locating the inert gases far from the nuclear plant and keeping local inventories of these gases at a minimum.

The release phenomenology of the inert gas determines how far from the source of the leak asphyxiants can have an adverse effect on operators and equipment. If the gas or cryogenic liquid release has a vapor density greater than that of air, the heavier-than-air cloud can accumulate at ground level and travel well away from the source. Locating the inert-gas storage well away from the plant and minimizing vessel inventory are effective ways to reduce the risk of adverse effects of asphyxiating plumes on the nuclear plant personnel and equipment.

B1.10.3 Flammable Releases

According to NUREG/CR-6944 [34], most hydrogen is produced by steam reforming of natural gas. As this is an endothermic process for hydrogen production, about 30% of the natural gas is used to produce high-temperature heat to drive the chemical reactions. Because the process involves large quantities of natural gas, some nuclear hydrogen processes have the potential for large-scale release of flammable gases. In addition, high-temperatures from nuclear reactors are being considered as a heat source for oil refineries, shale oil and tar sands production facilities, and coal gasification and liquefaction processes. All of these involve the production of flammable gases and/or liquids in large quantities. Flammable fluids will be present in process streams at elevated pressures and at high (or low) temperatures. There will also inevitably be on-site storage of intermediates and products, possibly in large tanks.

The potential impacts on the safety-related reactor plant SSCs may involve a leak of flammable fluid resulting in an explosion with blast effects on the nuclear plant or a flammable fluid leak resulting in a fire with significant heat flux that could damage the nuclear plant. Another potential impact is operator injury or impairment from burns due to high heat fluxes.

Heat flux is generally the most serious hazard posed by a flammable release. While hydrogen flames give off relatively little thermal radiation, hydrocarbon flames have intense heat fluxes due to their high heat content. The accidental release of a flammable gas or vapor could also result in an unconfined vapor cloud explosion (UVCE). If the accidental release involves a liquid above its atmospheric boiling point, the result could be a boiling liquid expanding vapor explosion (BLEVE). Blast effects from an UVCE or BLEVE in the coupled chemical process plant could conceivably impact the nuclear plant through incident overpressure or debris missiles. The blast could occur either at the site of the released flammable chemical or at some point downwind as the chemical disperses to within the flammable concentration range given an ignition source (i.e., delayed ignition).

The NRC provides guidance on the evaluation of the impact of explosions in Regulatory Guide 1.91, "Evaluations of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plants" [36]. The method involves the determination of the distance from the potential release point to the plant beyond which the blast overpressure from an explosion is not likely to have an adverse impact. This distance is based on a peak positive incident overpressure of 1 pound per square inch (psi) below which no significant damage to SSCs of concern is expected by the NRC. The distance is a function of the 1/3rd power of the TNT-equivalent weight of the released substance.

The blast overpressure impacts, as well as the heat flux effects, can also be evaluated using the U. S. Environmental Protection Agency (EPA) computer model ALOHA (Areal Locations of Hazardous Atmospheres) [33]. ALOHA allows the user to estimate the concentrations due to downwind dispersion of a chemical cloud based on the physical characteristics of the released chemical, atmospheric conditions, and specific circumstances of the release. ALOHA was jointly produced by the National Oceanic and Atmospheric Administration (NOAA) Hazardous Materials Response and Assessment Division and the EPA Chemical Emergency Preparedness and Prevention Office (CEPPO). It was originally based on a simple model with a continuous point source with a Gaussian plume distribution but now also includes a heavy gas dispersion model. The heavy gas dispersion model used by ALOHA is the Dense Gas Dispersion (DEGADIS) model that was originally developed for the U.S. Coast Guard and the Gas Research Institute (GRI), primarily for simulation of the dispersion of cryogenic flammable gases.

As in the case of asphyxiate gas releases, flammable hazards can be can be mitigated by locating the flammable liquids far from the nuclear plant and by keeping local inventory at a minimum. According to a report prepared by Idaho National Laboratory [32] on separation requirements of a hydrogen production plant and high-temperature nuclear reactor, storage of 100 kg or less of hydrogen should kept at least 110 meters away from the nuclear plant without mitigation and 60 to 120 meters away if a blast barrier is constructed between the two facilities. The report also recommends that the major nuclear plant structures be placed below ground level as an effective mitigation measure.

Consideration should also be given to the fact that heat flux from hydrocarbon fires is more intense than from a .hydrogen flame and the flammability range (i.e., upper and lower flammability concentrations) is smaller for hydrocarbons than it is for hydrogen.

Again, the release phenomenology of the flammable gas or liquid determines how far from the source of the leak an explosion or fire can occur. When the gas density is greater than that of air, flammable concentrations can accumulate near ground level and travel away from the source. Plumes of flammable vapors can travel a considerable distance but the use of containment walls or other impediments to flow can reduce the risk. Locating storage tanks as far as possible from the plant and minimizing tank inventory are also effective ways to diminish the risk of plumes that could affect the nuclear plant.

B1.10.4 Oxygen Releases

As discussed in NUREG/CR-6944 [34], a nuclear hydrogen plant converts water into hydrogen and oxygen, except for nuclear heat for steam reforming of natural gas, with oxygen as the byproduct. The oxygen may be sold if there is a local market or it may be vented to the atmosphere. Oxygen that is sold may be transported to storage sites or to the customer by pipeline. In such cases, the site inventory will be limited to the inventories within the hydrogen process.

The potential effects of an accidental oxygen release include damage, wear, or impairment of the safety-related reactor plant SSCs due to the contact of oxygen enriched air with combustible materials leading to a fire or degrading equipment over time. Another potential impact is operator injury from burns caused by the increased flammability of combustible materials in the oxygen enriched environment.

Nuclear hydrogen plants will produce large quantities of pure oxygen that escapes from the chemical plant and cools as it is depressurized, potentially creating a heavy gas cloud that can flow at ground level to the reactor. In addition, spontaneous combustion becomes likely for many materials with pure oxygen and materials such as steel that are normally considered as noncombustible can burn in an oxygen rich atmosphere.

Another concern is related to the plant continuously releasing oxygen, leading to locally higher oxygen concentrations that may have secondary impacts. The oxygen may be stored in very large quantities for some applications. The important considerations in terms of nuclear plant safety are the storage quantities, temperatures, and pressures of the oxygen.

B1.10.5 Summary

The discussion of the application of Regulatory Guide 1.78 [35] requirements regarding the impact of postulated toxic chemical releases on control room habitability may not apply to the NGNP. Chapter 10 of the NGNP PCDR indicates that control room operators are not required to respond to design basis events and their actions are not credited in safety analyses. In this case, Criterion 19 of Appendix A of 10 CFR Part 50 [54] would not be applicable to the NGNP, precluding consideration of the requirements of Regulatory Guide 1.78 [35]. The consideration of the effects of accidental toxic chemical releases on the NGNP would be then be confined to adverse health effects on plant personnel due to the propagation of hazardous chemicals through secondary or tertiary systems to the reactor building or on plant equipment in the case of asphyxiates such as emergency diesel generators.

In regard to flammable/explosive chemicals, NRC Regulatory Guide 1.91, “Evaluations of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plants” [36] would apply to the NGNP. The guide provides a method acceptable to the NRC staff that involves the determination of the distance from the potential flammable/explosive chemical release point to the plant beyond which the blast overpressure from an explosion is not likely to have an adverse impact. This distance is based on a peak positive incident overpressure of 1 pound per square inch (psi) below which no significant damage to SSCs of concern is expected by the NRC. If the distance is smaller than that determined by the Regulatory Guide, further analysis of the effects of the flammable/explosive chemical would need to be conducted, possibly including a PRA.

Both toxic and flammable/explosive hazards can be mitigated by locating the hazardous chemicals far from the nuclear plant and by keeping local inventory at a minimum, as well as embedding the major nuclear plant structures below ground level. However, a potential drawback of embedment is that the NGNP could be more vulnerable to the intrusion of toxic and/or flammable heavier-than-air gases.

B2 EVALUATION AND RECOMMENDATION REGARDING REACTOR EMBEDMENT

B.2.1 OBJECTIVE RANKING

Table 23 below characterizes the various objective considerations discussed previously. Weights are assigned defining the relative importance of each objective. Items with higher weights are considered major drivers in the decision process. Items with lower weight are considered less significant as a discriminator in making the selection of embedment depth.

Table 23 Objective Ranking and Relative Weights

Objective	Weight
Design for Operations and Proper System Mechanical Functions	
Operational Needs (Safety equipment layout) (Note 1)	10
Safety, Investment Protection, and Security	
Heat Dissipation to Environment (Note 2)	5
Design Basis Threat and Malevolent Hazards (Note 3)	15
Natural Phenomena (Note 4)	15
Natural Geological Phenomena (Note 5)	10
Chemical Releases and Explosions (Note 6)	5
Cost and Construction Complexity	
Cost Benefit Consideration (Note 7)	10
Water Table Effects (Note 8)	10
Geotechnical Constraints and Foundation Performance (Note 9)	10
Construction Considerations (Note 10)	10
Total Weight Percent	40

Notes: 1) Design for Operational effects (Reactor protection, access for refueling, etc.) for normal and emergency operations, and maintenance activities. Relative locations of sub-systems with respect to reactor are considered. This is an important consideration. However, this weight is moderate because there is very little difference in the impact of operations on the selection of embedment.

2) The ability of the building to transfer heat to the environment either to ground or to ambient air will be assessed. The reason this weight assigned to this item is

low is that heat transfer through walls to the environment is not a design basis event, also it is shown the there is very little numerical difference expected between heat transfer to soil or to air. This is primarily due to the thermal inertia of the concrete citadel walls.

3) The element addresses protection against the design basis safeguards and security threat DBT and the beyond design threat imposed by a potential non accidental aircraft strike, or other BDBE. This item carries a high weight due to the importance of safeguards and security in the licensability of the plant.

4) Natural Phenomenon Hazards (NPH), such as tornado (including missiles) and flooding will are considered. This carries a high weight because its impact on safety and licensibility of the plant. It is common practice to design the exterior walls of a nuclear plant to accommodate these hazards.

5) The potential impacts that natural geological hazards such as dissolution features in soluble rock, weak compressible soils, slope instability resulting in landslide potential, and liquefaction and other earthquake induced flow phenomena are considered. This carries and moderate weight. The design challenges are not in surmountable regardless of embedment selection. Seismic response will be improved by a moderate amount by partial embedment.

6) Consideration for the hazards of chemicals stored and or transported in range of the Reactor, the potential effects, and mitigating features that could be utilized. This type of concern does often materialize with operating facilities located of active industrial areas. The potential for hydrogen explosions is addressed here. It is expected that robust structure required by other elements will facilitate design to accommodate this hazard. There are many options to control this hazard.

7) Cost benefit including key elements of relative capital cost will be developed for various embedment options and compared in a cost benefit table. This is a major element to consider and there is a very significant impact on cost associated with full embedment of the structure.

8) Groundwater levels are evaluated with regard to impact on structural design, construction dewatering schemes and need for waterproofing. This has a moderate impact on cost.

9) A range of foundation materials are considered with regard to static and dynamic properties and foundation performance. This has a moderate impact on cost.

10) The effects of the embedment depth on constructability using modern techniques such as modularization are considered. The excavation method used will require an evaluation of overburden thickness, elevation of competent rock, site space

constraints, and cost and schedule impacts. This element has a major impact on cost.

B2.2 LIST OF ALTERNATIVE SITES CONSIDERED

The following types of sites are typically found throughout the U.S. and will be discussed in this study. Rock sites typically have deep water tables. Soils sites with a deep water table exhibit similar characteristics to rock sites with respect to effect- or lack thereof of water. So the two selected site types are:

- Rock sites such as INL with a deep water table
- Deep soil sites with high water table

B2.3 ALTERNATIVE REACTOR EMBEDMENT CONCEPTS

The following alternatives are considered in this study for scoring and ranking purposes. It is expected that the optimal embedment depth will be developed in conceptual design and will be dependent on specific site conditions and ongoing analysis of operations needs.

- Minimal embedment of the building at approximately 7 to 10 meters
- Partial embedment 20 to 30 meters
- Full Embedment 60+ meters

B2.4 SCORING CRITERIA

The following range of scoring is used to establish relative ranking of the alternatives to satisfy objectives as shown in Tables 24 through 29 below.

Score of 0	Alternative does not satisfy objective
Score of 10	Alternative is clearly best suited to satisfy objective

Table 24 Alternative Ranking Rock Sites (Deep Water Table) Minimum Embedment

Objective	Weight	Alternative Satisfaction Score	Weighted Score
Design for Operations and Proper System Mechanical Functions			
Operational Needs (Safety equipment layout) (Note 1)	10	6	60
Safety, Investment Protection, and Security			
Heat Dissipation to Environment (Note 2)	5	6	30
Design Basis Threat and Malevolent Hazards (Note 3)	15	4	60
Natural Phenomena (Note 4)	15	4	60
Natural Geological Phenomena (Note 5)	10	4	40
Chemical Releases and Explosions (Note 6)	5	4	20
Cost and Construction Complexity			
Cost Benefit Consideration (Note 7)	10	8	80
Water Table Effects (Note 8)	10	5	50
Geotechnical Constraints and Foundation Performance (Note 9)	10	6	60
Construction Considerations (Note 10)	10	6	60
Total Weight Percent	40		250

Table 25 Alternative Ranking Rock Sites (Deep Water Table) Partial Embedment

Objective	Weight	Alternative Satisfaction Score	Weighted Score
Design for Operations and Proper System Mechanical Functions			
Operational Needs (Safety equipment layout) (Note 1)	10	8	80
Safety, Investment Protection, and Security			
Heat Dissipation to Environment (Note 2)	5	6	30
Design Basis Threat and Malevolent Hazards (Note 3)	15	6	90
Natural Phenomena (Note 4)	15	6	90
Natural Geological Phenomena (Note 5)	10	6	60
Chemical Releases and Explosions (Note 6)	5	6	30
Cost and Construction Complexity			
Cost Benefit Consideration (Note 7)	10	6	60
Water Table Effects (Note 8)	10	5	50
Geotechnical Constraints and Foundation Performance (Note 9)	10	5	50
Construction Considerations (Note 10)	10	8	80
Total Weight Percent	40		240

Table 26 Alternative Ranking Rock Sites (Deep Water Table) Full Embedment

Objective	Weight	Alternative Satisfaction Score	Weighted Score
Design for Operations and Proper System Mechanical Functions			
Operational Needs (Safety equipment layout) (Note 1)	10	5	50
Safety, Investment Protection, and Security			
Heat Dissipation to Environment (Note 2)	5	6	30
Design Basis Threat and Malevolent Hazards (Note 3)	15	7	105
Natural Phenomena (Note 4)	15	6	90
Natural Geological Phenomena (Note 5)	10	3	30
Chemical Releases and Explosions (Note 6)	5	4	20
Cost and Construction Complexity			
Cost Benefit Consideration (Note 7)	10	2	20
Water Table Effects (Note 8)	10	5	50
Geotechnical Constraints and Foundation Performance (Note 9)	10	4	40
Construction Considerations (Note 10)	10	2	20
Total Weight Percent	40		130

Table 27 Alternative Ranking Soil Sites (High Water Table) Minimal Embedment

Objective	Weight	Alternative Satisfaction Score	Weighted Score
Design for Operations and Proper System Mechanical Functions			
Operational Needs (Safety equipment layout) (Note 1)	10	6	60
Safety, Investment Protection, and Security			
Heat Dissipation to Environment (Note 2)	5	6	30
Design Basis Threat and Malevolent Hazards (Note 3)	15	4	60
Natural Phenomena (Note 4)	15	4	60
Natural Geological Phenomena (Note 5)	10	4	40
Chemical Releases and Explosions (Note 6)	5	4	20
Cost and Construction Complexity			
Cost Benefit Consideration (Note 7)	10	8	80
Water Table Effects (Note 8)	10	10	100
Geotechnical Constraints and Foundation Performance (Note 9)	10	8	80
Construction Considerations (Note 10)	10	6	60
Total Weight Percent	40		320

Table 28 Alternative Ranking Soil Sites (High Water Table) Partial Embedment

Objective	Weight	Alternative Satisfaction Score	Weighted Score
Design for Operations and Proper System Mechanical Functions			
Operational Needs (Safety equipment layout) (Note 1)	10	8	80
Safety, Investment Protection, and Security			
Heat Dissipation to Environment (Note 2)	5	6	30
Design Basis Threat and Malevolent Hazards (Note 3)	15	6	90
Natural Phenomena (Note 4)	15	6	90
Natural Geological Phenomena (Note 5)	10	6	60
Chemical Releases and Explosions (Note 6)	5	6	30
Cost and Construction Complexity			
Cost Benefit Consideration (Note 7)	10	6	60
Water Table Effects (Note 8)	10	7	70
Geotechnical Constraints and Foundation Performance (Note 9)	10	6	60
Construction Considerations (Note 10)	10	8	80
Total Weight Percent	40		270

Table 29 Alternative Ranking Soil Sites (High Water Table) Full Embedment

Objective	Weight	Alternative Satisfaction Score	Weighted Score
Design for Operations and Proper System Mechanical Functions			
Operational Needs (Safety equipment layout) (Note 1)	10	5	50
Safety, Investment Protection, and Security			
Heat Dissipation to Environment (Note 2)	5	6	30
Design Basis Threat and Malevolent Hazards (Note 3)	15	7	105
Natural Phenomena (Note 4)	15	6	90
Natural Geological Phenomena (Note 5)	10	3	30
Chemical Releases and Explosions (Note 6)	5	4	20
Cost and Construction Complexity			
Cost Benefit Consideration (Note 7)	10	2	20
Water Table Effects (Note 8)	10	0	0
Geotechnical Constraints and Foundation Performance (Note 9)	10	2	20
Construction Considerations (Note 10)	10	2	20
Total Weight Percent	40		60

B2.5 ALTERNATIVE RANKING SUMMARY

The following Table 30 provides a summary of scores for the design options from Table 24 through 29.

Table 30 Design Alternatives Ranking

Design Alternative	Total Weighted Score	Remarks
Rock Site Minimum Embedment	510	Maximum exposure to hazards
Rock Site Partial Embedment	610	Best Score for rock site with balance of protection from hazards and reduced cost
Rock Site Full Embedment	465	Water table is not a concern. Excavation and foundation complexity contributes to construction cost
Soil Site Minimum Embedment	580	Maximum exposure to hazards but Benefit with reduced water table concern
Soil Site Partial Embedment	650	Best score for soil site
Soil Site Full Embedment	395	Water table and foundation complexity contribute to very high costs

B2.6 ROLE OF EMBEDMENT TO SATISFY T&FRS

Part A of this study defines reactor building functions and requirements that must be fulfilled independently of the extent of embedment. It also provides a preliminary set of licensing basis events involving leaks and breaks in the PHTS and SHTS piping that the building must withstand. It evaluates a number of alternative design strategies for mitigating pipe breaks and minimizing radiological releases from the building. These options can all be applied to any level of reactor embedment. Hence the functions and requirements section is not dependent on the outcome or conclusions of the reactor embedment section.

The reactor embedment section is somewhat dependent on the results of functions and requirements section. The reactor embedment section also describes external hazards and physical security requirements that the building design must be able to accommodate. The capabilities of the reactor to protect against these hazards can be significantly influenced by the level of embedment.

A complete correlation between identified T&FRs and the consideration of reactor embedment is included in Table 31. This table shows the full list of T&FR, the role, if any, that embedment serves to accommodate these requirements, and the rational for selecting a particular embedment alternative. T&FRs also shown in Table 6 in the functions and requirements section.

Table 31 Role of Embedment in Satisfying Reactor Building Functions and Requirements

T&FR Number	Requirement	Role of Reactor Embedment
NHSB-1.0	The scope of functions and requirements for the reactor building shall be allocated to the Nuclear Heat Supply Building (NHSB), NHSS HVAC System, and NHSB Pressure Relief System (PRS) as described in the PCDR.	Not Applicable this is a General Requirement
NHSB-2.0	The Nuclear Heat Supply Building (NHSB) shall perform the following functions:	
NHSB-2.1	House and provide civil structural support for the reactor and all the SSCs within the scope of the NHSS (building geometry/space)	Partial embedment is favored with reduced seismic response for up to 50 percent embedment. Partial embedment also favorable for operation and equipment location. See section B1.1 and B1.9.
NHSB-2.1.1.	Provide personnel and equipment access for plant construction, maintenance, operation, and inspection of SSCs in the NHSS and for routine and emergency ingress and egress of plant workers	Partial embedment slightly favorable for operation and maintenance as well as for ease of construction. See section B1.1 and B1.5.
NHSB-2.1.2.	Provide radiation shielding for plant workers and the public during normal operation to keep radiation exposures ALARA	Not affected by embedment.
NHSB-2.1.3.	Limit air flow in neutron fields to keep routine releases of activation products and other releases during normal plant operation from all sources of radioactivity inside the NHSB ALARA	Not affected by embedment.

T&FR Number	Requirement	Role of Reactor Embedment
NHSB-2.1.4.	Maintain internal environmental conditions (temperature, humidity, air refresh) for NHSS SSCs and operators.	It could be speculated that embedment might reduce HVAC system loads with reduction of solar and transmission heat loads. HVAC system sizing and performance requirements are not usually driven significantly by external conditions. Concrete structures have substantial thermal inertial to dampen effects of external temperatures. HVAC flows are driven more by internal loads and contamination control needs.
NHSB-2.1.5.	Support zoning requirements for HVAC, radiation, fire protection, flood protection, and physical security protection	Not affected by embedment except for security. See item NHSB 3.1 for physical security discussion.
NHSB-2.1.6	The NGNP shall be designed to physically protect the safety related SSCs in the NHSS from hazards associated with the Hydrogen Production System, Power Conversion System and BOP facilities.	Not affected significantly by embedment. Further analysis will be required in conceptual design. See section B1.10.
NHSB-2.2	Resist all structural loads as required to support safety functions allocated to the NHSB	Partial embedment is favored for reduced seismic response for up to 50 percent embedment. See section B1.9.
NHSB-2.2.1.	Provide structural support for the reactor pressure vessel and its internals	Partial embedment is favored for reduced seismic response for up to 50 percent embedment. See section B1.9.
NHSB-2.2.2.	Provide structural support for the PHTS Helium Pressure Boundary (HPB) and the part of the SHTS HPB inside the NHSB	Partial embedment is favored for reduced seismic response for up to 50 percent embedment. See section B1.9.
NHSB-2.2.3.	Provide structural support and protection for the NSSS SSCs that contain sources of radioactive material outside the reactor vessel including the FHSS, HSS, and systems containing radioactive waste	Partial embedment is favored for reduced seismic response for up to 50 percent embedment. See section B1.9.

T&FR Number	Requirement	Role of Reactor Embedment
NHSB-2.2.4.	Protect the NHSB, PHTS HPB and other NHSS SSCs against loads imposed by faults including pipe breaks, chemical releases, fires, seismic failures, and explosions located in the HPS Acid Decomposer Building, SG Building, and other adjacent buildings associated with the HPS, PCS, and balance of plant. Protect the PHTS HPB from loads imposed by SHTS piping resulting from faults including structural failures in the HPS Acid Decomposer building and SG Building	Not affected significantly by embedment. See section B1.10. Analysis will be required per RG 1.78 and RG 1.91.
NHSB-2.2.5.	Provide structural support for all NHSS SSCs that provide a required safety function for all sources of radioactive material within the NHSB	Partial embedment is favored for reduced seismic response for up to 50 percent embedment. See section B1.9.
NHSB-2.2.5.1.	Provide structural support for SSCs required for confinement of radioactive material	Partial embedment is favored for reduced seismic response for up to 50 percent embedment. See section B1.9.
NHSB-2.2.5.2.	Provide structural support for SSCs required to maintain core and reactor pressure vessel geometry	Partial embedment is favored for reduced seismic response for up to 50 percent embedment. See section B1.9.
NHSB-2.2.5.3.	Provide structural support for SSCs required to control core heat removal including the core, reactor vessel, reactor cavity and RCCS	Partial embedment is favored for reduced seismic response for up to 50 percent embedment. See section B1.9.
NHSB-2.2.5.4.	Provide structural support for SSCs required for control of heat generation including the reactivity control rods	Partial embedment is favored for reduced seismic response for up to 50 percent embedment. See section B1.9.
NHSB-2.2.5.5.	Provide structural support for SSCs required for control of chemical attack	Partial embedment is favored for reduced seismic response for up to 50 percent embedment. See section B1.9.

T&FR Number	Requirement	Role of Reactor Embedment
NHSB-2.2.6.	Provide structural support for all NHSS SSCs that provide a supportive safety function for all identified LBEs as necessary to meet the TLRC (supportive safety function)	Partial embedment is favored for reduced seismic response for up to 50 percent embedment. See section B1.9.
NHSB-2.2.7.	Provide confinement of radioactive material during normal operation and during LBEs (AOOs, DBEs, and BDBEs) as necessary to meet the TLRC. This function is also supported by the NHSB PRS and NHSS HVAC (supportive safety function) (A design goal of a source term reduction factor of 10 is set in Section A3 for I-131 and Cs-137 for releases from the PHTS during design basis events involving DLOFC)	Leak tightness in not affected by level of embedment. Most significant leak paths are doors and HVAC dampers and valves. These will require leak tightness regardless of embedment level. Embedment does not contribute significantly to leak tightness.
NHSB-2.2.8.	Control building leakage and limit air ingress to the core following a large breach or breaches in the PHTS HPB, FHSS, and HSS for all identified LBEs as necessary to meet the TLRC (supportive safety function) (quantification of this requirement to be determined)	Leak tightness in not affected by level of embedment. Most significant leak paths are doors and HVAC dampers and valves. These will require leak tightness regardless of embedment level. Embedment does not contribute significantly to leak tightness.
NHSB-2.3	Protect the SSCs within the NHSS that perform safety functions from all internal and external hazards as identified in the LBEs as necessary to meet the TLRC	This requirement favors full embedment. However, protection is feasible and common for structures without full embedment. See Section B1.8 Natural Phenomenon Hazards
NHSB-2.3.1.	Protect the SSCs within the NHSS that perform required safety functions from LBEs involving PHTS and SHTS HPB leaks and breaks. This function is also supported by the NHSS Pressure Relief System.	Not affected by embedment. See section B1.1
NHSB-2.3.2.	Protect the SSCs within the NHSS that perform required safety functions from LBEs involving missiles from rotating machinery and other internal sources	Not affected by embedment.

T&FR Number	Requirement	Role of Reactor Embedment
NHSB-2.3.3.	Protect the SSCs within the NHSS that perform required safety functions from LBEs involving internal fires	Not affected by embedment.
NHSB-2.3.4.	Protect the SSCs within the NHSS that perform required safety functions from LBEs involving internal floods	Not affected by embedment.
NHSB-2.3.5.	Protect the SSCs within the NHSS that perform required safety functions from LBEs involving hydrogen process hazards	Not affected significantly by embedment. See section B1.10. Analysis will be required per RG 1.78 and RG 1.91.
NHSB-2.3.6.	Protect the SSCs within the NHSS that perform required safety functions from LBEs involving seismic events	Partial embedment is favored for reduced seismic response for up to 50 percent embedment. See section B1.9.
NHSB-2.3.7.	Protect the SSCs within the NHSS that perform required safety functions from LBEs involving accident aircraft crashes and transportation accidents	This requirement favors full embedment. However, protection is feasible and common for structures without full embedment. See Section B1.7 Malevolent Hazards
NHSB-2.3.8.	Protect the SSCs within the NHSS that perform required safety functions from LBEs involving high winds and wind generated missiles	This requirement favors full embedment. However, protection is feasible and common for structures without full embedment. See Section B1.8 Natural Phenomenon Hazards
NHSB-2.3.9.	Protect the SSCs within the NHSS that perform required safety functions from LBEs involving other internal and external hazards (requirements to be determined)	This requirement favors full embedment. However, protection is feasible and common for structures without full embedment. See Section B1.8 Natural Phenomenon Hazards
NHSB-2.3.10.	Protect the SSCs within the NHSS that perform supportive safety functions from LBEs as necessary to meet the top level regulatory criteria as necessary to meet the TLRC (requirements to be determined)	Not affected by embedment. See section B1.1
NHSB-2.4	Provide physical security of vital areas within the NHSB against acts of sabotage and terrorism	This requirement favors full embedment. However, protection is feasible and common for

T&FR Number	Requirement	Role of Reactor Embedment
		structures without full embedment. See Section B1.7 Malevolent Hazards
NHSB-3.0	The NHSB Pressure Relief System shall perform the following functions:	
NHSB-3.1	The PRS shall be compatible with the NHSB boundary functional requirement.	Not Applicable this is a General Requirement
NHSB-3.2	The PRS shall be designed, and the NHSB compartments shall be sized so that leaks and breaks up to 10 mm equivalent break size on the PHTS HPB or that part of the SHTS HPB inside the NHSB do not open the PRS so that HVAC filtration capability shall be continuously maintained	Not affected by embedment. It is expected that PRS will be vented to a release point at the top the building with or without filters.
NHSB-3.3	PRS shall open to prevent overpressure and thermal damage to any safety related SSC, remove the pressure driving force for radionuclide releases from the reactor building, and depending on the actual design that is chosen, may also be required to reclose to enable post-blow-down filtration for the following design basis event conditions.	Not affected by embedment. It is expected that PRS will be vented to a release point at the top the building with or without filters.
NHSB-3.3.1.	Breaks in the PHTS HPB in each NHSB compartment containing NHSS piping or vessels up to 100 mm equivalent single ended break size	Not affected by embedment. It is expected that PRS will be vented to a release point at the top the building with or without filters.
NHSB-3.3.2.	Breaks in the SHTS HPB in each NHSB compartment containing NHSS piping or vessels up to 1 meter equivalent single ended break size	Not affected by embedment
NHSB-3.3.3	Breaks in the HSS piping up to 100 mm equivalent single ended break size	Not affected by embedment

T&FR Number	Requirement	Role of Reactor Embedment
NHSB-3.4	The PRS shall open and prevent excessive damage to the NHSB in response to a Beyond Design Basis 1 meter break in the PHTS piping; excessive damage is defined by exceeding the TLRC using realistic assumptions	Not affected by embedment
NHSB-4.0	The NHSS HVAC System shall perform the following functions:	
NHSB-4.1	a) Maintain internal environmental conditions for all NHSS SSCs during normal operation, b) survive conditions for all AOO and DBE LBEs, and c) provide a post-event cleanup function. (supportive safety function)	It could be speculated that embedment might reduce HVAC system loads with reduction of solar and transmission heat loads. HVAC system sizing and performance requirements are not usually driven significantly by external conditions. Concrete structures have substantial thermal inertial to dampen effects of external temperatures. HVAC flows are driven more by internal loads and contamination control needs.
NHSB-4.2	Maintain environmental conditions for SSCs that perform supportive safety functions during LBEs (supportive safety function)	Not affected by embedment
NHSB-4.3	Perform radionuclide filtration functions as required to meet the TLRC for all LBEs (supportive safety function)	Not affected by embedment
NHSB-4.4	Perform protective functions to isolate the HVAC during DBE and BDBE HPB breaks to prevent damage from high temperatures and pressures in building compartments during depressurization to enable post blow-down filtration (supportive safety function)	Not affected by embedment

B2.7 RECOMMENDATIONS

The partial embedment scheme scores best for both Rock site and Soil site, and is therefore be recommended. The degree of partial embedment will be optimized for the site during the conceptual design phase. The case of soil site and a deep water table is not shown in ranking table but it is expected the scores would be essentially the same as for a rock site with deep water table.

The PBMR design is quite flexible to accommodate varying site geotechnical conditions. Access to the reactor building can be adjusted for a given site without a significant impact on the layout of major systems within the building.

B3 OPEN ISSUES AND ADDITIONAL R&D AND ENGINEERING STUDIES

1. Analysis will be required for quantities of explosive or flammable gasses and materials at or near the site including Hydrogen in order to meet requirements of Regulatory Guide 1.91. The intent will be to limit these quantities and maintain safe distances as required to prevent any substantial damage to the safety related nuclear heat supply system SSCs.
2. Review of life cycle costs associated with operations, inspections, maintenance, and energy costs, associated with HVAC loads, is recommended for the conceptual design.
3. Preparation of a Seismic Basis of Design Document is expected early in Conceptual Design to implement methodology promulgated by the current regulations.
4. There no R&D items identified in this portion of the study.

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BIBLIOGRAPHY

All references are provided in the references section of the report

DEFINITIONS

None

REQUIREMENTS

Not used

ASSUMPTIONS

1. For purposes of estimating cost, the reactor building is assumed to be 65.8 m high measured from the top of concrete mat to the roof. Two assumed configurations of structures with approximately equal volumes were considered.

   - A circular structure with a 56 m outside diameter
   - A square structure with 50 m long exterior sides

2. Two foundation conditions were assumed to bound the potential site geological conditions. The first condition assumed a rock site with a water table below the bottom of the reactor building mat under full embedment of the structure. This condition is comparable to the condition expected at the INL site. The second condition is a deep competent soil site with a high water table. This condition would be anticipated along the southeastern coast and Gulf coast of the United States.

3. Table 20 identifies the major systems to be housed in the Reactor Building, the relative safety classification and requirements for location with respect to the Reactor and other systems. The following safety classifications were taken from Reference [2] for input to this table. These safety classifications are assumed for the purposes of this study only in order to gain an understanding of the portion of the Reactor Building that will house SSCs requiring protection from internal and external hazards. Table 20 is not to be interpreted as a formal position on the SSC safety classification. Safety classification will be developed during conceptual and preliminary design based on the RI-PB safety analysis and licensing approach.

   Safety-Related SSCs (SR):

   This category is for SSCs relied on to perform required safety functions to mitigate the public consequences of Design Basis Events (DBEs) to comply with the dose limits of 10 CFR §50.34[8]

   This category is also for SSCs relied on to perform required safety functions to prevent the frequency of Beyond Design Basis Accidents (BDBEs) with consequences greater than the 10 CFR §50.34[8] dose limits from increasing into the DBE region.

   Non-Safety-Related with Special Treatment (NSR-ST):

   This category is for SSCs relied on to perform safety functions to mitigate the consequences of Anticipated Operational Occurrences (AOOs) to comply with the offsite dose limits of 10 CFR Part 20[79].

This category is also for SSCs relied on to perform safety functions to prevent the frequency of DBEs with consequences greater than the 10 CFR Part 20 [79] offsite dose limits from increasing into the AOO region.

Non-Safety-Related with No Special Treatment (NSR)

This category is for all SSCs not included in either of the above two categories.

4. For additional assumptions see “Key Assumptions” in section A3.3.

APPENDICES

APPENDIX A: 90 PERCENT DESIGN REVIEW PRESENTATION TO BEA

.

Reactor Building Functional and Technical Requirements

90% Design Review

September 5, 2008

Evaluation of Reactor Embedment

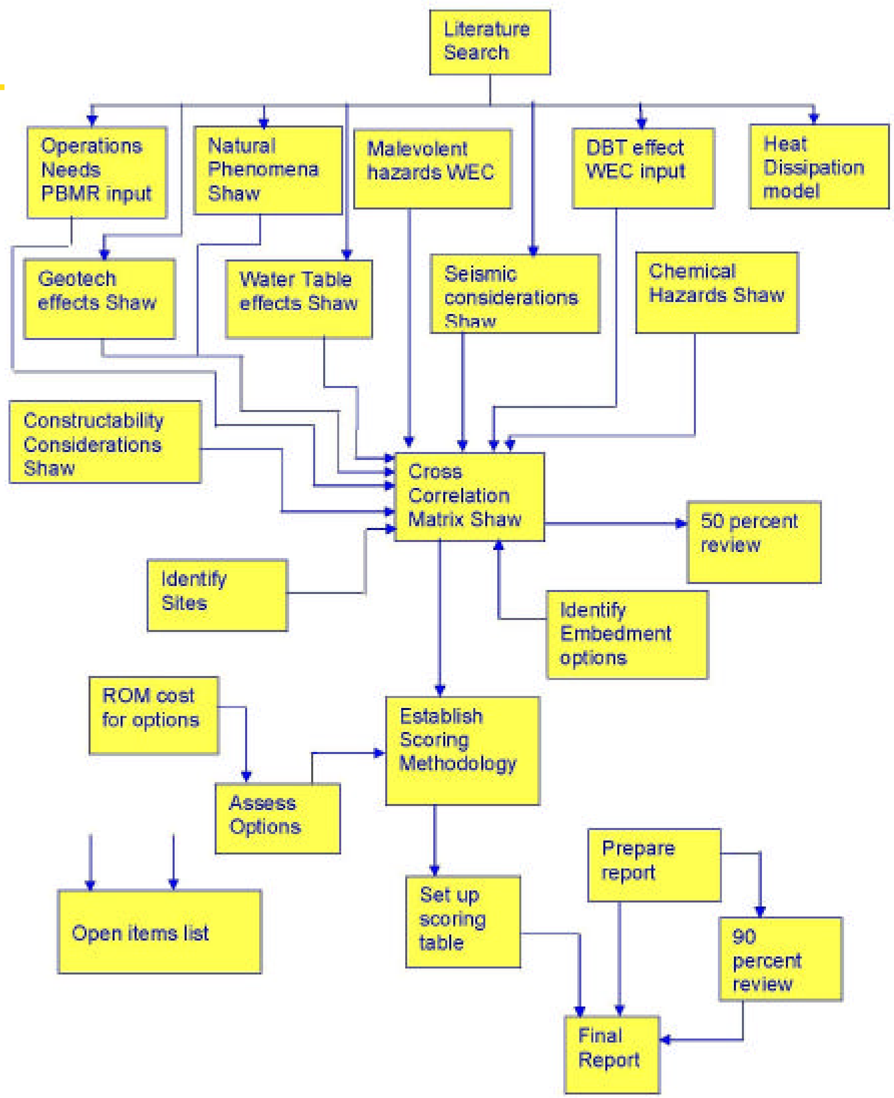
90% Design Review

September 5, 2008

Objectives of Study

- Review issues that are crucial for determining the reactor embedment depth
- Identify alternative Reactor embedment configurations applicable to a variety of site conditions including the INL site
- Develop an objective scoring method to evaluate and rank alternatives with respect to site conditions and embedment for INL and for soil sites
- Make recommendations of the preferred embedment configuration
- The study report of Reactor Embedment is to be combined as Part B with the report on Reactor Building Functions and Technical Requirements Part A. The relationship of the two parts will be discussed.

Flow Chart



Literature Search
Operations Needs PBMR input
Natural Phenomena Shaw
Malevolent hazards WEC
DBT effect WEC input
Heat Dissipation model
Geotech effects Shaw
Water Table effects Shaw
Seismic considerations Shaw
Chemical Hazards Shaw
Constructability Considerations Shaw
Cross Correlation Matrix Shaw
50 percent review
Identify Sites
Identify Embedment options
ROM cost for options
Establish Scoring Methodology
Assess Options
Prepare report
Open items list
Set up scoring table
90 percent review
Final Report

Contributors

- PBMR: Dirk Uys, Roger Young
- Westinghouse: Jim Winters
- Shaw: Peter Wells, Bob Wilmer, William Martin, Mike Davidson, Richard Deconto, Steve Vigeant, Paul Slatkavitz
- Technology Insights: Karl Fleming,

Issues relative to Reactor Embedment

- **Operational Needs**
  - Design for Operational effects (Reactor protection, access for refueling, etc.) for normal and emergency operations, DBEs, and maintenance activities. Relative locations of sub-systems with respect to reactor are considered.
- **Heat Dissipation to Environment**
  - The ability of the building to transfer heat to the environment either to ground or to ambient air will be assessed.
- **Water Table Effects**
  - Groundwater levels are evaluated with regard to impact on structural design, construction dewatering schemes and need for waterproofing.
- **Geotechnical Constraints and Foundation Performance**
  - A range of foundation materials are considered with regard to static and dynamic properties and foundation performance.
- **Construction Considerations**
  - The effects of the embedment depth on constructability using modern techniques such as modularization are considered. The excavation method used will require an evaluation of site space constraints, and cost and schedule impacts

Issues relative to Reactor Embedment (cont)

- **Cost Benefit**
  - Cost benefit including key elements of relative capital cost will be developed for various embedment options and compared in a cost table.
- **Malevolent Hazards**
  - This element addresses protection against the design basis safeguards and security threat DBT and the beyond design threat imposed by a potential non accidental aircraft strike, or other BDBE.
- **Natural Hazards**
  - Natural Phenomenon Hazards (NPH), such as tornado (including missiles) and flooding will are considered.
- **Geological Hazards**
  - The potential impacts that natural geological hazards such as dissolution features in soluble rock, weak compressible soils, slope instability resulting in landslide potential, and liquefaction and other earthquake induced flow phenomena are considered
- **Chemical Releases**
  - Consideration for the hazards of chemicals stored and or transported in range of the Reactor, the potential effects, and mitigating features that could be utilized. This type of concern does often materialize with operating facilities located of active industrial areas. The potential fro hydrogen explosions is addressed here.

Alternative Reactor Embedment Concepts

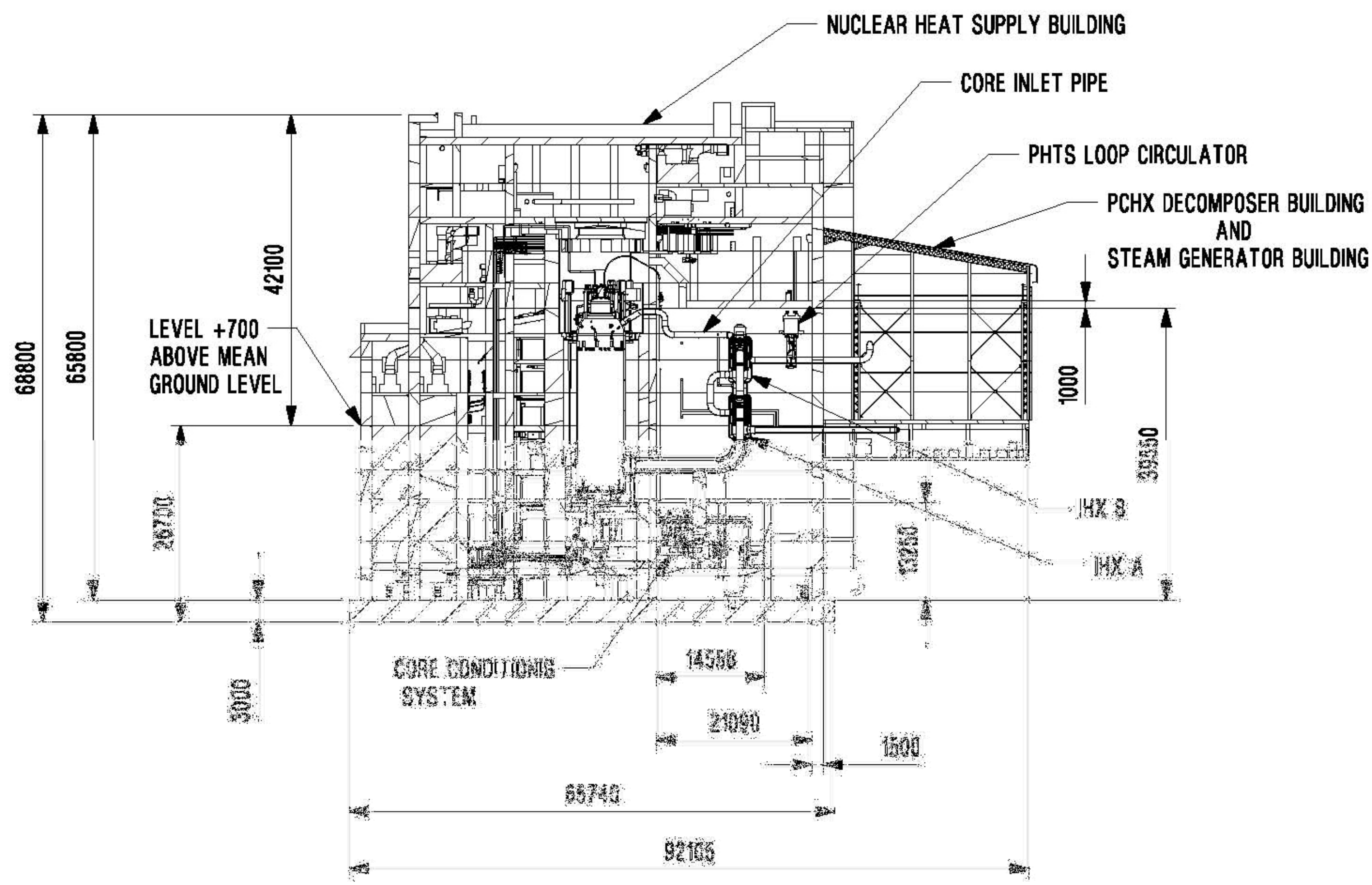
- Minimal embedment of the building 7 - 10 m
- Partial embedment 20 - 30 m
- Full Embedment >60 m

- Alternatives to be assessed for rock sites with deep water table similar to the INL site and soil site with shallow water table

Operations Needs

- Provide maintenance access to major equipment
- Provide safe personnel access and egress paths (elevators, corridors and stairwells)
- Enable placement of system components as needed relative to the elevation of the reactor itself, to meet thermal hydraulic and other operational needs.
- Provide protection of safety related components from external hazards and hazards due to internal SSC failures
- The PBMR design does not require routine access to the Reactor head for refueling activities. This issue drives other HTGR configurations to favor full embedment.
- Embedment could effect the arrangement of the Pressure Relief System with vent stack if required. It is expected that the PRS will discharge from the top of the building. All off site dose estimates used in this study Part assume ground level release.
- The general finding is that embedment decision is not driven significantly by operations needs. Access and maintenance activities are feasible with all options but would be slightly easier with a partially embedded configuration with reduced access and egress travel distances from point of entry to the building.
- A table is provided in the report showing relative location requirements of major systems

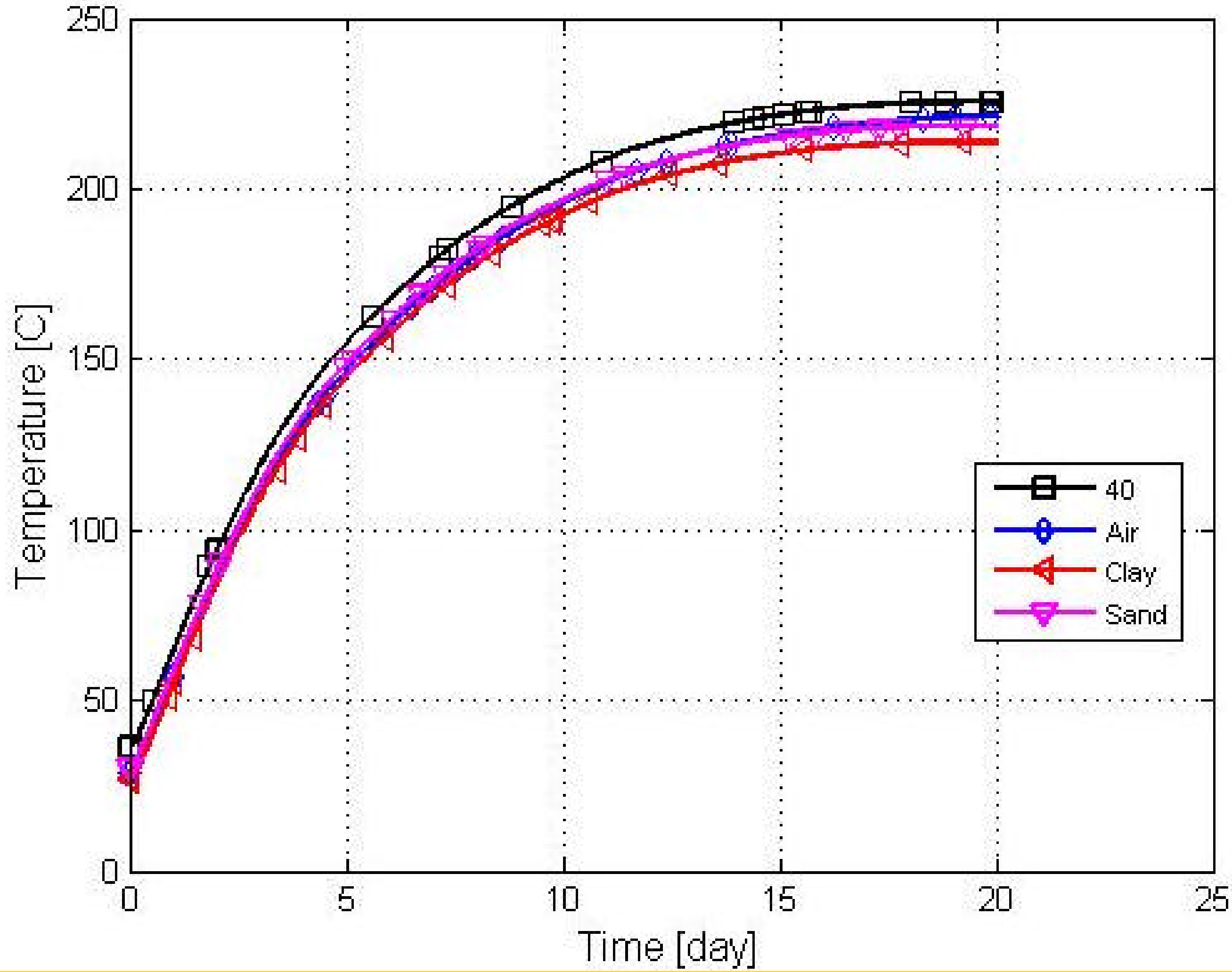
Initial NGNP Layout (Partial Embedment)



Heat Dissipation to Environment from the Reactor

- Normal Heat Transfer to the Environment is through RCCS and PHTS/SHTS, PCS, and HPS.
- DBE Heat transfer from the reactor is through RCCS. This scenario does not rely on heat transfer through the building walls.
- In the specific BDBE case where RCCS is also lost, the reactor decay heat is absorbed in the Reactor Citadel wall.
- Scoping calculations show that walls have so much thermal capacity that the reactor actually starts to cool down before the temperature of the outside of citadel wall rises significantly.
- PBMR has performed a simplified analysis to demonstrate that there is very little difference between heat transfer to Soils (Clay, Sand) and to Air outside the Citadel walls.
- General finding is that embedment decision is not driven significantly by concern for heat dissipation.

Average concrete temperature at height corresponding to middle of pebble bed



RMY5

Slide 11

RMY5 Peter,
Our models were adjusted and re-run as part of the review process and the results differ slightly, but not the outcome. I have inserted the new graph here for you.

The plot "40" is a comparative plot for the case where the outside temperature of the citadel is held fixed at 40 deg C

Roger M Young, 9/2/2008

Water Table Effects

- The water table at most sites will likely be encountered at depths less than about 15m, excluding arid regions
- For sites where the water table surface is above the NHSB foundation mat the following issues to be addressed:
  - **Dewater excavation. For very permeable soil or rock may need additional measures to control groundwater inflow**
  - **Obtain ground water withdrawal permits**
  - **Increases soil liquefaction risk**
  - **Design structures to resist uplift pressure (buoyancy) and static water pressure**
  - **Waterproof the foundation mat and exterior walls of the NHSB**
- General finding favors minimal embedment or partial embedment on sites with high water table.

Geotechnical Constraints and Foundation Performance

- Deep embedment requires generally more sophisticated excavation and foundation construction techniques.
- On Rock Sites controlled blasting is required with anchor bolting required at greater depths
- On soil sites sloped open cut is feasible for shallow excavations.
- On deep soil sites sheet piling and other cantilevered types are feasible for shallow excavations. Braced wall is required at greater depths.
- Engineered backfilling with lean concrete or compacted fill increases with depth.
- Resistance to overturning improves with depth but is negated by increase in buoyancy effect at greater depths.
- General finding favors partial embedment.

Construction Considerations

- Distance below Grade will directly impact the level of construction complexity
- To embed a structure on a site where solid rock is at or near the surface would start with drilling and blasting and rock removal to the depth of the bottom of the foundation. This is a costly and dangerous process
- Pumping systems are still required to remove water that may seep in through rock cracks or that is deposited by rains
- On soil sites, ground water and stability of the soils is of concern. Ground water must be pumped away from the excavation. Continuous pumping is well proven. Disposal of the water must be considered. Protection from cave-in is required. Side walls may require benching

Construction Considerations (cont.)

- Sheet piling to 12 m or other engineered deep excavation support techniques beyond 12 m may be required.
- Deep embedment will impact ability to use modular construction. Modules on lower level would be required to be complete for early installation.
- Modules and their components must be protected from significant hazards including weather and construction operations above including concrete placement since these modules would have to be placed in the structure before concrete floors are installed.

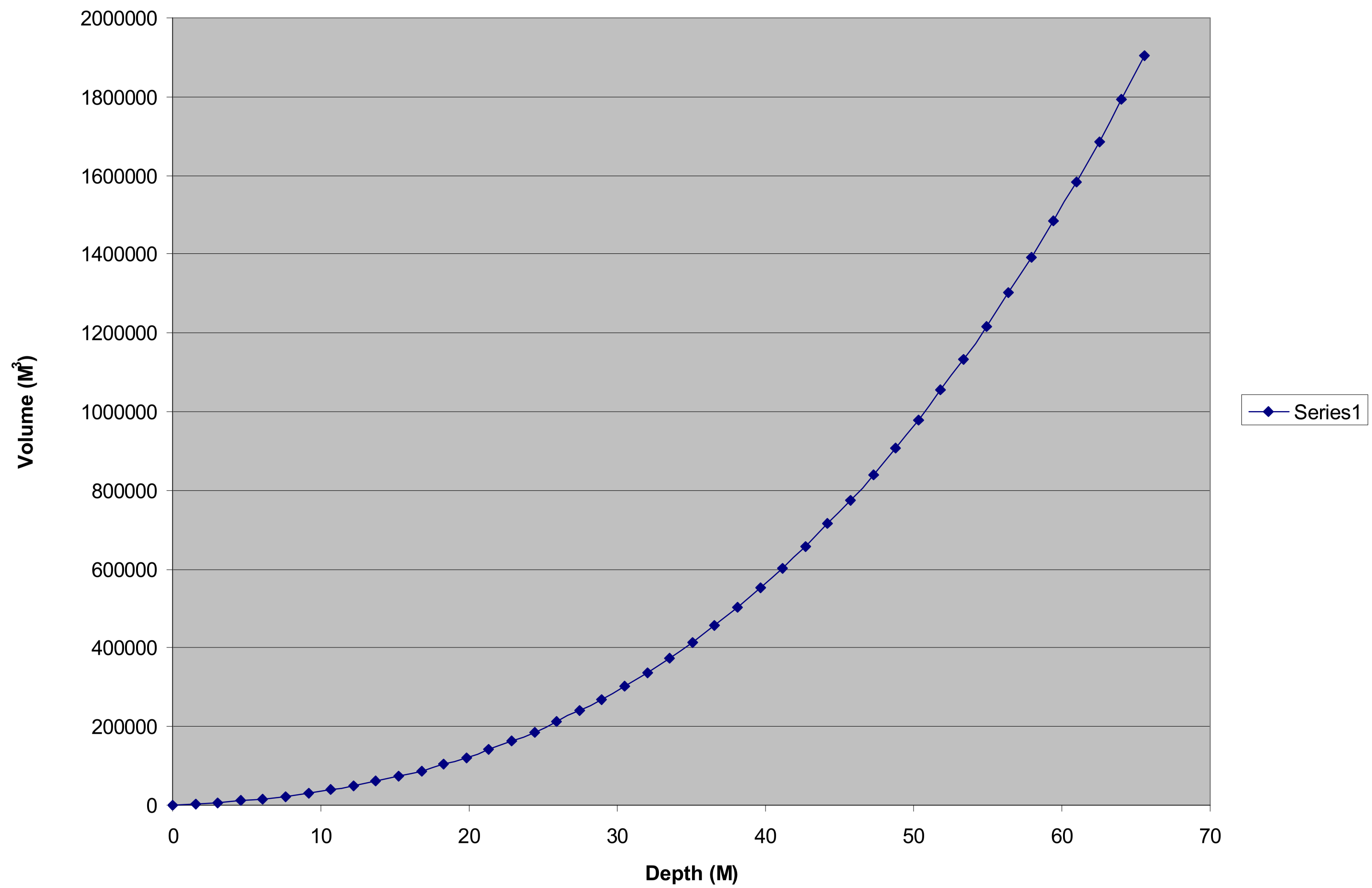
Construction Considerations (cont.)

- **Working below grade requires significant ventilation and continuous air monitoring to assure worker safety. Some welding processes require the use of inert gasses such as argon for a shielding medium. Argon gas is extremely dangerous in confined spaces since it displaces oxygen. Argon is also heavier than air and tends to find its way to the lowest spaces in the building volume thus presenting a potential hazard to workers in those areas. This problem is worst in a fully embedded case where construction openings would not be able to promote ventilation.**
- **In summary, construction complexity and cost increases exponentially with depth of embedment**
- **General finding significantly favors minimal embedment over partial and full embedment**

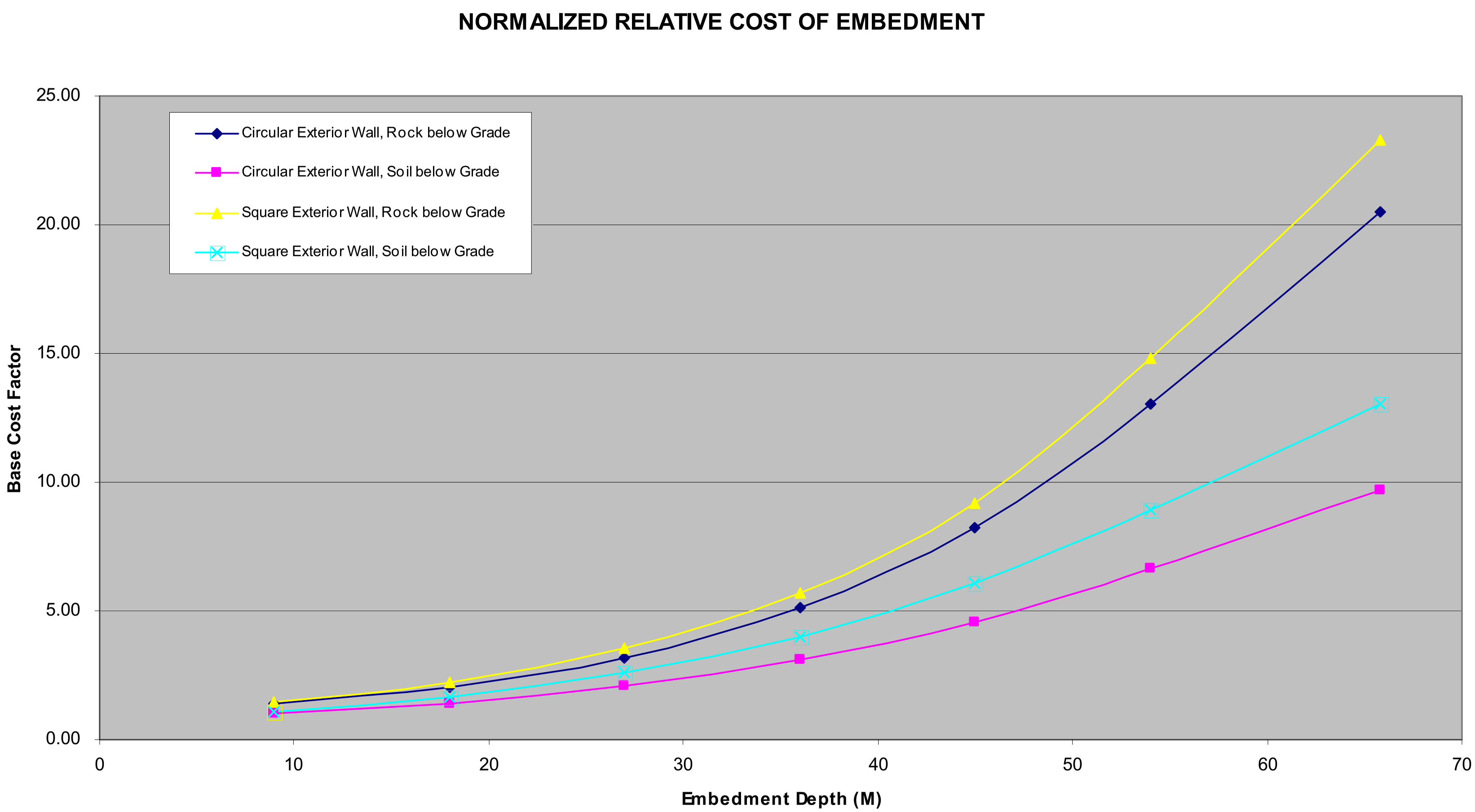
Cost Benefit Consideration

- In general it expected the geotechnical and foundation cost will increase exponentially with depth of embedment
- There is a trade off between heavy wall thicknesses required for security and resistance for natural phenomena above grade compared with increased wall thickness required to resist soil forces as embedment increases. This will be reviewed in more detail in the report.
- A comparison has been made of relative cost of the proposed embedment options for rectangular and circular structures and two site options, (Rock with low water table, and Deep soil with high water table).
- Key Cost differential elements are excavation, backfill, dewatering, subsurface foundation and exterior concrete walls, above and below ground.
- Initial finding is that Full embedment is significantly more costly.
- Deep excavation on a Rock site is more costly than on a soil site.

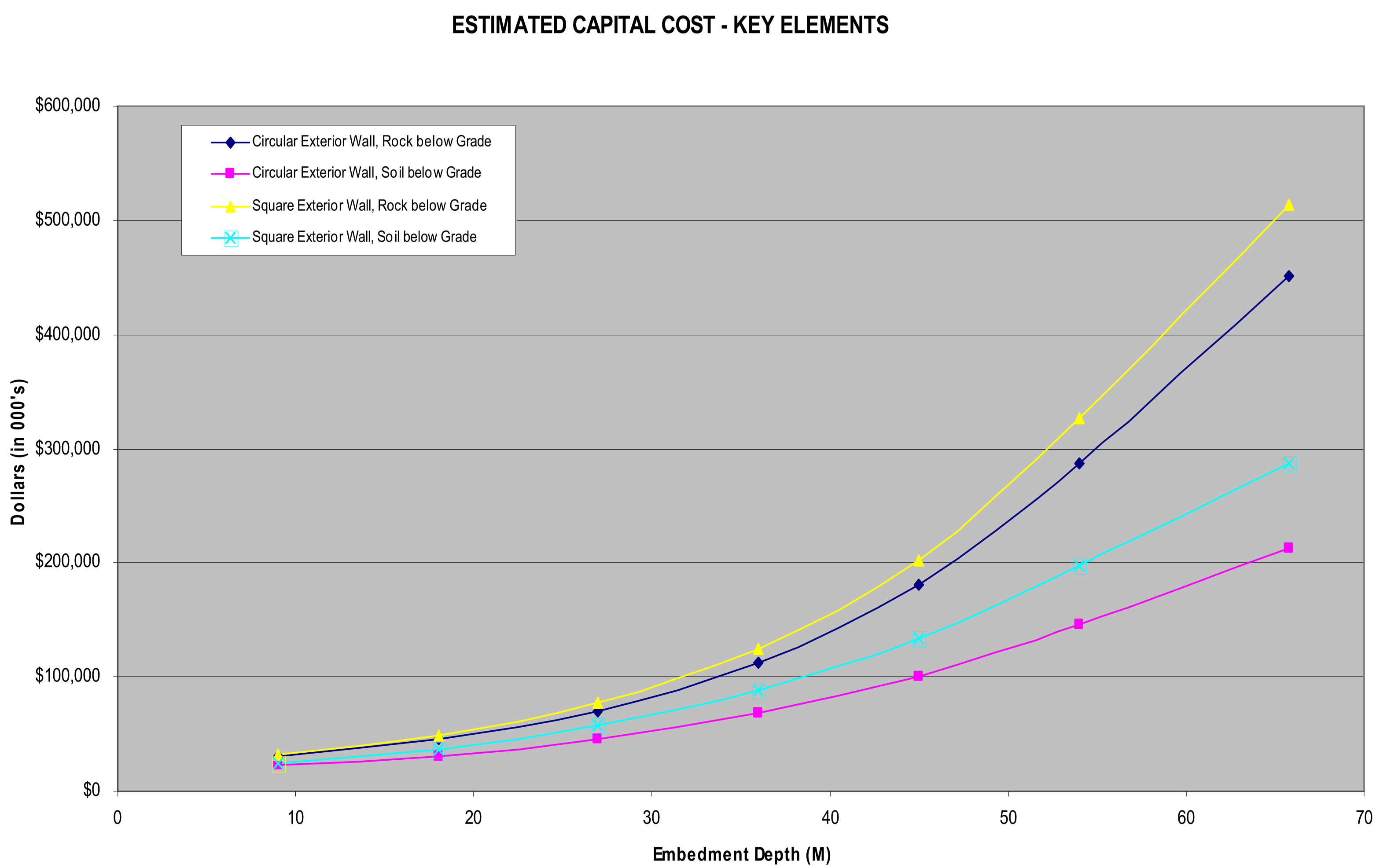
REACTOR BUILDING – OPEN CUT EXCAVATION EXCAVATION VOLUME vs. EMBEDDED DEPTH



NORMALIZED RELATIVE COST REACTOR BUILDING EMBEDMENT



ESTIMATED CAPITAL COST - KEY ELEMENTS REACTOR BUILDING EMBEDMENT



Design Basis Threat and Malevolent Hazards

- **Following Sept 11, 2001 NRC issued ICMs to better define security and protective requirements. These ICMs are generally safeguards information.**
- **Licensees are requested to assess issues such as large airplane crash, large fires, and explosions as beyond design basis events.**
- **Assessment guidelines are included in the "Methodology for Performing Aircraft Impact Assessments for "New Plant Designs" soon to be issued as NEI 08-13**
- **If the plant is totally underground, it presents no target for aircraft crash and no further assessment is needed other than for crash induced vibration effects on in-plant equipment.**
- **If portions are above grade, then a beyond design basis assessment must be made to show no significant radionuclide release (reactor core, new fuel, and spent fuel handling and storage systems)**

Design Basis Threat and Malevolent Hazards (cont.)

- Reactor buildings incorporating robust structural design for seismic and DBT resistance can generally show acceptable aircraft crash assessment results.
- Event mitigation features and licensee strategies must be provided to NRC during the Combined License application phase of plant deployment in accordance with proposed Rule 10CFR50.54(hh)2
- Large fires are defined as those greater than those defined in typical FHA. They affect large portions of the plant, well in excess of those having their origin from on-site sources. No explicit initiation mechanism is assumed for these fires. Historically, one source of these large fires is the release of jet fuel from a large commercial aircraft crash.

Design Basis Threat and Malevolent Hazards (cont.)

- Rules of thumb:
  - Minimize or eliminate the effective target area for aircraft crash
  - In addition to general structural robustness of the reactor building for seismic, enhance the penetration resistance of reactor building structures.
  - Maximize the inherent and passive fuel cooling and radiation containment features of the plant.
  - Maximize separation of redundant safety function initiation features.
  - Maximize separation of safety and defense in depth system features.
  - Ensure that alternate sources and means of fuel cooling are available following a beyond design basis threat event.
- Initial finding favors full embedment. But it is found that protection for seismic and protection from natural hazards would most likely provide adequate resistance.

Natural Phenomenon

1. Wind Loading will follow ASCE 7 and International Building Code. Hurricane winds of 65 m/s (3 second gusts) at 10 m above grade with importance factors assigned based on building function.
2. Recent R.G. 1.76 Tornado Loading Criteria are less severe for pressure drop and wind speed than URD. Use R.G. 1.76 for these conditions.

	EPRI-URD	RG-1.76
• Max Tornado Wind Speed	134 m/s	103 m/s
• Max Rotational Speed	107 m/s	82 m/s
• Max Translational Speed	27 m/s	21 m/s
• Radius of Max Rotational Speed	45.7 m	45.7 m
• Maximum Pressure Drop	138 mb	83 mb
• Rate of Pressure Drop	83 mb/sec	37 mb/sec

Natural Phenomenon

- Tornado Missiles per R.G. 1.76

Tornado Missiles per R.G. 1.76	Mass	Velocity
– **Sch 40 Pipe x 4.57m long**	**130 kg**	**41 m/s**
– **Automobile**	**1810 kg**	**41 m/s**
– **Solid sphere**	**0.0669 kg**	**8 m/s**

Notes:

1. **Velocity listed are in horizontal direction. Vertical velocities shall be 67 percent of the horizontal velocities.**
2. **Automobile to a height of 9.14 m only.**
3. **Region I missiles used.**
4. **Reinforced concrete walls about 0.75 m thick are typically adequate for tornado missiles.**

- Flood and Dam Break – Site to be chosen with grade to be 0.3 m above maximum Probable Flood including dam break .
- Seismic response spectra to be determined based on Seismic Probabilistic Hazard Assess. The URD proposes a generic SSE comprising a single ground motion spectrum conforming to a peak ground acceleration of 0.3 g.
- Finding favors embedment, but it is common to design of structures to resist these hazards.

Issues Relative to Seismic Analysis & Design

- **Seismic Design performed in accordance with EPRI-URD**
- **Seismic Classification System consistent with Nuclear Regulatory Guide 1.29.**
- **Seismic Category**
  - **Seismic Category I – This classification includes all structures, systems and components whose safety class is SC-1, SC-2, or SC-3. Seismic category I shall also include spent fuel storage pool structures including all fuel racks.**
  - **Seismic Category II – This classification applies to all plant structures, systems and components which perform no nuclear safety function but whose failure could degrade SC-1, SC-2, and SC-3 structures, systems and/or components.**
  - **Non-seismic – This classification includes all structures that do not fall into Seismic Category I or Seismic Category II structures.**
- **No OBE Load Case**
- **SSE comprising a single ground motion spectrum conforming to Regulatory Guide 1.60 anchored to a 0.3g peak ground acceleration. This peak ground acceleration will allow the NGNP to be constructed at most sites in the continental United States east of the Rocky Mountains.**
- **The design response spectra shall be in accordance with Regulatory Guide 1.60 – "Design Response Spectra for Seismic Design of Nuclear Power Plants" with a time history to envelop the design spectra.**

Issues Relative to Seismic Analysis & Design

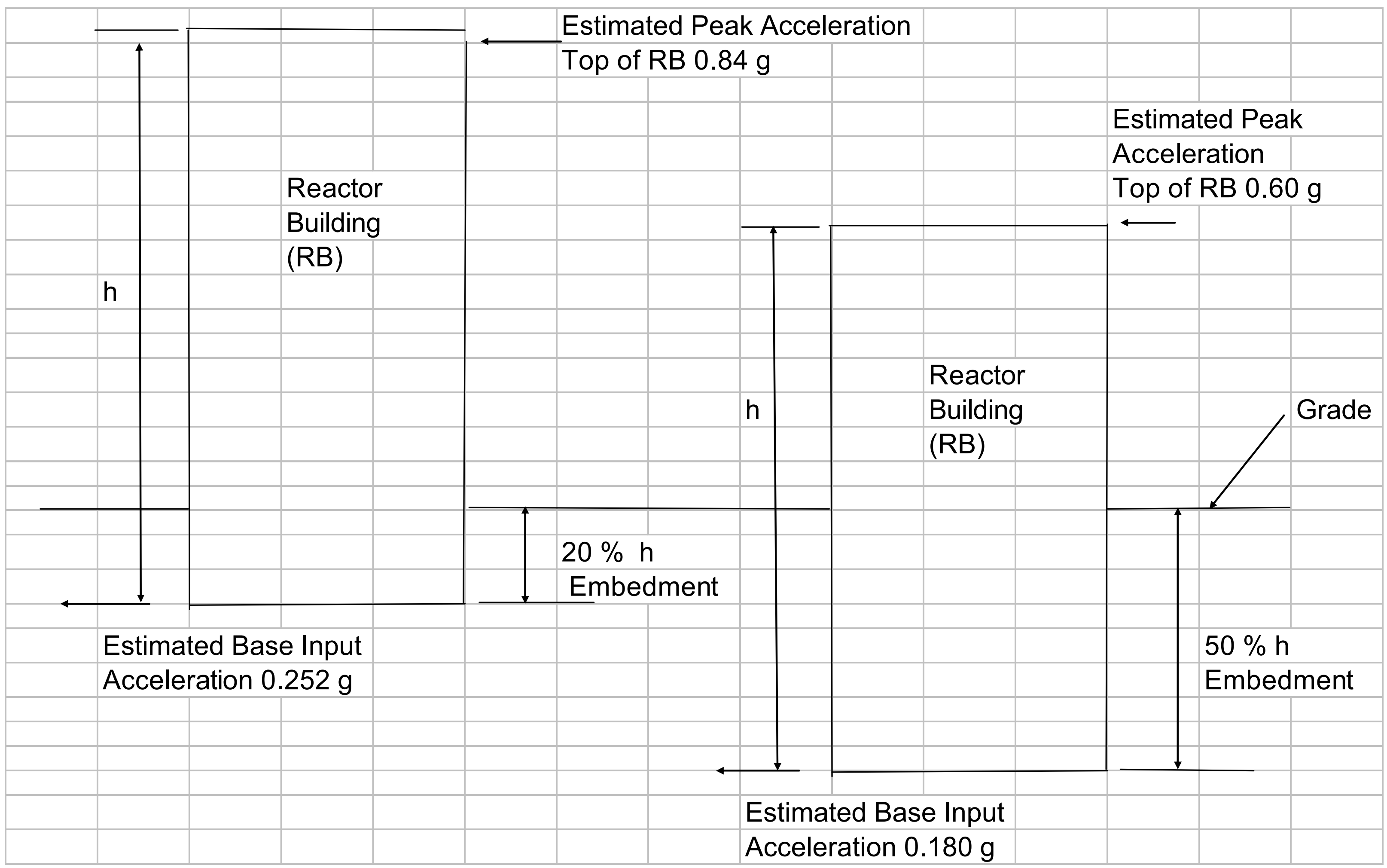
Analysis & Design

- **Design of Seismic Category I structures shall follow the ground motion characteristics specified previously, the analytical techniques specified in ASCE Standard 4-98 and meet all quality assurance requirements of 10 CFR, Appendix B. Design of Seismic Category I concrete structures shall be in accordance with ACI 349-06. Design of Seismic Category I steel structures shall be in accordance with AISC N690L-03.**
- **Design of seismic category II structures shall utilize the same analysis and design codes as those used for seismic category I structures; however, these structures may be constructed to the requirements of the non-nuclear building codes. In general, these requirements will invoke the International Building Code (IBC-06) or the version specified by the local building authority.**
- **Design of Seismic Category III structures shall follow the analytical requirements of IBC-06. IBC-06 specifies that ACI-318-05 by used for the design of concrete structures and that AISC – ASD/LRFD Steel Construction manual, 13th Edition be used for steel design.**

Estimated Input Base Acceleration of Reactor Building Due to Embedment Depth
0.3 g Ground Surface Acceleration - SOIL SITE

REACTOR BUILDING EMBEDMENT PERCENTANGE	ESTIMATED INPUT ACCELERATION AT BASE OF REACTOR BUILDING (g's)	ESTIMATED PEAK ACCELERATION TOP OF REACTOR BUILDING
0	0.300	1.0 (ASSUMED)
10	0.276	0.92
20	0.252	0.84
30	0.228	0.76
40	0.204	0.68
50	0.180	0.60
100	0.180	0.60

Estimated Input Base Acceleration of Reactor Building Due to Embedment Depth 0.3 g Ground Surface Acceleration - SOIL SITE



Natural Geological Phenomena

- **Seismic Hazards**
  - Earthquake hazard major consideration to site selection
  - Embedded mat foundations bearing on thick soil are subject to soil amplification.
  - Motion at the mat is greater than motion at top of bedrock.
  - Site ground motion established using NRC methods in 10 CFR 100.23 “Geologic & Seismic Siting Criteria”
  - Probabilistic methods for site seismic hazard in Regulatory Guide 1.208
  - Structural design in high seismic areas (e.g. parts of the western U.S. and in proximity to the New Madrid Zone in the Mississippi Valley) require more robust seismic restraints whether founded on rock or soil
  - Rock sites are not subject to amplification effects
  - Liquefaction of loose saturated soil can be triggered by earthquake
  - Rock sites are not subject to liquefaction

Natural Geological Phenomena (cont)

- **Geologic Hazards**
  - Avoid site locations that may be subject to:
    - Capable faults and faults exhibiting surface rupture
    - Karstic terrain formed by dissolution of soluble rocks/soil such as limestone, salt and gypsum
    - Surface collapse resulting from underground mining operations
    - Active or former slope instability
    - Landslide generated flood waves or seiche flood waves
    - Active or recent volcanism
    - Subsidence resulting from the extraction of groundwater or hydrocarbons
    - Future subsurface mining
    - High stress concentrations in rock
    - Weak and very compressible soil

Soil Structure Interaction

- An additional consideration of this study is to evaluate the effects of structure embedment on the seismic input ground motion at the base of the structure. In general, embedding the structure reduces the input motion at the base of the structure; thereby, reducing the seismic load. However, there are limits to this effect. Based on published literature, it is reasonably conservative to assume that embedment will not reduce the input base ground motion to less than approximately 60% of the surface ground motion. For the assumed reactor building, this limit is reached at a depth of approximately 30 m. In addition, ASCE 4 states that the top 6 m or one-half of the embedment depth, whichever is less, must be neglected in the soil structure interaction formulation. Based on the above, it is estimated that little or no reduction will be realized at embedments exceeding 36 m or about 50% embedment of the reactor building.
- It is not unrealistic to assume that the input motion at the base of the structure varies linearly from 100% at grade to 60% at 36 m.
- There are other factors that contribute to amplification of the peak ground acceleration. As an example, the stiffness and layer thickness of the soil significant impact the amplification factor.

Chemical Releases and Explosions

- Regulatory Guide 1.78 requires consideration of quantities greater than 100 pounds (45kg) of hazardous chemicals within 0.3 mile (483m) of the control room
- Stationary and Mobile Sources will be considered
- Dispersion modeling will be used to establish plume concentrations vs. distance from Source.
- Corrosive chemicals, Toxic gases, Asphyxiates, Flammable gases, Oxygen releases will be considered.
- NUREG/CR - 6944 “PIRTs” will be considered.
- Concern for heavier than air gas releases encroaching on reactor building.
- Concern for concentration of flammable gases in low lying locations

Chemical Releases and Explosions

- Concern for fires and deflagration event near the Reactor building site.
- Concern for clouds of explosive gases collecting and detonating in proximity to reactor building.
- Propagation of hazardous chemical through secondary or tertiary systems to the reactor building
- Explosion Analysis per R.G. 1.91 required to establish safe distance and quantities
- Initial Finding is that embedded building could be slightly more vulnerable to collection of heavier than air gases. It is expected that features required for other items will provide protection.

Summary of Findings

Technical Issue	Initial Finding	Relative Importance to Design Selection
Operational Needs (Safety equipment layout)	Partial embedment preferred (but not required) to allow quick access and egress	Minor to Moderate
Heat Dissipation to Environment	Difference in heat transfer not significant	Minor
Water Table Effects	Minimal embedment preferred on soil sites with high water table	Moderate
Geotechnical Constraints and Foundation Performance	Partial embedment preferred to balance resistance to overturning	Major cost driver
Construction Considerations	Minimal or partial embedment preferred	Major cost Driver

Summary of Findings

Technical Issue	Initial Finding	Relative Importance to Design Selection
Cost Benefit Consideration	Clearly favors less embedment	Major consideration
Design Basis Threat and Malevolent Hazards	Favors Embedment but other alternatives are feasible.	Major
Natural Phenomena	Favors Embedment to reduce hazard protection but other alternatives are feasible.	Moderate
Natural Geological Phenomena	Favors Embedment but expect less advantage at greater depths	Moderate important trade off
Chemical Releases and Explosions	Favors Minimal embedment due to concern for intrusion of heavier than air gases	Moderate

Objective Ranking

Objective	Weight
Design for Operations and Proper System Mechanical Functions	
Operational Needs (Safety equipment layout) (Note 1)	10
Safety, Investment Protection, and Security	
Heat Dissipation to Environment (Note 2)	5
Design Basis Threat and Malevolent Hazards (Note 3)	15
Natural Phenomena (Note 4)	15
Natural Geological Phenomena (Note 5)	10
Chemical Releases and Explosions (Note 6)	5
Cost and Construction Complexity	
Cost Benefit Consideration (Note 7)	10
Water Table Effects (Note 8)	10
Geotechnical Constraints and Foundation Performance (Note 9)	10
Construction Considerations (Note 10)	10
Total Weight Percent	100

Objective Ranking Weight Notes

1) Design for Operational effects (Reactor protection, access for refueling, etc.) for normal and emergency operations, DBEs, and maintenance activities. Relative locations of sub-systems with respect to reactor are considered.
2) The ability of the building to transfer heat to the environment either to ground or to ambient air are assessed.
3) The element addresses protection against the design basis safeguards and security threat DBT and the beyond design threat imposed by a potential non accidental aircraft strike, or other BDBE.
4) Natural Phenomenon Hazards (NPH), such as tornado (including missiles) and flooding are considered.
5) The potential impacts that natural geological hazards such as dissolution features in soluble rock, weak compressible soils, slope instability resulting in landslide potential, and liquefaction and other earthquake induced phenomena are considered
6) Consideration for the hazards of chemicals stored and or transported in range of the Reactor, the potential effects, and mitigating features that could be utilized. This type of concern does often materialize with operating facilities located of active industrial areas. The potential fro hydrogen explosions is addressed here.
7) Cost benefit including key elements of relative capital cost will be developed for various embedment options and compared in a cost benefit table.
8) Groundwater levels are evaluated with regard to impact on structural design, construction dewatering schemes and need for waterproofing.
9) A range of foundation materials is considered with regard to static and dynamic properties and foundation performance.
10) The effects of the embedment depth on constructability using modern techniques such as modularization are considered. The excavation method used will require an evaluation of overburden thickness, elevation of competent rock, site space constraints, and cost and schedule impacts

Objective Ranking Summary

Design Alternative	Total Weighted Score	Remarks
Rock Site Minimum Embedment	520	Max exposure to hazards
Rock Site Partial Embedment	620	Best Score
Rock Site Full Embedment	455	Less water table concern; Excavation Cost unfavorable
Soil Site Minimum Embedment	590	Benefit with reduced water table concern
Soil Site Partial Embedment	650	Best Score
Soil Site Full Embedment	385	Difficulties with water table and construction

Alternative Site Conditions

Rock Site

- Site is characterized by sound rock at or near grade (< 15 m deep)
- Foundation mat for the Nuclear Heat Source Building (NHSB) will be on sound rock.
- Full embedment of the NHSB will require a deep excavation in rock (45 – 60 m)
- Minimal or partial embedment may be preferred due to:
  - lower excavation cost
  - reduced need for groundwater control (dewatering and waterproofing)
  - reduced storm water control issues during construction
- Seismic design is simplified (soil-structure interaction analysis not required) for rock with shear wave velocity >2800 m/s
- Liquefaction is not an issue for NHSB but, may be for at-grade structures on soil
- Regional Considerations: Rock sites most likely to be found in upland areas of NE, SE, NW and S Central US.

Alternative Site Conditions (cont.)

Soil over Shallow Rock

- Site characterized by soil overlying sound rock at 15 – 50 m depth
- Foundation mat for the NHSB on sound rock
- Full embedment will require excavation in soil and, possibly, in rock
- Soil excavation will have max side slopes of 1.5H:1V or 2H:1V requiring a much larger excavation as the rock founding layer becomes deeper (or embedment greater). Excavation in rock may be near-vertical; but, may require engineered support
- Minimal or partial embedment may be preferred due to:
  - lower excavation cost
  - reduced need for groundwater control (dewatering and waterproofing)
  - reduced storm water control issues during construction
- Seismic design is simplified (soil-structure interaction analysis not required) for rock with shear wave velocity >2800 m/s
- Liquefaction is not an issue for NHSB; but may be for at-grade structures on soil
- Regional Considerations: Soil over shallow rock sites may be found in most areas of the US, inland of the Atlantic and Gulf Coastal Plains and away from the higher upland areas.

Alternative Site Conditions (cont.)

- **Deep Soil**
  - Site is characterized by a competent soil bearing layer within 60 m of ground surface. Rock is much deeper.
  - Foundation mat for the NHSB will be on the competent soil layer
  - Optimal embedment depth will tend to be determined by the depth to the competent soil layer
  - Soil excavation will have max side slopes of 1.5H:1V or 2H:1V requiring a much larger excavation as the founding layer becomes deeper (or embedment is greater)
  - Minimal or partial embedment may be preferred due to:
  - - lower excavation cost
  - - reduced need for groundwater control (dewatering and waterproofing)
  - - reduced storm water control issues during construction
  - Soil liquefaction during earthquake may be a problem for NHSB and at-grade structures
  - Greater embedment depth may provide increased foundation bearing capacity and reduced risk of liquefaction
  - Seismic design will require soil-structure interaction analysis
  - Regional Considerations: Deep soil sites are typical of the Atlantic and Gulf Coastal Plains, and intermountain alluvial valleys (primarily in the western US)

Recommendations

- The partial embedment scheme scores best for both Rock site and Soil site, and will therefore be recommended. The degree of partial embedment will be optimized for the site during the conceptual design phase. The case of a soil site with a deep water table is not shown in ranking table but it is expected the scores would be essentially the same as for a rock site with deep water table.
- The PBMR design is quite flexible to accommodate varying site geotechnical conditions. Access to the reactor building can be adjusted for a given site without a significant impact on the layout of major systems within the building.

Role of Reactor Embedment in Satisfying Reactor Building Functions and Requirements

- The Reactor Building study is conducted in two parts, Reactor Building Functions and Requirements and Reactor Embedment
- The Reactor Building Functions and Requirements portion of this study defines reactor building functions and requirements
- Functions and Requirements must be fulfilled independently of the extent of embedment. The functions and requirements portion of this study also provides as a preliminary set of licensing basis events involving leaks and breaks in the PHTS and SHTS piping that the building must withstand. The functions and requirements section also evaluates a number of alternative design strategies for mitigating pipe breaks and minimizing radiological releases from the building. These options can all be applied to any level of reactor embedment. Hence the functions and requirements section is not dependent on the outcome or conclusions of the reactor embedment section.

Role of Reactor Embedment in Satisfying Reactor Building Functions and Requirements (cont.)

- The reactor embedment section is somewhat dependent on the results of functions and requirements section but it is concluded that considerations that have been included in the reactor embedment section are to a major extent related to ensuring that functional needs and requirements identified in functions and requirements section can be met. The reactor embedment section also defines a set of external hazards and physical security requirements that the building design must fulfill, the capabilities for which are significantly influenced by the level of embedment.
- A further level of synergy between functions and requirements section and the reactor embedment section is also provided by the use of a common set of criteria for evaluating the various design strategies for mitigating pipe breaks in the functions and requirements section and the evaluating the alternative embedment options in the reactor embedment section. These evaluation criteria are derived from the functions and requirements section functional requirements.

Role of Reactor Embedment in Satisfying Reactor Building Functions and Requirements (cont.)

Functional needs to be met by the reactor building are to:

- House and provide civil structural support for the reactor and all the SSCs within the scope of the NHSS (building geometry/space)
  - This is addressed by discussion of foundation and seismic considerations and operation needs to house SSCs with proper orientation with respect to the reactor.
- Resist all structural loads as required to support both required and supportive safety functions allocated to the NHSB
  - This is addressed by discussion of foundation and seismic considerations
- Protect the SSCs within the NHSS that perform safety functions from all internal and external hazards as identified in the LBEs as necessary to meet the TLRC
  - This is addressed in definition of various pipe break scenarios in the functions and requirements section and the discussion of Natural and Geological hazards in The reactor embedment section
- Provide physical security of vital areas within the NHSB against acts of sabotage and terrorism
  - This is addressed in the discussion of Malevolent Hazards of design basis threats and beyond design basis hazards such as a non accidental aircraft strike.
- Provide radionuclide retention during blow-down and delayed fuel release phases of depressurization events.
  - This is addressed in the discussion of operational requirements noting the level of embedment can affect the arrangement of a vent stack if required. An elevated vent stack is not accounted for in determining dose rates at the EAB. Ground level release is assumed for all alternative schemes.

Back Up Slides

- Scoring Tables
  - Rock Sites (Deep Water Table)
    - Minimal Embedment
    - Partial Embedment
    - Full Embedment
  - Soil Sites (High Water Table)
    - Minimal Embedment
    - Partial Embedment
    - Full Embedment

Objective	Weight	Alternative Satisfaction Score	Weighted Score
Rock Minimum Embedment Alternative			
Design for Operations and Proper System Mechanical Functions			**0**
Operational Needs (Safety equipment layout) (Note 1)	10	6	60
Safety, Investment Protection, and Security			**0**
Heat Dissipation to Environment (Note 2)	5	6	30
Design Basis Threat and Malevolent Hazards (Note 3)	15	4	60
Natural Phenomena (Note 4)	15	4	60
Natural Geological Phenomena (Note 5)	10	4	40
Chemical Releases and Explosions (Note 6)	5	4	20
Cost and Construction Complexity			**0**
Cost Benefit Consideration (Note 7)	10	8	80
Water Table Effects (Note 8)	10	5	50
Geotechnical Constraints and Foundation Performance (Note 9)	10	6	60
Construction Considerations (Note 10)	10	6	60
Total Weight Percent	100		520

Rock Partial Embedment Alternative			
Design for Operations and Proper System Mechanical Functions			**0**
Operational Needs (Safety equipment layout) (Note 1)	10	8	80
Safety, Investment Protection, and Security			**0**
Heat Dissipation to Environment (Note 2)	5	6	30
Design Basis Threat and Malevolent Hazards (Note 3)	15	6	90
Natural Phenomena (Note 4)	15	6	90
Natural Geological Phenomena (Note 5)	10	6	60
Chemical Releases and Explosions (Note 6)	5	6	30
Cost and Construction Complexity			**0**
Cost Benefit Consideration (Note 7)	10	6	60
Water Table Effects (Note 8)	10	5	50
Geotechnical Constraints and Foundation Performance (Note 9)	10	5	50
Construction Considerations (Note 10)	10	8	80
Total Weight Percent	100		620

Rock Full Embedment Alternative			
Design for Operations and Proper System Mechanical Functions			**0**
Operational Needs (Safety equipment layout) (Note 1)	10	5	50
Safety, Investment Protection, and Security			**0**
Heat Dissipation to Environment (Note 2)	5	6	30
Design Basis Threat and Malevolent Hazards (Note 3)	15	7	105
Natural Phenomena (Note 4)	15	6	90
Natural Geological Phenomena (Note 5)	10	3	30
Chemical Releases and Explosions (Note 6)	5	4	20
Cost and Construction Complexity			**0**
Cost Benefit Consideration (Note 7)	10	2	20
Water Table Effects (Note 8)	10	5	50
Geotechnical Constraints and Foundation Performance (Note 9)	10	4	40
Construction Considerations (Note 10)	10	2	20
Total Weight Percent	100		455

Objective	Weight	Alternative Satisfaction Score	Weighted Score
Soil Minimum Embedment Alternative			
Design for Operations and Proper System Mechanical Functions			**0**
Operational Needs (Safety equipment layout) (Note 1)	10	6	60
Safety, Investment Protection, and Security			**0**
Heat Dissipation to Environment (Note 2)	5	6	30
Design Basis Threat and Malevolent Hazards (Note 3)	15	4	60
Natural Phenomena (Note 4)	15	4	60
Natural Geological Phenomena (Note 5)	10	4	40
Chemical Releases and Explosions (Note 6)	5	4	20
Cost and Construction Complexity			**0**
Cost Benefit Consideration (Note 7)	10	8	80
Water Table Effects (Note 8)	10	10	100
Geotechnical Constraints and Foundation Performance (Note 9)	10	8	80
Construction Considerations (Note 10)	10	6	60
Total Weight Percent	100		590

Soil Partial Embedment Alternative			
Design for Operations and Proper System Mechanical Functions			**0**
Operational Needs (Safety equipment layout) (Note 1)	10	8	80
Safety, Investment Protection, and Security			**0**
Heat Dissipation to Environment (Note 2)	5	6	30
Design Basis Threat and Malevolent Hazards (Note 3)	15	6	90
Natural Phenomena (Note 4)	15	6	90
Natural Geological Phenomena (Note 5)	10	6	60
Chemical Releases and Explosions (Note 6)	5	6	30
Cost and Construction Complexity			**0**
Cost Benefit Consideration (Note 7)	10	6	60
Water Table Effects (Note 8)	10	7	70
Geotechnical Constraints and Foundation Performance (Note 9)	10	6	60
Construction Considerations (Note 10)	10	8	80
Total Weight Percent	100		650

Soil Full Embedment Alternative			
Design for Operations and Proper System Mechanical Functions			**0**
Operational Needs (Safety equipment layout) (Note 1)	10	5	50
Safety, Investment Protection, and Security			**0**
Heat Dissipation to Environment (Note 2)	5	6	30
Design Basis Threat and Malevolent Hazards (Note 3)	15	7	105
Natural Phenomena (Note 4)	15	6	90
Natural Geological Phenomena (Note 5)	10	3	30
Chemical Releases and Explosions (Note 6)	5	4	20
Cost and Construction Complexity			**0**
Cost Benefit Consideration (Note 7)	10	2	20
Water Table Effects (Note 8)	10	0	0
Geotechnical Constraints and Foundation Performance (Note 9)	10	2	20
Construction Considerations (Note 10)	10	2	20
Total Weight Percent	100		385

Discussion Topics

- Introduction
  - Objectives and Scope
  - Approach
  - Relationship with Reactor Embedment
- Review of Functions and Requirements
- Evaluation of Alternative Building Design Strategies
  - Evaluation Criteria
  - Definition of Alternative Design Strategies
  - Evaluation of Radionuclide Retention and Pressure Capacity
  - Evaluation of Costs
  - Integrated Evaluation vs. Functional Requirements
  - Allocation of Radionuclide Retention Capability
- Open Issues and Additional R&D and Engineering Studies
- Conclusions and Recommendations

Overview

- Study structured into two complementary sections:
  - Reactor Building Technical and Functional Requirements
    - Overall requirements for all design alternatives
    - Definition of preliminary licensing basis events with focus on HPB leaks and breaks
    - Evaluation of alternative design approaches to meeting reactor building safety functions independent of embedment
  - Evaluation of Reactor Building Embedment
    - Definition of preliminary licensing basis events involving external hazards and physical security threats
    - Evaluation of alternative approaches to reactor embedment independent of reactor building design alternatives
- Integrated evaluation and recommendations for consideration in Conceptual Design of NGNP RB

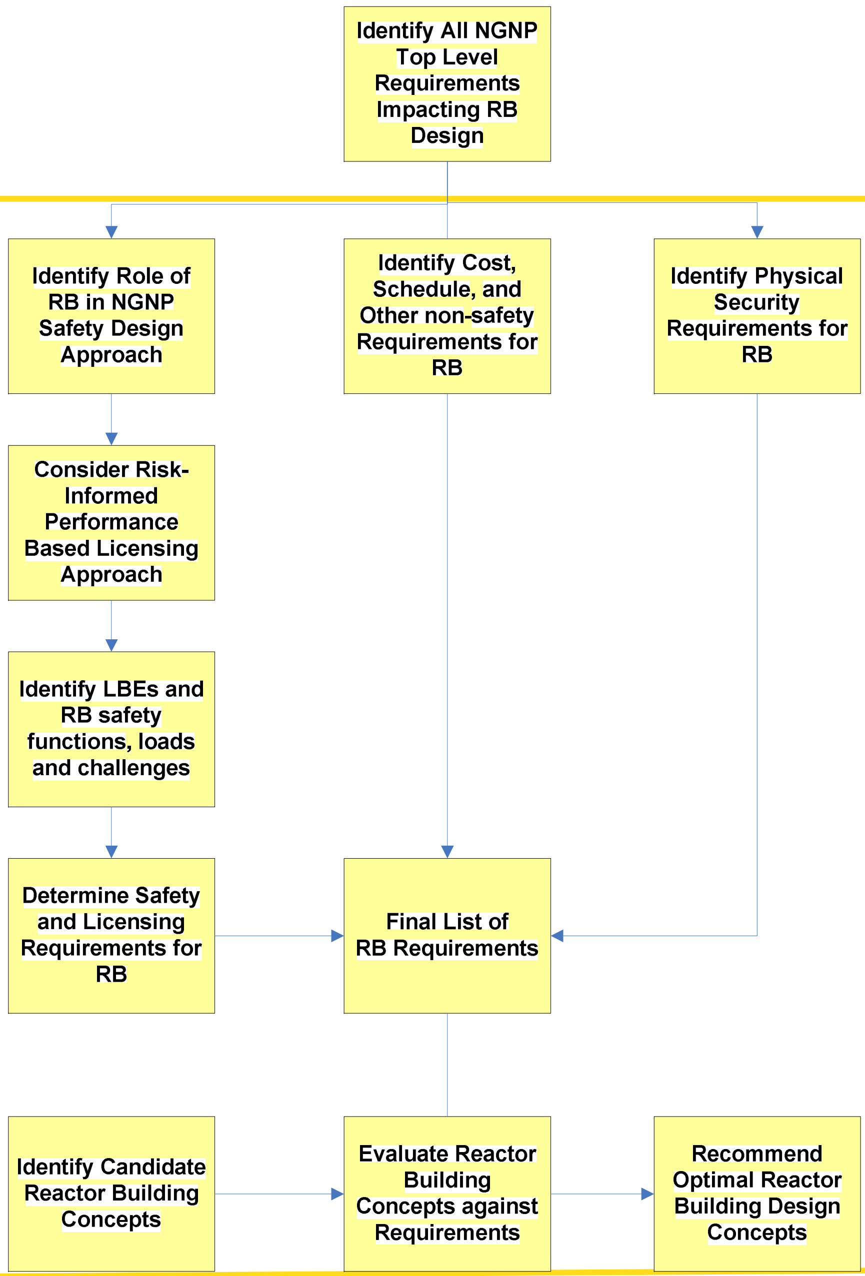
Key Contributors to Reactor Building Section

- PBMR:
  - Roger Young, Dirk Uys, Nilen Naidu, Johan Strauss, Wilma van Eck
  - Shaw:
  - Peter Wells, Bob Wilmer, Nicholas Menounos, Michael Davidson
- Technology Insights:
  - Karl Fleming, Fred Silady, Dave Dilling, Fred Torri, Lori Mascaro
- Westinghouse:
  - Jim Winters

Objectives and Scope of Reactor Building Section

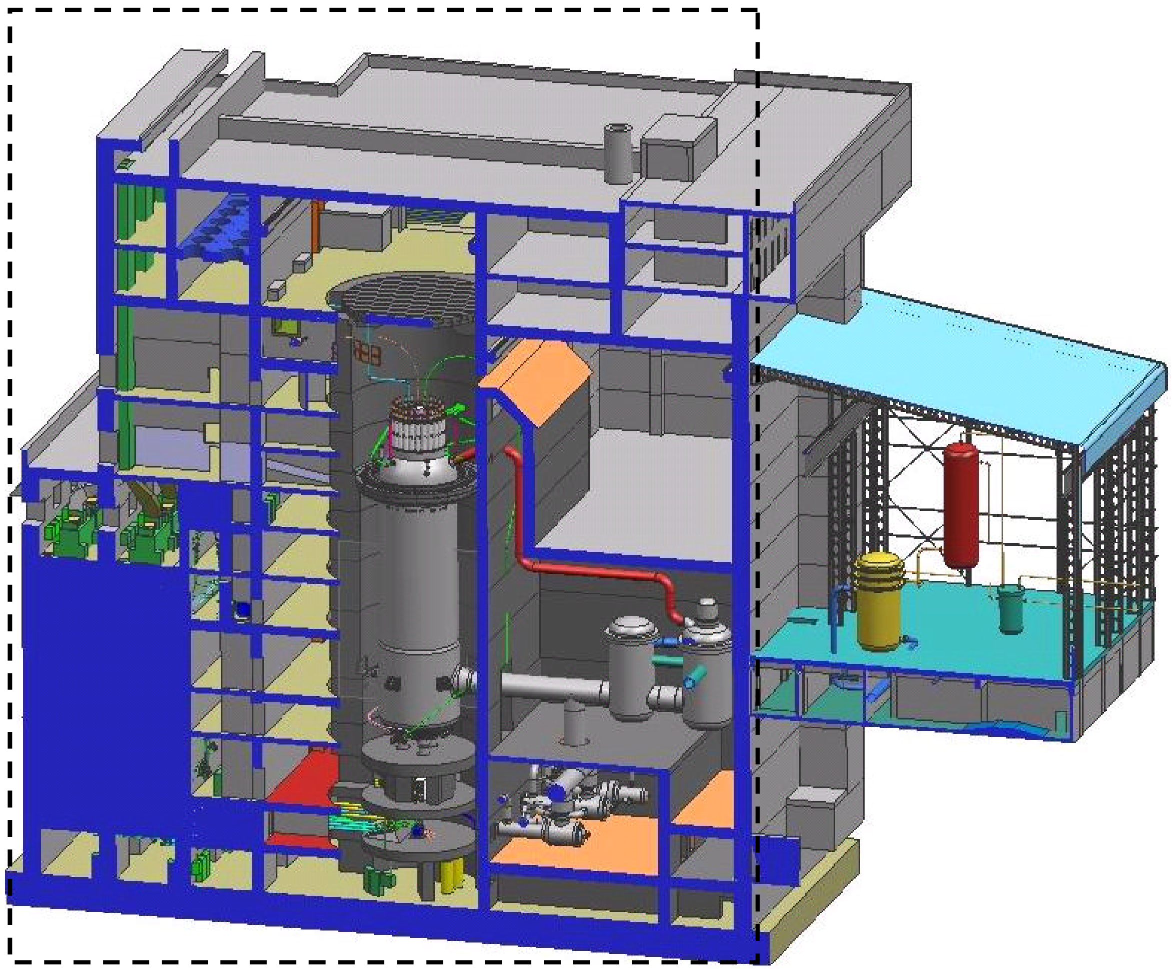
- Objectives:
  - To develop Technical and Functional Requirements (T&FRs) for the NGNP reactor building,
  - To identify and evaluate alternative reactor building design strategies
- Scope:
  - Definition of overall technical requirements for the reactor building
  - Evaluation of design options to meeting these requirements
  - Allocation of radiological retention requirements for reactor building

RB Requirements Approach



Scope of Reactor Building

Reactor Building



P B M R

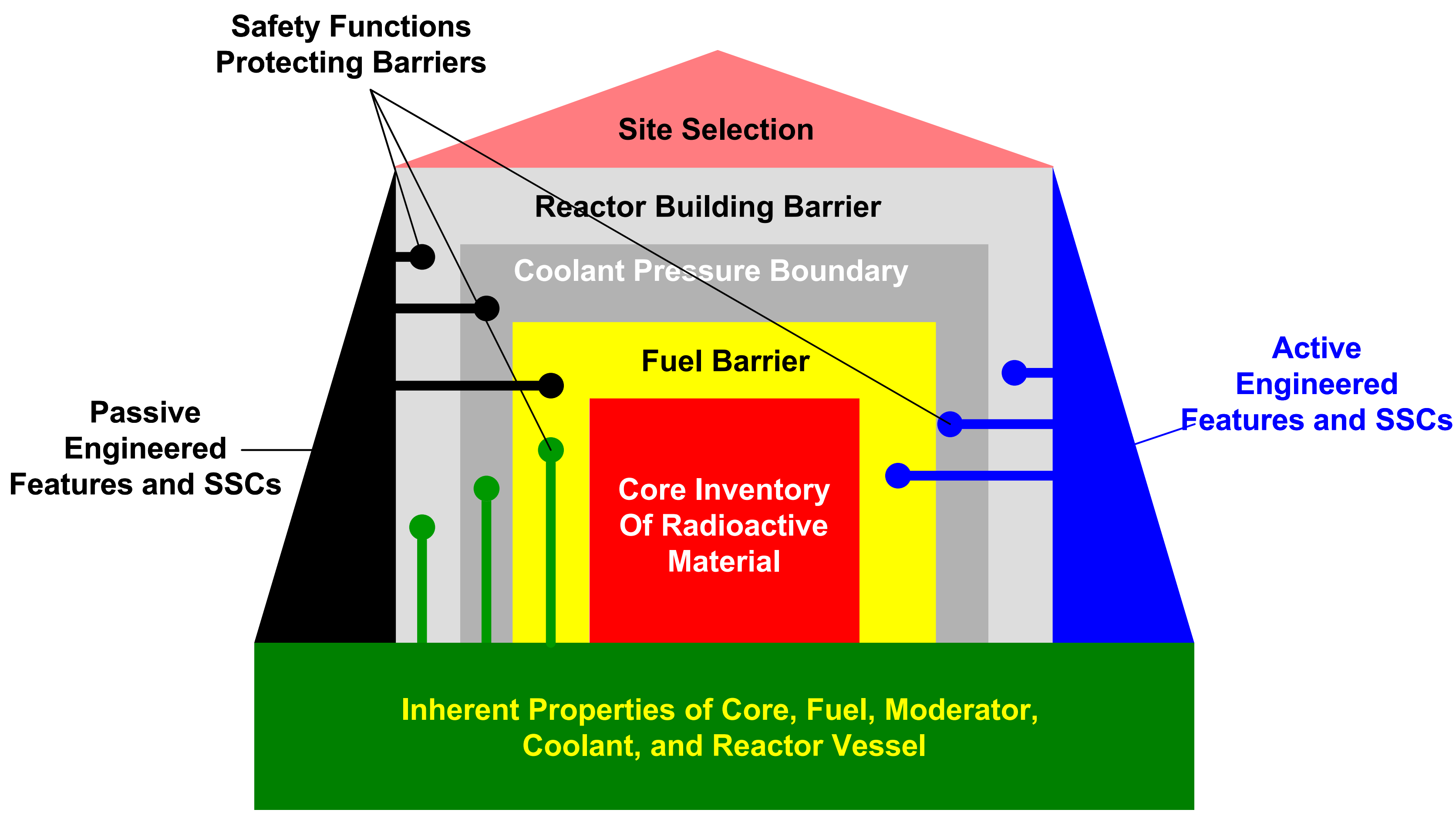
Major Contents of Reactor Building

- Reactor Unit System,
- Core Conditioning System,
- Reactor Cavity Cooling System,
- Fuel Handling and Storage System,
- Helium Service System,
- Heating Ventilation and Air Conditioning (HVAC)
- Pressure Relief System
- Other NHSS support systems.
- Portion of the Heat Transport System inside the building
  - Primary Heat Transport System
  - IHX vessels, circulators and
  - Part of Secondary Heat Transport System

NGNP PBMR Safety Design Approach

- PBMR safety philosophy starts with the application of defense-in-depth principles in making fundamental design selections:
  - Inherent reactor characteristics
  - Multiple, robust, and concentric barriers to radionuclide release.
  - Conservative design approaches to support stable plant operation with large margins to safety limits
  - In the event safety functions are challenged, the safety philosophy is to utilize the inherent reactor characteristics and passive design features to fulfill the required safety functions.
  - Additional active engineered systems and operator actions are provided to reduce the challenge to plant safety and provide defense-in-depth in preventing and mitigating accidents.
  - Avoid need for early operator intervention, or early functioning of any active systems to maintain safe stable state.

PBMR NGNP Approach to Plant Capability Defense-in-Depth



Reactor Building Functions

- Safety Functions
  - **Required Safety Functions** are those functions that are necessary and sufficient to meet the dose limits for Design Basis Events and deterministically selected Design Basis Accidents
  - **Supportive Safety Functions** are all other functions that contribute to the prevention or mitigation of accidents and support the plant capabilities for defense-in-depth
- Physical Security Functions
  - Protect the reactor and vital equipment from design basis threats associated with acts of sabotage and terrorism
- Other Functions
  - Functions necessary for plant construction, operation, maintenance, access, inspection, worker protection, and control of routine releases of radioactive material during normal operation

RB Functional Requirements

- The scope of functions and requirements for the reactor building are allocated in the PCDR to :
  - Nuclear Heat Supply Building (NHSB)
  - NHSS HVAC System
  - NHSB Pressure Relief System (PRS)
- The RB shall perform the following required functions:
  - House and provide civil structural support for the reactor and all the SSCs within the scope of the NHSS (building geometry/space)
  - Resist all structural loads as required for required safety functions allocated to the NHSS
  - Protect the SSCs within the NHSS that perform required safety functions from all internal and external hazards as identified in the LBEs
  - Provide physical security of vital areas of the NHSS against acts of sabotage and terrorism
- The RB shall perform supportive safety functions involving the retention of radionulcides released from the PHTS HPB and the limiting of air ingress following HPB leaks and breaks

PRS Functional Requirements

- The Pressure Relief System shall perform the following functions:
  - The PRS must be compatible with the NHSB boundary functional requirement.
  - The PRS shall be designed, and the NHSB compartments shall be sized so that leaks and breaks up to 10mm equivalent break size anywhere along the PHTS HPB does not open the PRS so that HVAC filtration shall be maintained
  - Generally, PRS must open to prevent overpressure and damage to safety related SSCs, and reclose to enable post-blow-down filtration and leakage control for the following design basis event conditions.
    - Breaks in the PHTS HPB in each NHSB compartment containing NHSS piping or vessels up to 100mm equivalent break size
    - Breaks in the SHTS HPB in each NHSB compartment containing NHSS piping or vessels up to 1meter equivalent break size

HVAC Functional Requirements

- The NHSS HVAC System shall perform the following functions:
  - The HVAC system shall
    - maintain internal environmental conditions for NHSS SSCs during normal operation
    - Maintain negative pressure in the RBVV areas to keep normal releases ALARA
    - survive conditions for all AOOs and DBEs, and
    - provide a post-event cleanup function for all AOOs and DBEs.
  - The HVAC system <u>may</u> also play a role in mitigating the release of radioactive material depending on the selected design strategy

Physical Security Requirements

- Insights from experience with AP1000 being used to establish physical security requirements for the RB
- Security requirements established in 10 CFR 50, 52, and 73
- Need to define "vital areas" and "vital equipment" in light of NGNP risk-informed performance-based licensing approach and SSC safety classification method
- Vital areas subjected to rigorous analysis to address classified Design Basis Threats which involve both design basis and beyond design basis accident conditions
- Robust physical protection provided in design to address DBT expected to contribute to margins in the capacity to withstand loads from internal and external event accidents
- Consideration of these requirements not expected to influence evaluation of reactor building confinement options but could influence evaluation of embedment

Licensing Basis Events
Framing Reactor Building Requirements

NGNP Licensing Basis Events

- LBEs to be defined via full scope PRA during conceptual design,
- For this study LBEs are defined by engineering judgment based on reviews of the plant design per PCDR and experience with HTGR and specifically PBMR PRAs
- Categories of LBEs include
  - Internal Events
    - HPB leaks and breaks in PHTS and SHTS
    - Transients with and without reactivity addition
  - Internal Plant Hazard Events
    - Internal fires and floods
    - Rotating machinery missiles
    - High energy line and tank breaks
  - External Plant Hazard Events
    - Hydrogen Process events
    - Seismic Events
    - Aircraft crashes and transportation accidents
    - High winds and wind generated missiles

Assumed LBEs Involving Leaks and Breaks

- AOOs assumed to include leaks up to 10mm in break size anywhere along the PHTS and SHTS HPB inside the RB– PRS and HVAC designed to keep HVAC running during such leaks
- DBEs are expected include:
  - Up to 50mm breaks in the FHSS pipes above or below the RPV
  - Up to 100mm breaks in the CIP, COP or other PHTS piping above or below the core including pipe to vessel nozzle welds
  - Breaks in HSS piping bounded by 100m break size
  - Up to 1meter (SEGB) breaks in the SHTS piping inside the NHSB
  - Requirement is to maintain structural integrity and avoid damage to any safety related SSC inside the NHSB
- **BDBEs are expected to include**
  - Leaks and small breaks in the RPV
  - Simultaneous small leaks above and below the core
  - Up to 1meter (SEGB) breaks in the PHTS
  - Requirement is to meet TLRC using realistic assumptions
  - Structural integrity of the RPV and major PHTS SSCs maintained

HPB leaks and Breaks Licensing Basis Events

LBE Type	Frequency Range (per plant year	Break Category	Equivalent Break Size (mm)
Anticipated Operational Occurrence	$= 10^{-2}$	Small	1 to 10
Design Basis Events	10^{-4} to 10^{-2}	Medium	> 10 to 100
Beyond Design Basis Events	$<10^{-4}$	Large	> 100 to 1,000*

*corresponds to a double ended guillotine break of a 700mm pipe

NHSB Requirements Summary

- Plant Performance Requirements
  - Plant reliability and capacity factor
  - Plant investment risk
- Safety and Licensing Requirements
  - Top Level Regulatory Criteria
  - Applicable existing NRC guidance
  - Policies and guidance specific to advanced reactors
  - Safety and licensing requirements keyed to specific categories of LBEs
    - Leaks and breaks in HPB
    - Transients
    - Internal fires and floods
    - Seismic events
    - Beyond design basis aircraft crashes and large fires and explosions
    - Hydrogen process hazards
- Physical Security Requirements
- Plant Costs
  - Capital costs
  - Operating and maintenance costs
  - Technology development costs
- NGNP Schedule
- Public Acceptance

EVALUATION OF ALTERNATIVE REACTOR BUILDING DESIGN STRATEGIES (Other than Embedment)

Approach to Evaluation of Alternative RB Configurations

- Identify required and supportive safety functions of RB
- Identify alternative design strategies to meet each safety function
- Define discrete set of alternative RB design configurations with appropriate combinations of design features
- Evaluate each configuration against technical and functional performance criteria
  - Normal Operating Functional Requirements
  - Investment Protection Functional Requirements
  - Safety Functional Requirements
  - Physical Security / Aircraft Crash Requirements
  - Capital and O&M Cost Impact
  - Licensability
- Identify basis for recommending one or more alternative configurations
- Result is a recommendation for the Conceptual Design to consider in meeting the functional and technical requirements and not a proposed design decision

RB Study Evaluation Criteria and Weights

Criteria		Relative Weight
• Normal Operation Requirements		10%
• Investment Protection Requirement		5%
• Safety Functional Requirements		35%
– HPB leaks/breaks	20%	
– Seismic	5%	
– Hydrogen/process hazards	10%	
• Security /aircraft crash		10%
• Capital and Operating Cost		25%
• Licensability		15%
		100%

Range of Options to Meet Structural Building Functions

- Alternative pressure relief approaches
  - Open vent pathways
  - Blow out panels
  - Building shape
- Protection against internal hazards
  - Zoning strategies
  - Internal barriers
- Protection against external hazards and physical security threats
  - Round vs. rectangular shape
  - Reactor embedment

Range of Options to Mitigate Releases and Air Ingress

- Radionuclide retention
  - HVAC pressure zones
  - HVAC filtration
    - Active
    - Passive
  - Controlled release pathway
  - Building leak rate control

- Air ingress control
  - Building leak rate control
  - Re-closable vent dampers
  - Inert gas injection not considered because capability to inject He or N_2 into reactor core is already inherent in reference PBMR design

Range of Optional Design Approaches for Radionuclide Retention and Air Ingress Mitigation

- **Unfiltered Vented**
- **Filtered Vented w/ Controlled leakage**
  - **Option with delayed release filter capability**
  - **Option with capability to filter immediate and delayed phases of release**
  - **Options with different vent area volumes**
- **Pressure Retaining**
  - Options with different vent area volumes

Alternative Approaches to Meeting Required RB Safety Functions

RB Required Safety Function	Alternative Design Approaches
Provide structural integrity in response to HPB break pressure and temperature loads	• Robust structural design • Large open vent pathways • Rupture panels • Added expansion volume
Provide structural integrity in response to seismic and external event loads and malevolent threats	• Robust structural design • Embedment
Prevent damage to fuel storage and safety related SSCs from HPB breaks	• Robust Structural design • Physical separation of SSCs

Alternative Approaches to Meeting Supportive RB Safety Functions

RB Supportive Safety Function	Alternative Design Approaches
Prevent damage to HVAC during blow-down from HPB breaks	• Isolation dampers • Blast panels • Physical separation of HVAC SSCs
Maintain environmental conditions and control radionuclide releases during normal plant operation	• HVAC with active filtration • pressure and radiation zones
Provide radionuclide retention during blow-down phase	• Passive filtration • Limit RB leak rate
Provide radionuclide retention during delayed fuel release phase	• Passive filtration • Re-establish HVAC filtration • Control release pathway • Limit RB leak rate
Limit potential for air ingress to PHTS following large or multiple HPB breaks	• Limit RB leak rate • Provisions for inerting of PHTS • Provisions for inerting of RB

Design Features Common to all Selected Alternative Configurations

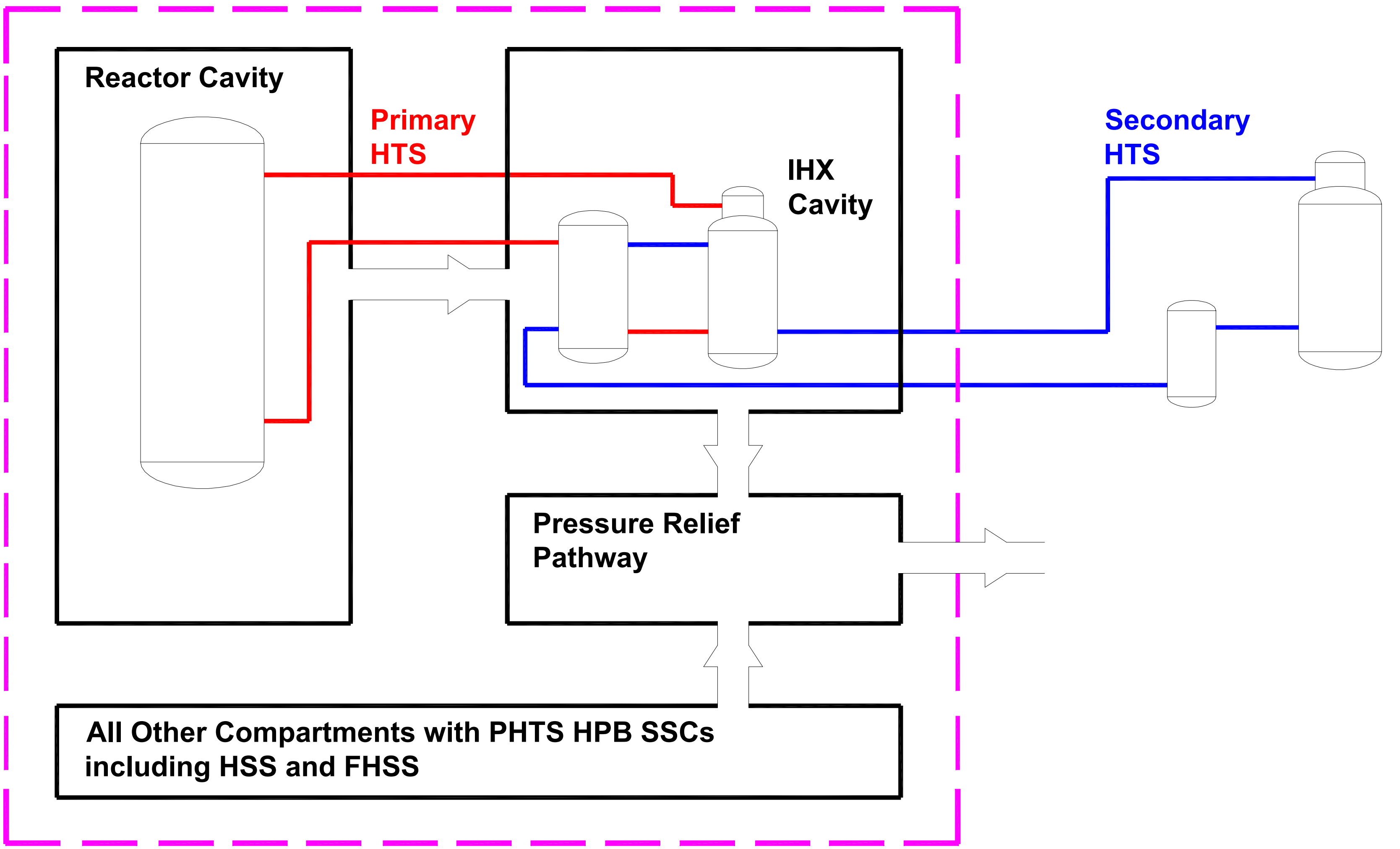
- Robust structural design to protect against loads from internal and external hazards including seismic events, hydrogen process hazards and physical security against malevolent acts as described in selected licensing basis events
- HVAC filtration system provided to maintain environmental conditions and provide radionuclide retention ALARA during normal operation and for anticipated operational occurrences (AOOs)
- Blast panels or isolation devices provided to isolate/ protect normal operation HVAC from HPB breaks
- Physical isolation/separation provided for fuel storage and safety related SSCs from blow-down vent path following HPB breaks
- Each alternative configuration considered can be combined with all evaluated embedment alternatives
- Design of the reactor and HPB pressure boundary and passive safety characteristics will limit the potential for air ingress; capability to support ad hoc emergency actions to inert the reactor core cavity via injection of helium or nitrogen is inherent in the PBMR fuel handling system design
- All designs considered limit the supply of outside air by controlling building leak rate but leak rate varies among the considered alternatives

Fission Product Transport and Pressure Relief Pathways during HPB Leak/Break Events

- HBP leaks/breaks in Reactor Cavity
  - Expands to reactor cavity via engineered flow path to IHX cavity
  - Flows through engineered flow path from IHX cavity to PRS vent shaft
- HPB leak/break in CCS Cavity (not shown in following schematics)
  - Expands to reactor cavity via engineered flow path to IHX cavity
  - Flows through engineered flow path from IHX cavity to PRS vent shaft
- HPB leak/break in FHSS or HSS areas
  - Expands to FHSS or HSS spaces
  - Flows through engineered flow path to PRS vent shaft
- HPB leak/break in IHX space on either the PHTS or the SHTS side
  - Expands to IHX space
  - Flows through engineered flow path to PRS vent shaft

Some expansion into other building spaces may also occur through imperfect doors, seals, and penetrations. Leakage from RB envelope to environment will depend on envelope leak rate.

Alternative 1a
Unfiltered and Vented



NHSB Alternative 1a
Unfiltered and Vented

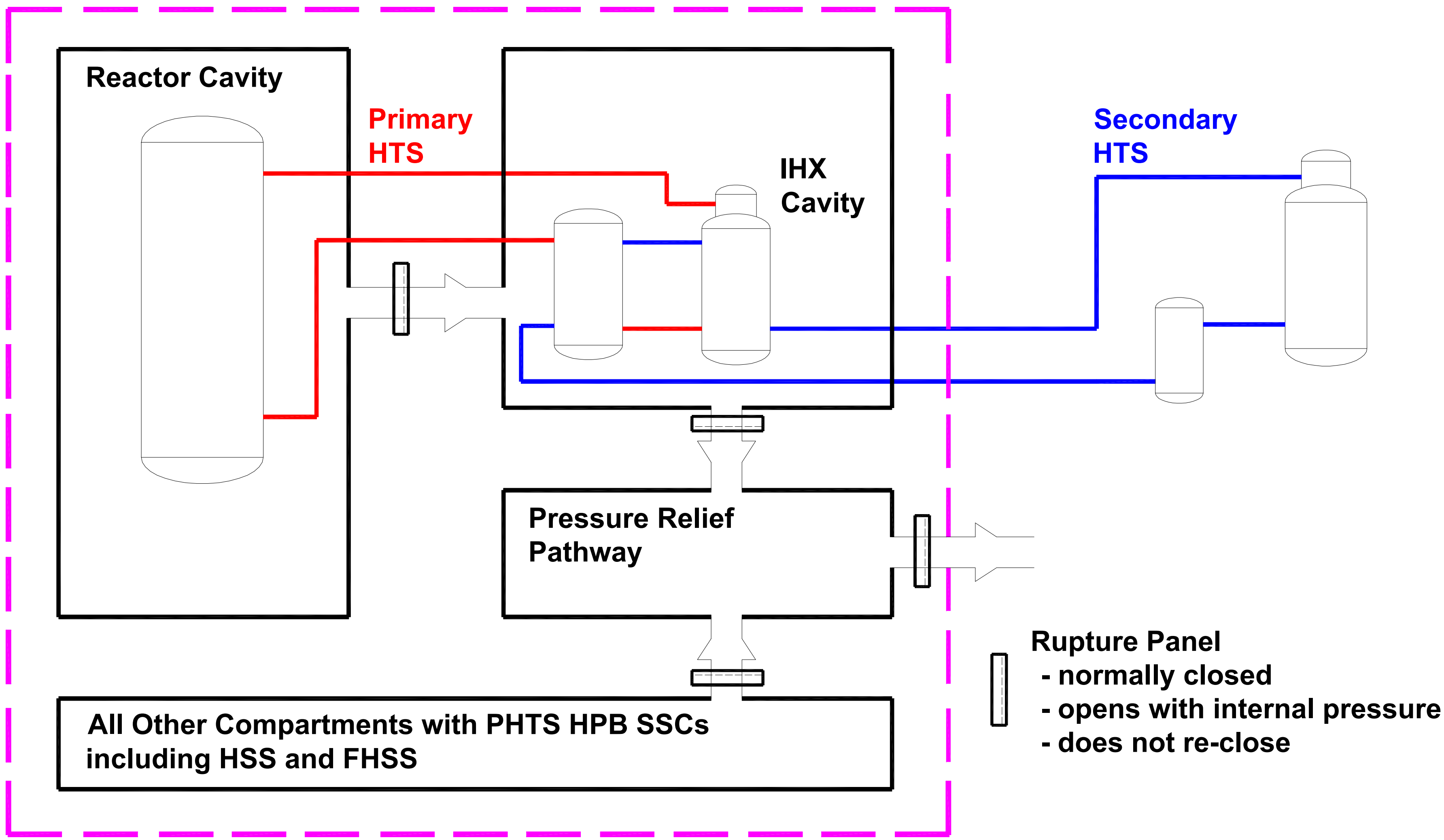
<u>Key Design Features:</u>

- Pressurized Zone Leak Rate - 50-100 vol % /day (post blow-down)
  - Doors and penetrations- Industrial grade
  - Sealing and finish – industrial grade
- Pressure Relief Pathway
  - Isolated from normal HVAC by low pressure rupture panels
  - Clean surfaces to minimize deposition during blow-down
  - Smooth transitions from room to room to minimize flow resistance
  - Internal open vents to limit compartment over-pressure and control release pathway
  - External open vents to a relief shaft
  - No capability for vent shaft re-closure
- Filtration of Accidental releases
  - No filtration of blow-down phase of release. Some passive decontamination due to plate-out, condensation, and decay.
  - Passive plate-out, condensation, settling, radioactive decay for delayed fuel release
- Normal HVAC isolated from pressure transient by isolation device but has no safety function
- Post-event cleanup using normal HVAC if available

Pros and Cons of Alternative 1a

Feature	Advantage	Disadvantage
Open vent paths	• Minimizes pressure transient • Inherently reliable as it is completely passive	• More difficult to control release of air activation products during normal operation • No engineered features to retain accident related fission products or limit air ingress to building following blow-down • May require very large HVAC equipment to maintain zoning and zone pressure gradients

Alternative 1b
Unfiltered and Vented with Blowout Panels



NHSB Alternative 1b
Unfiltered and Vented with Blowout Panels

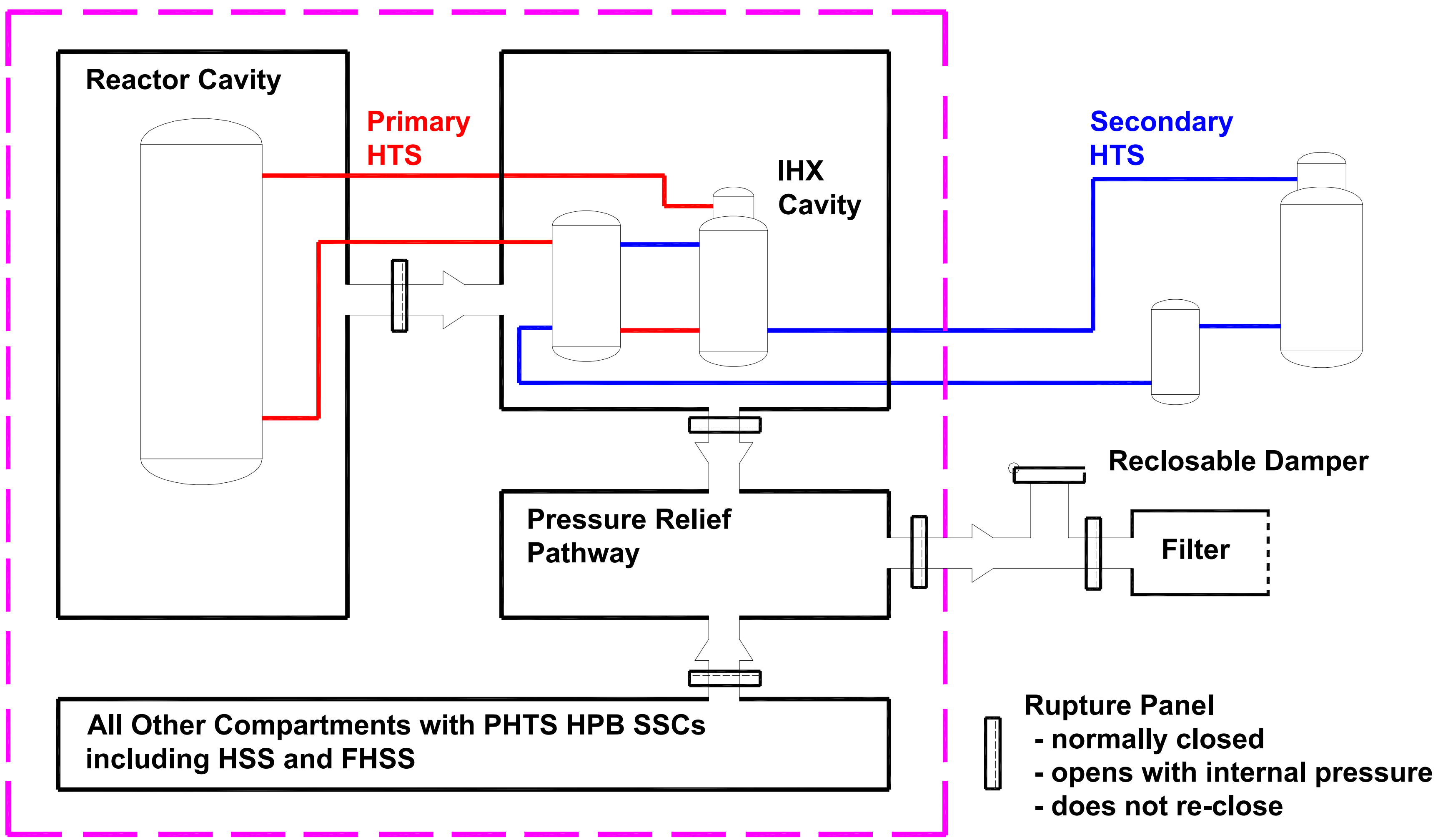
Key Design Features:

- Building Leak Rate - 50-100 vol % /day (post blow-down)
  - Doors and penetrations- Industrial grade
  - Sealing and finish – industrial grade
- Pressure Relief Pathway
  - Isolated from normal HVAC by low pressure rupture panels
  - Clean surfaces to minimize deposition during blow-down
  - Smooth transitions from room to room to minimize flow resistance
  - Internal blowout panels to limit compartment over-pressure and control release pathway
  - External blowout panels to control release path via a relief shaft
  - No capability for vent shaft re-closure
- Filtration of Accidental releases
  - No filtration of blow-down phase of release. Some passive decontamination due to plateout, condensation, and decay.
  - Passive plate-out, condensation, settling, radioactive decay for delayed fuel release
- Normal HVAC isolated from pressure transient by isolation device but has no safety function
- Post-event cleanup using normal HVAC if available

Pros and Cons of Alternative 1b

Feature	Advantage	Disadvantage
Rupture panel on vent paths	• Easier to control release of air activation products during normal operation • Easier for HVAC to control pressure zones	• Increases pressure transient and design pressure • Reliable but less so than open vent path
Vent paths remain open after rupture		• No engineered features to retain accident related fission products

Alternative 2
Partially Filtered and Vented with Blowout panels



NHSB Alternative 2
Delayed Release Filtered and Vented with Blowout Panels

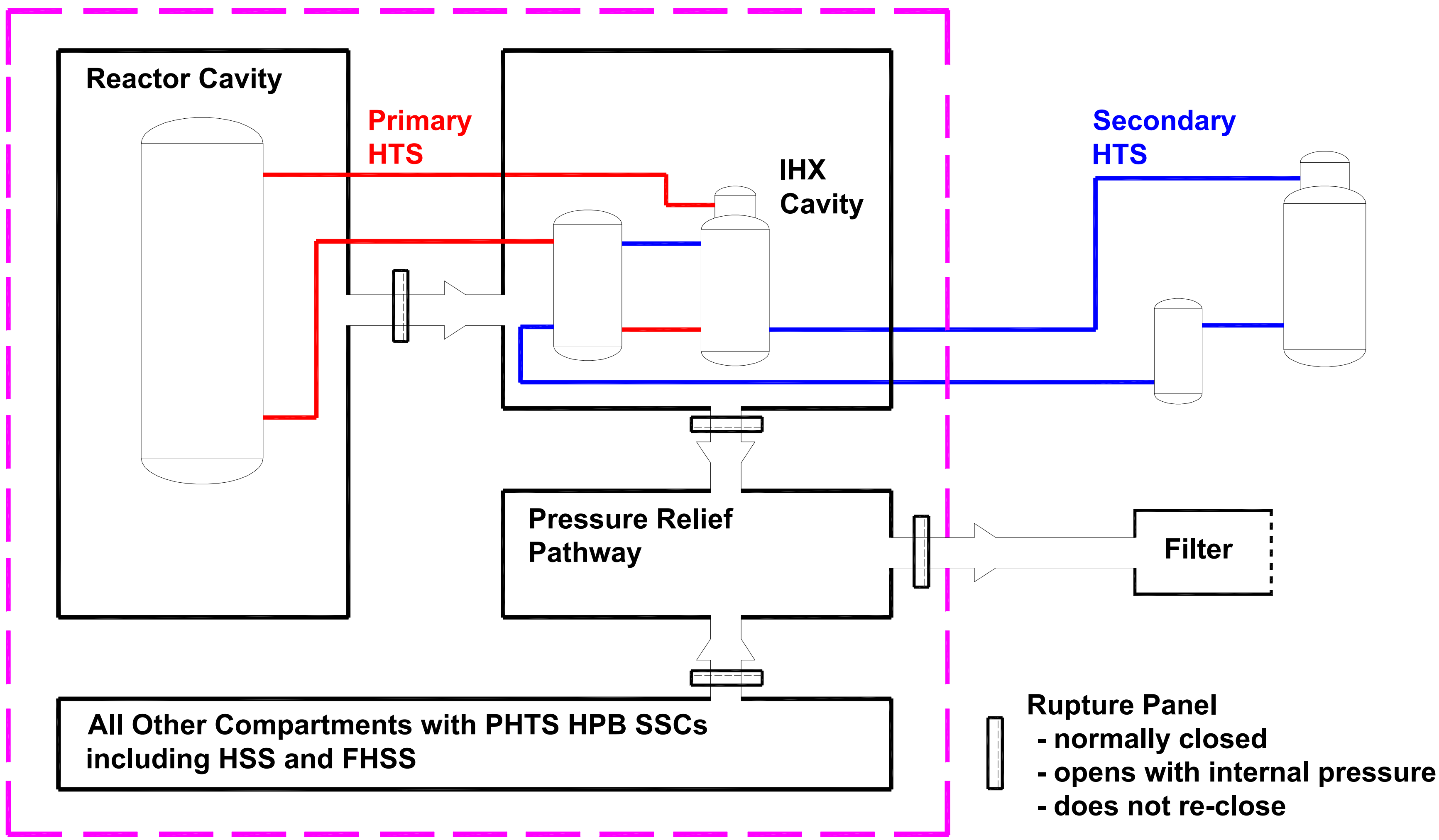
Key Design Features:

- Building Leak Rate - 25-50 vol % /day (Assumed during delayed release)
  - Doors and penetrations– soft and inflatable seals
  - Sealing and finish – Building boundary sealed at element joints by gaskets, caulking, or welding
- Pressure Relief Pathway
  - Isolated from normal HVAC by low pressure rupture panels
  - Clean surfaces to minimize deposition during blow-down
  - Smooth transitions from room to room to minimize flow resistance
  - Internal blowout panels to limit compartment overpressure and control release pathway
  - External blowout panels to control release path via relief shaft
  - Exit point through pressure actuated damper during initial blow-down and through Post-Event HVAC filtration following pressure equalization and damper re-closure
  - HVAC filter options include sintered metal, HEPA, wet scrubbers, and others
- Normal HVAC isolated from pressure transient by blast panels - HVAC restart after pressure equalization to assist post blow-down filtration

Pros and Cons of Alternative 2

Feature	Advantage	Disadvantage
Rupture panel on vent paths	• Easier to control release of air activation products during normal operation • Easier for HVAC to control pressure zones	• Increases pressure transient
Re-closeable damper on vent exit	• Minimizes pressure transient by opening vent during large HPB break • Increases effectiveness of filter	• Increases cost
Filter on vent exit	• Reduces the delayed fuel release source term	• Increases cost

Reactor Building Alternative 3a
Filtered and Vented with Blowout Panels



NHSB Alternative 3a
Filtered and Vented with Blowout Panels

Key Design Features:

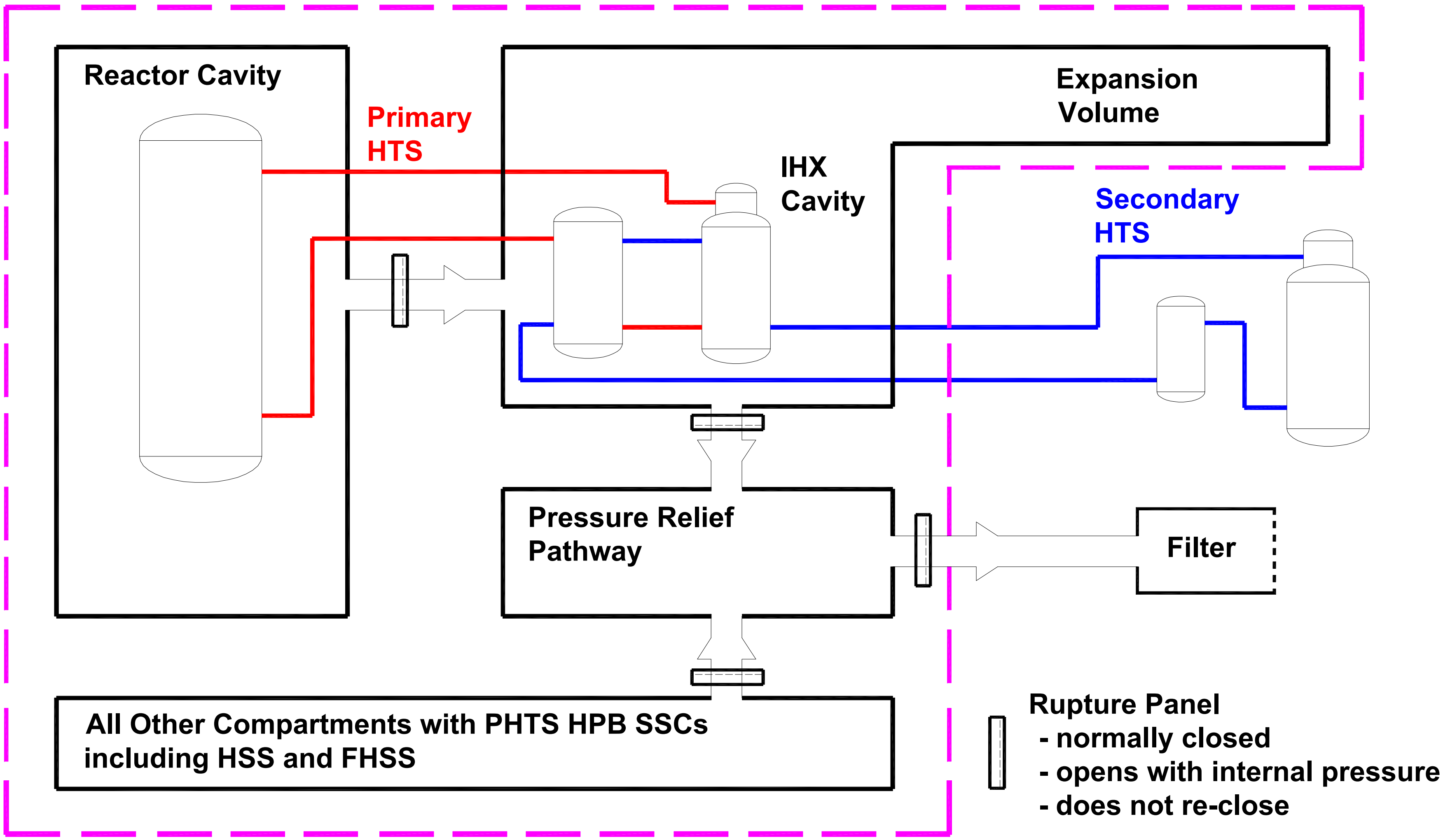
Building Leak Rate - 25-50 vol % /day (during delayed release)
- Doors and penetrations– soft and inflatable seals
- Sealing and finish – Building boundary sealed at element joints by gaskets, caulking, or welding

- Pressure Relief Pathway
  - Isolated from normal HVAC by low pressure rupture panels
  - Rough surfaces to maximize deposition during blow-down
  - Sharp transitions from room to room to maximize flow resistance
  - Internal blowout panels to limit compartment overpressure and control release pathway
  - External blowout panels to control release path via relief shaft
  - Exit point via passive filters during blow-down – passive filter must tolerate pressure and temperature
  - Exit point through filter following pressure equalization
  - Post blow-down filter options include sintered metal, HEPA, wet scrubbers, and others
- Normal HVAC isolated from pressure transient by blast panels

Pros and Cons of Alternative 3a

Feature	Advantage	Disadvantage
Rupture panel on vent paths	• Easier to control release of air activation products during normal operation • Easier for HVAC to control pressure zones	• Increases pressure transient
Filter on vent exit	• Reduces the prompt release source term • Reduces the delayed fuel release source term	• Increases pressure transient
Reduced leak rate for pressurized zone	•Reduces release of radionuclides through building boundary (filter bypass)	• Increases cost

Alternative 3b
Filtered and Vented with Blowout Panels and Expansion Volume



Alternative 3b
Filtered and Vented with Blowout Panels and Expansion Volume

Key Design Features:

Building Leak Rate - 25-50 vol % /day (during delayed fuel release)

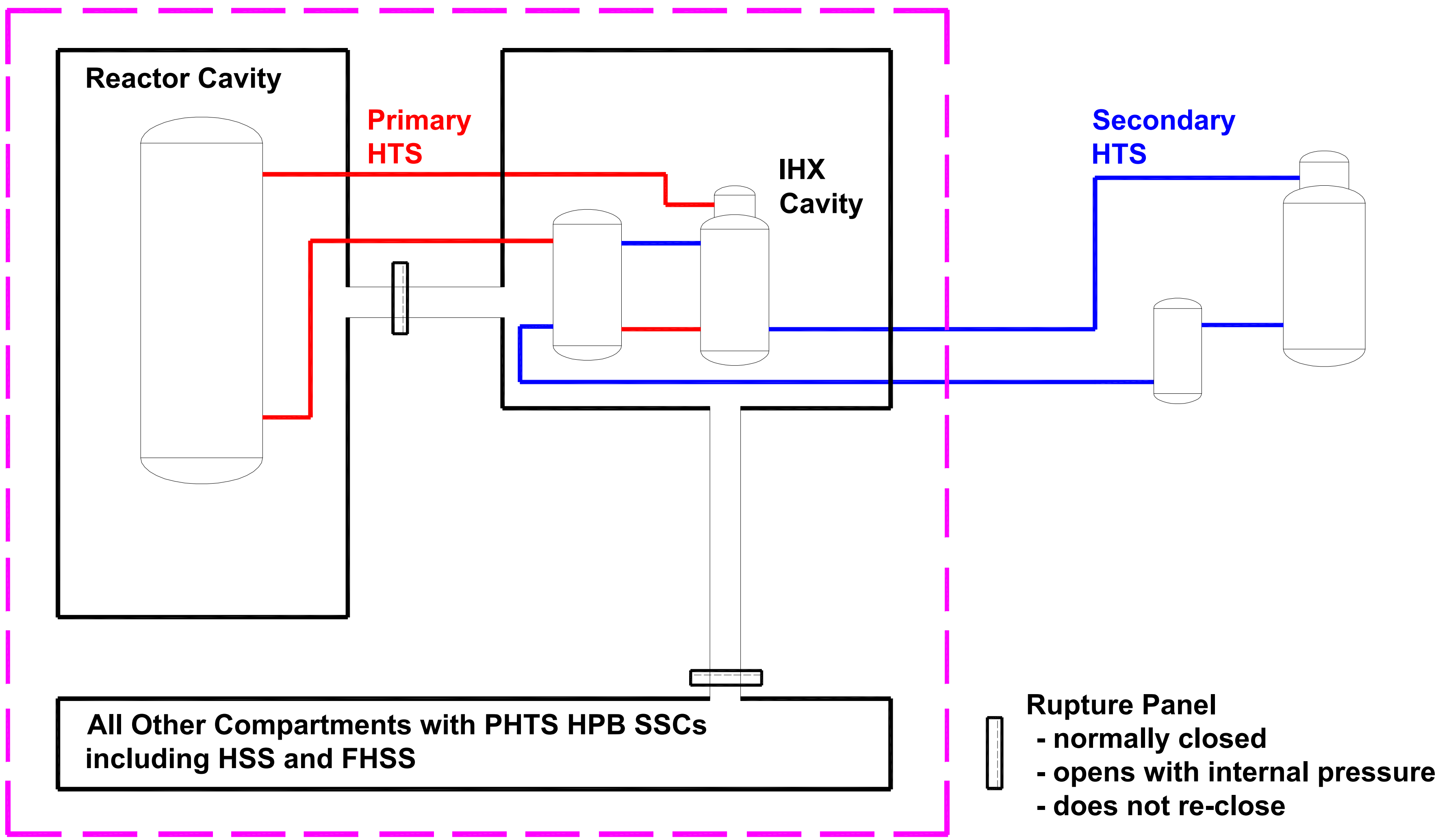
  - Doors and penetrations- Nuclear Grade – soft and inflatable seals
  - Sealing and finish – Building boundary sealed at element joints by gaskets, caulking, or welding

- Pressure Relief Pathway
  - Isolated from normal HVAC by low pressure rupture panels
  - Rough surfaces to maximize deposition during blowdown
  - Sharp transitions from room to room to maximize flow resistance
  - Internal blowout panels and expansion volume to limit compartment overpressure and control release pathway
  - External blowout panels to control release path via relief shaft
  - Exit point via passive filters during blow-down – passive filter must tolerate pressure and temperature (sintered metal?)
  - Exit point through filter following pressure equalization
  - Post blow-down filter options include sintered metal, HEPA, wet scrubbers, and others
- Normal HVAC isolated from pressure transient by blast panels

Pros and Cons of Alternative 3b

Feature	Advantage	Disadvantage
Rupture panel on vent paths	•Controls release of air activation products	•Increases pressure transient
Filter on vent exit	•Retains some radionuclides from prompt release •Retains some radionuclides from delayed release	•Increases pressure transient
Increased expansion volume	•Reduces pressure transient	•Increases cost
Reduced leak rate for pressurized zone	•Reduces release of radionuclides through building boundary (filter bypass)	•Increases cost

Alternative 4a
Pressure Retaining with Internal Blowout Panels



Alternative 4a
Pressure Retaining with Internal Blowout Panels

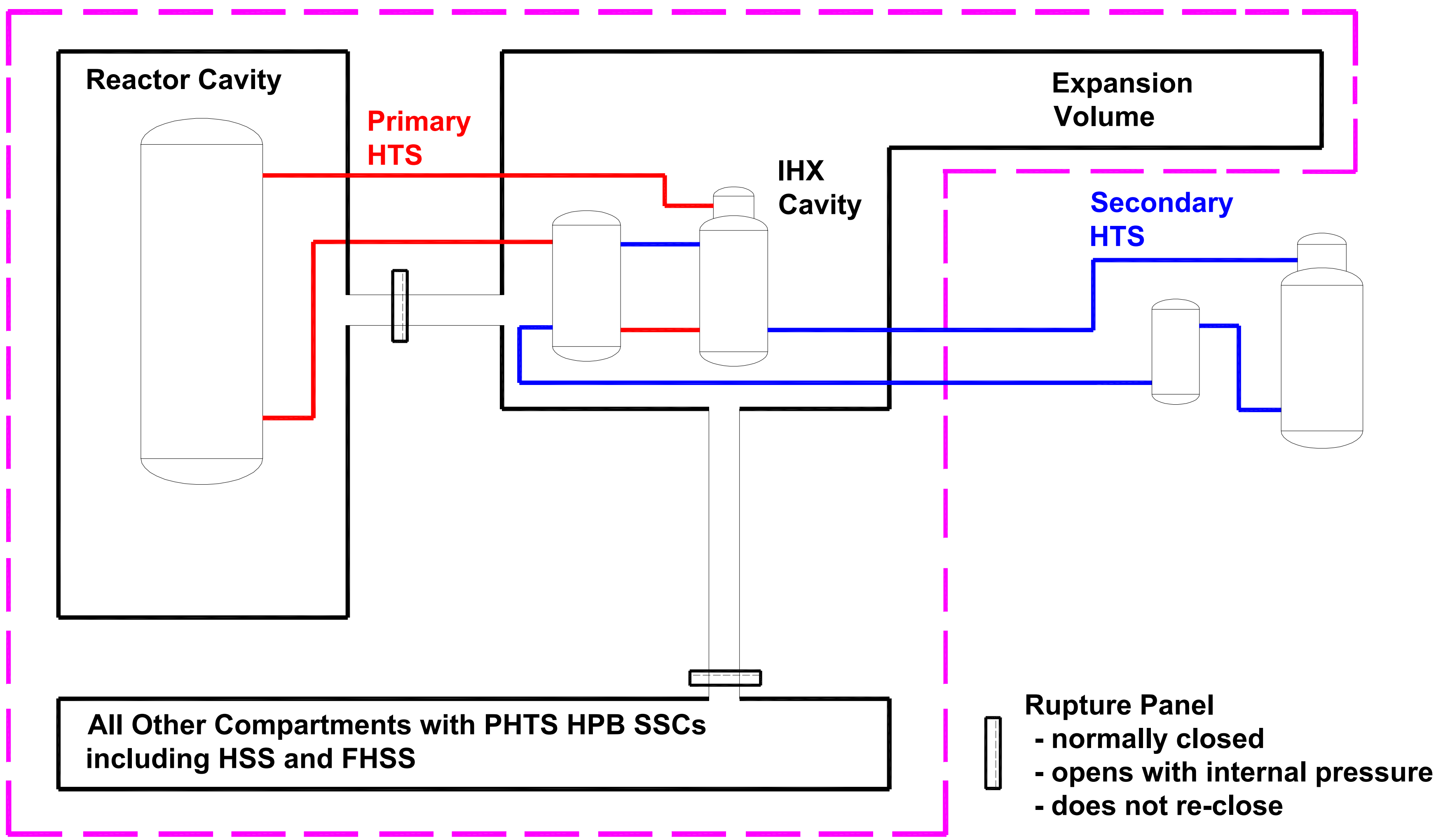
Key Design Features:

- Building Leak Rate - 0.1-1 vol % /day
  - Doors and penetrations- Airlock doors with inflatable seals, hard penetrations (welded)
  - Sealing and finish – Epoxy or metal building liner, welded
  - Building geometry may change to enable resistance to pressure loads
  - Achieving 1%/d leak rate may complicate design for RCCS
- Pressure Relief Pathway
  - Internal blowout panels to limit compartment overpressure and control release pathway
  - No external release point other than normal leakage and post event purge lines
- Normal HVAC isolated from pressure transient by blast panels
- Post-event cleanup using dedicated extraction fans and filters and perhaps heat exchangers (to protect filters)
  - Post event building depressurization will include long lived noble gases

Pros and Cons of Alternative 4a

Feature	Advantage	Disadvantage
Elimination of engineered vent path	• All PHTS HPB releases retained within building	• Increases pressure transient • Increases cost • May require curved walls and slabs to resist pressure transient
Reduced leak rate for pressurized zone	• Reduces release of radionuclides through building boundary	• Requires engineered penetrations • May require design changes in RCCS, HVAC • Increased requirements for leak rate testing • Containment failures modes may reduce effectiveness

Alternative 4b
Pressure Retaining with Internal Blowout Panels and Expansion Volume



Alternative 4b
Pressure Retaining with Internal Blowout Panels and Expansion Volume

Key Design Features:

- Building Leak Rate - 0.1-1 vol % /day
  - Doors and penetrations- Airlock doors with inflatable seals, hard penetrations (welded)
  - Sealing and finish – Epoxy or metal building liner, welded
  - Building geometry may change to enable resistance to pressure loads
  - Achieving 1%/d leak rate may complicate design for RCCS
- Pressure Relief Pathway
  - Internal blowout panels and expansion volume to limit compartment overpressure and control release pathway
  - No external release point other than normal leakage and post event purge lines
- Normal HVAC isolated from pressure transient by blast panels
- Post-event cleanup using dedicated extraction fans and filters and perhaps heat exchangers (to protect filters)
  - Post event building depressurization will include long lived noble gases

Pros and Cons of Alternative 4b

Feature	Advantage	Disadvantage
Elimination of engineered vent path	•All PHTS HPB releases retained within building	•Increases pressure transient •Increases cost •May require curved walls and slabs to resist pressure transient
Reduced leak rate for pressurized zone	•Reduces release of radionuclides through building boundary	•Requires engineered penetrations •May require design changes in RCCS, HVAC • Increased requirements for leak rate testing • Containment failures modes may reduce effectiveness
Increased Expansion Volume	•Decreases pressure transient	•Increases cost

Summary of Alternative RB Configurations

No.	Design Description	Pressurized Zone Leak Rate Vol % /day	NHSB Boundary Leak Rate Vol%/day	Pressure Relief Design Features	Post blow-down re-closure of PRS shaft?	Radionuclide Filtration	
						Blow-down phase	Delayed fuel release phase
1a.	Unfiltered and vented	50-100	50-100	Open vent	No	None	Passive
1b	Unfiltered and vented with blowout panels	50-100	50-100	Internal + External blowout panels	No	None	Passive
2	Partially filtered and vented with blowout panels	25-50	50-100	Internal + External blowout panels	Yes	None	Active HVAC
3a	Filtered and vented with blowout panels	25-50	50-100	Internal + External blowout panels	Yes	Passive	Active HVAC
3b	Filtered and vented with blowout panels and expansion volume	25-50	50-100	Internal + External blowout panels + expansion volume	Yes	Passive	Active HVAC
4a	Pressure retaining with internal blowout panels	0.1-1	50-100	Internal blowout panels	N/A	Passive	Passive
4b	Pressure retaining with internal blowout panels and expansion volume	0.1-1	50-100	Internal blowout panels + expansion volume	N/A	Passive	Passive

Some Cause and Effect Relationships

- Decrease in vent flow volume causes:
  - Increase in peak transient pressure
- Decrease in building leak rate causes:
  - Increase in peak transient pressure
  - Increased sophistication in penetrations
  - Increased difficulty in designing systems that operate across building boundary (RCCS, SHTS, HVAC)
- Increase in peak transient pressure causes:
  - Increased loads on walls and slabs
- Increase in wall and slab loads causes:
  - Increase in wall and slab thickness, or
  - Change to curved walls and slabs (round building)

Summary of Reactor Building Functional Requirements for Radionuclide Retention and Pressure Capacity

Objectives

- Establish requirements for the NGNP Reactor Building with respect to:
  - Retention of radionuclides during selected licensing basis events (LBEs) involving HPB leaks and breaks
  - Pressure capacity of the reactor building for LBEs
  - Dimensions of the Reactor Building Vent Volume (RBVV)
  - Leak Rate of the RBVV
  - Pressure set-points of rupture panels and HVAC dampers
- Evaluate identified Reactor Building concepts with respect to above requirements
- Establish radiological Safety Margins of identified Reactor Building concepts against Top Level Regulatory Criteria
- Evaluate phenomena responsible for trends in releases over a wide set of analysis cases

Boundary Conditions

- Top Level Regulatory Criteria
  - Site Boundary Dose Assessment
    - Total Effective Dose Equivalent (TEDE) = Effective Dose Equivalent (for external exposures) plus Committed Effective Dose Equivalent (for internal exposures).
    - EPA Protective Action Guides (PAGs)
      - PAG for TEDE: 1 rem/event
      - PAG for Thyroid Dose: 5 rem per event
    - 10 CFR 50.34 Criteria for Design Basis Accidents
      - TEDE: 25 rem/event
      - Thyroid: 75 rem/event

Constraints

- No integrated analysis model available for Blowdown Phase, Heatup Phase and Cooldown Phase coupled with a Reactor Building and Offsite Dose Model
- Project schedule and budget constraints
- Selected LBEs limited to Depressurized Loss of Forced Cooling (DLOFC) cases for a range of break sizes in the Core Inlet Pipe
  – Equivalent break sizes of 2,3,10,100, and 1000mm(DEGB of the 710mm CIP)
- For establishing RB design pressure a design basis event involving 1000mm DEGB of the SHTS HPB is assumed
  – Lack of active cooling for SHTS based on PCDR design introduces large uncertainties in reliability of hot gas pipe liner and insulation whose failure would directly lead to major rupture of the HPB

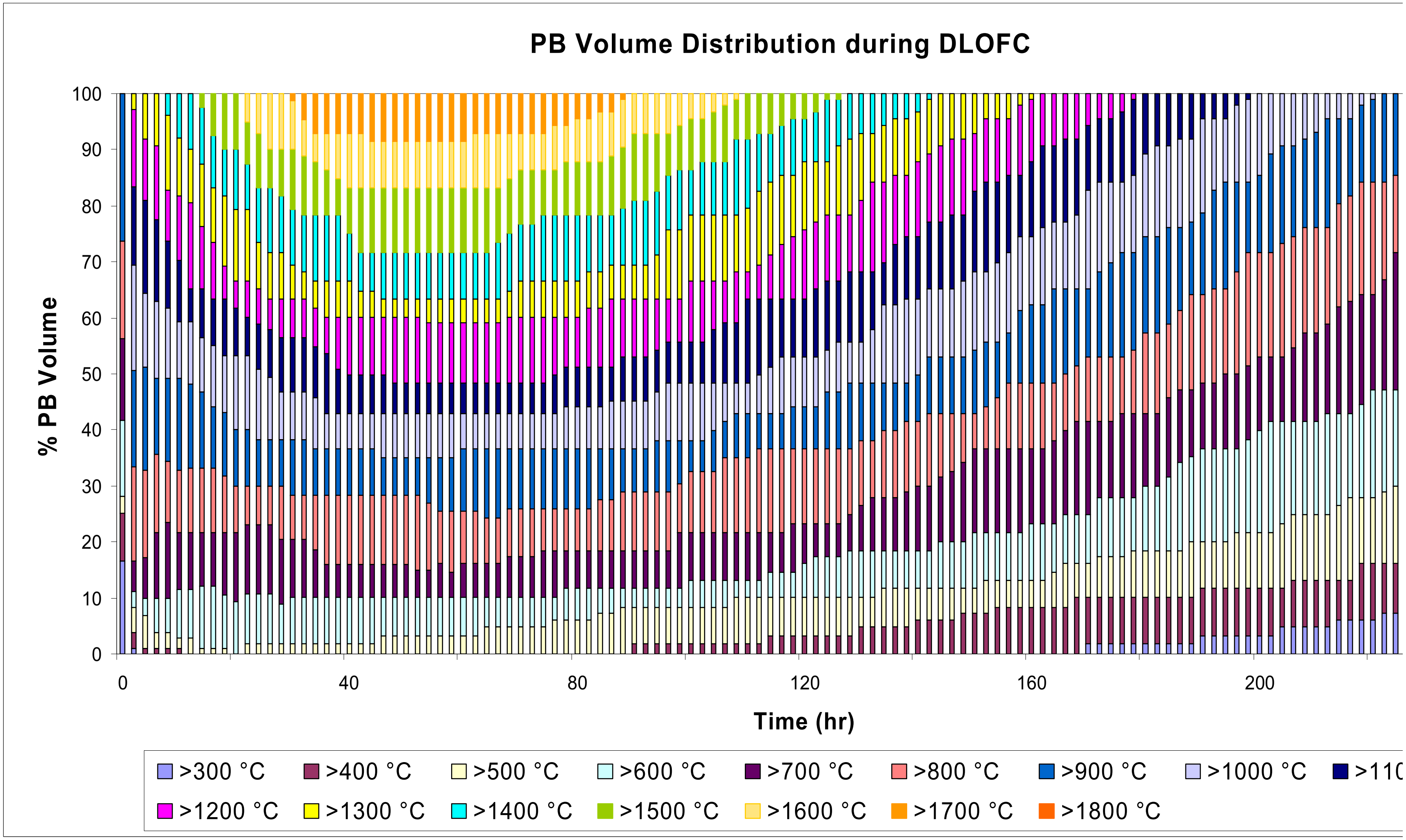
Radionuclide Information for DLOFC Cases

- Radionuclide Inventory in HPB at Beginning of Incident
  - Circulating Activity at End of Life Full Power Operation
  - Lifetime Plateout and Deposition Activity
  - Equilibrium Dust Activity
  - Source: NGNP Contamination Control Report (050108)

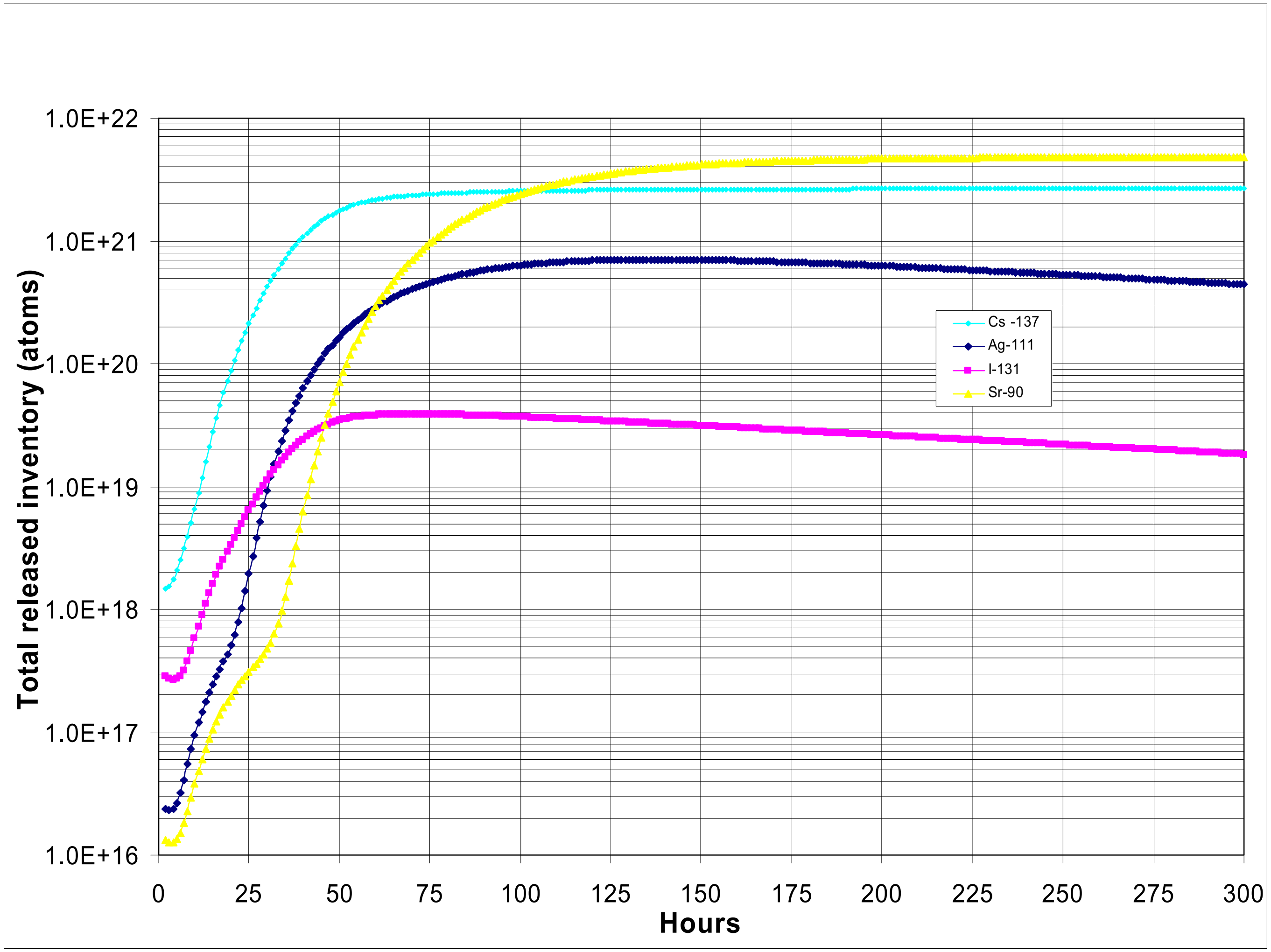
Radionuclide Information for DLOFC Case (continued)

- Transient Radionuclide Release from Fuel During Incident
  - Core temperature transient during DLOFC
  - Calculations assume atmospheric pressure at t = 0
  - Time dependent activity release during DLOFC core heatup as a function of core temperature for duration of releases from the core
  - Source: Existing PBMR calculations for a 500MWt design with different operating conditions

Core Fuel Time at Temperature Distribution during DLOFC



Activity Release from Fuel during DLOFC



Technical Approach

- Use technical information available to develop a simplified 3 volume integrated model of PHTS, RB and Site Boundary Dose
  - Volume 1: PHTS Hot Volume
  - Volume 2: PHTS Cold Volume with Break
  - Volume 3: Generalized RB Vent Volume (RBVV)
- Constant Volume Size with sensitivity analysis of alternative RBVV dimensions
- During heatup the expanding hot volume transfers hot coolant mass & energy to cold volume, mixes instantly in cold volume.

Technical Approach (continued)

- RBVV model capable of simulating all proposed RB design alternatives.
- Total volume of RB including vented and non-vented areas is approximately 100,000m^3 per PCDR
- Volume of RBVV of 10 to 20% of RB Volume is typical, up to 100% with expansion volume.
- For analyzed cases the RBVV is approximately equal to volume of IHX compartment and PRS relief shaft in NGNP PCDR design
- Radionuclide releases from RBVV directly to environment.
- Deposition and holdup in remaining RB volume (typically 80 to 90%) is neglected
- Model blowdown, heatup and cooldown phases for cold break DLOFC

Technical Approach (continued)

- Insights from previous HTGR studies indicate I-131 and Cs-137 are key to determining off-site doses
- Model calculates I-131 and Cs137 transport from fuel release to offsite boundary and evaluates dose at the 425m site boundary
- Model accounts for resuspension of deposited activity in PHTS during blow-down but neglects plate-out and resettling of nuclides following blow-down
- Model uses “GT MHR Preliminary Safety Assessment Report (DOE-GT-MHR-100230, Rev. 0, September 1994)”, Table 4.4.3-3 to scale from calculated Doses for I-131 and Cs-137 to total dose.
- Model addresses air Ingress and CO formation during core cool-down phase when thermal expansion of primary system changes to contraction
- A separate analysis addresses air ingress for an assumed fuel inlet pipe rupture at top of reactor vessel.

Radiological Release Case Matrix

- Design Basis Events (DBE) considered:
  - HPB breaks in CIP up to 100 mm equivalent diameter
  - Loss of forced circulation cooling assumed (DLOFC)
  - Volume of RBVV 10 % to 100 %
- Beyond Design Basis Events (BDBE) considered:
  - HPB breaks in CIP from 230 mm to 1000 mm
  - Loss of forced circulation cooling assumed (DLOFC)
  - For evaluation of margins and limiting RB Pressure
- RB Design Alternatives:
  - Consider RB design alternatives 1a, 1b, 2, 3a, 3b, 4a and 4b identified in previous slides

Radiological Release Case Matrix (continued)

- Special Cases:
  - A bounding case (alternative 0) that shows the impact of no radionuclide retention by the RB
  - Additional Alternative 4c: Alternative 4a/b with gross RB failure (puff release of 100% of activity in RB at worst point in time)
  - 1000 mm equivalent diameter SHTS break for limiting RB pressure response (larger He inventory than PHTS)
- Selected cases cover a spectrum of scenarios in-which the various RB design features (blow-out panels, filters, pressure retaining features) are successful and unsuccessful in performing their respective safety functions such as would be developed in a full PRA

Major Assumptions

- Normal operation core temperatures and initial radionuclide inventory from "NGNP and Hydrogen Production Preconceptual Design Study - Contamination Control (NGNP-NHS 50-CC, Rev. A, April 2008)
- Liftoff and resuspension based on model calculated shear force ratio
- Radionuclide release from fuel during DLOFC based on PBMR existing calculation for an existing 500MWt PBMR design
- These delayed fuel releases are conservative for break sizes less than 100mm due to enhanced convection cooling of the core which lowers peak temperatures during heat-up transient
- Mass transfer between hot and cold PHTS volume and between cold PHTS volume and RBVV based on loss of pressure blowdown and on expansion/contraction during heatup/cooldown phase

Major Assumptions (continued)

- No convection or buoyancy driven flow after depressurization:
  - All DBE breaks in PHTS outside reactor vessel are breaks from cold volume.
  - Circulator discharge check valve closes at end of blowdown phase (at the latest)
  - Flow path for buoyancy driven convection from core to break location requires flow reversal and counter-current flow with multiple up-down paths
  - Flow patterns are complex and do not favor buoyancy driven convection

- Breathing rate and weather X/Q for 24 to 96 hour time frame because significant core heatup releases from fuel do not start until > 24 hours.

Analytical Models

- Three volume model mass flow rate equations based on “Preliminary Safety Information document for the Standard MHTGR and responses to NRC Comments”, DOE-HTGR-86-024, Amendment 13, August 7, 1992.
  - Isentropic choked/un-choked equations used during depressurization
  - Ideal gas thermal expansion/contraction equations used after depressurization
- Decay of I131 modeled in all volumes and in release from the RBVV.

Analytical Models (continued)

- Lift-off of plated out activity model based on “Preliminary Safety Information Document for the Standard MHTGR and Responses to NRC Comments”, DOE-HTGR-86-024, Amendment 13, August 7, 1992.
  - 0.2% Cs137 and I131 lift-off for all shear force ratios less than 1.0.
  - 15% Cs137 lift-off for all shear force ratios greater than 1.0 and less than 30.0
  - 25% I131 lift-off for all shear force ratios greater than 1.0 and less than 30.0
- Dust re-suspension model based on existing PBMR analyses.
  - 0.02% Cs137 resuspension for all shear force ratios less than 1.0
  - 11% Cs137 resuspension for all shear force ratios greater than 1.0 and less than 10.0
  - 34% Cs137 resuspension for all shear force ratios greater than 10.0

Analytical Models (continued)

- Settling of I131 and Cs137 in the HPB was conservatively neglected however retention of gas-borne activity in the HPB was modeled due to thermal contraction during core cool-down phase
- Settling of I131 and Cs137 modeled in the RBVV based on IAEA TECDOC-978, “Fuel Performance and fission product behavior in gas cooled reactors”, Nov. 1997
  - I131 deposition constant = 0.3 hr^{-1}
  - Cs137 deposition constant = 0.1 hr^{-1}
- Dose calculation methodology based on NRC Reg. Guide 1.183, July 2000
- Dose conversion factors based on Federal Guidance Reports 11 (1988) and 12 (1993)

Analytical Models (continued)

- Constant breathing rate of 2.3E-4 m^3/s based on the >24 hour breathing rate in NRC Reg. Guide 1.183, July 2000.

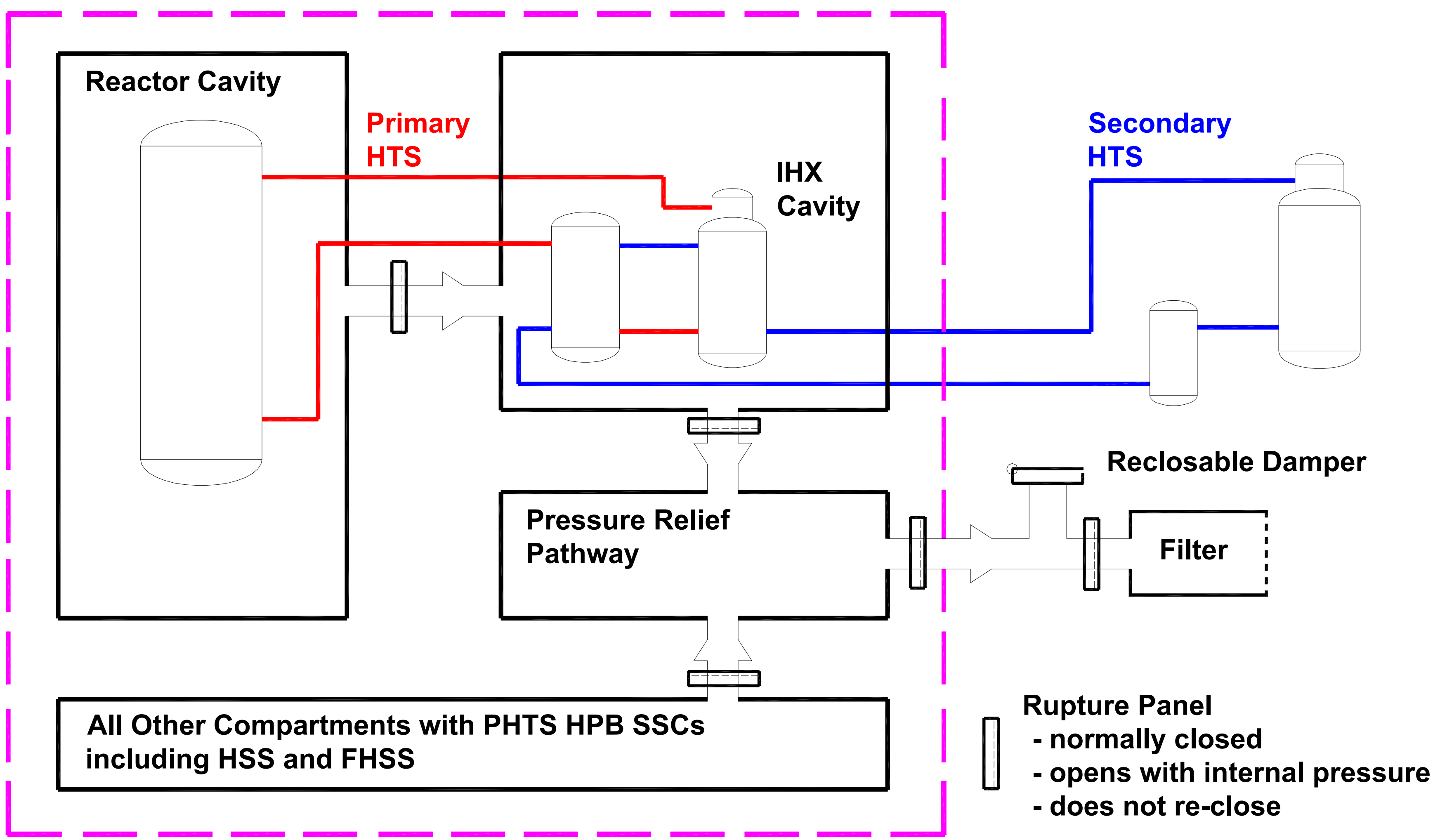
- Constant weather X/Q of 2.3E-5 s/m^3 based on 10% of the X/Q at 425 m in NRC Reg Guide 1.4, June 1974.

- Total doses from all radionuclides scaled based on "GT MHR Preliminary Safety Assessment Report (DOE-GT-MHR-100230, Rev. 0, September 1994)", Table 4.4.3-3.
  - Total thyroid dose from all radionuclides assumed to be a factor of 2 greater than I131 thyroid dose.
  - Total TEDE dose from all radionuclides assumed to be a factor of 2.5 greater than sum of Cs137 and I131 TEDE doses.

Radiological Results - No RB Case (Alternative 0)

- Assumes release from HPB is released to environment
- Limiting break size for DLOFC is 3mm
- TEDE 0.4 Rem, factor of 62.5 less than the 10CFR50.34 limit for DBEs; 40 % of PAG limit
  - No mitigation by a RB is needed to meet either limit
- Thyroid 10 Rem, 2 times PAG limit
  - Some mitigation by a RB is needed to meet the PAG limit

Analysis Model - Alternative 2



RB Model Input Parameters for RB Alternatives

- Base Model is RB Alternative 2
- RBVV = 100% is 100,000 m3
- All RB Alternatives are simulated with following inputs:

RB Option	Leak Rate (v%/d)	At Pressure (bar-d)	RB Vent Volume Fractions (% of RB)	Reclose-able Damper opening pressure (bar-a)	Reclose-able Damper reclosing pressure (bar-a)	Reclose-able Damper Flow Area, (m^2)	Filter Damper opening pressure (bar-a)	Filter and Filter Damper Flow Area, (m^2)	Filter Deconta-mination Factor for Cs (% removed)	Filter Deconta-mination Factor for I (% removed)
1a	100	0.1	10 & 20	> 0 (4)	<1 (2)	3	100 (1)	3	99	95
1b	100	0.1	10 & 20	1.113 incr	<1 (2)	3	100 (1)	3	99	95
2	50	0.1	10 & 20	1.213 incr	1.113 decr	3	(3)	3	99	95
3a	50	0.1	10 & 20	100 (1)	<1 (2)	3	1.033 incr	3	99	95
3b	50	0.1	50 & 100	100 (1)	<1 (2)	3	1.033 incr	3	99	95
4a	1	10	10 & 20	100 (1)	<1 (2)	3	100 (1)	3	99	95
4b	1	10	50 & 100	100 (1)	<1 (2)	3	100 (1)	3	99	95

(1) Damper never opens
(2) Damper never closes
(3) Filter damper opens when reclosable damper re-closes.
(4) Damper always open

Results - RB Alternatives with RBVV = 10%

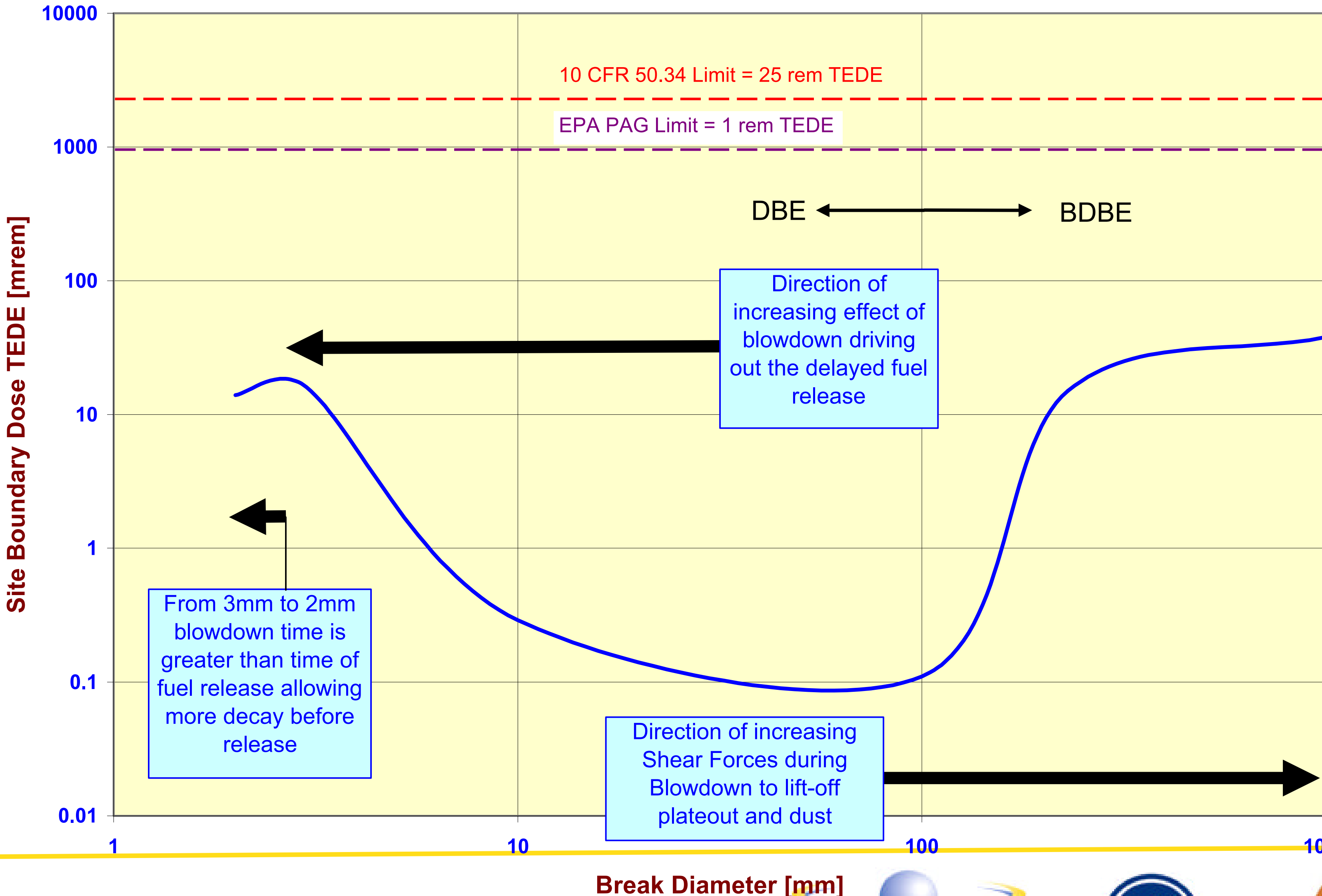
- 3 mm break size for DLOFC is always limiting
- Margin Factor is (PAG Limit Dose) / (Dose) (Min. TEDE or Thyroid)

Configuration	Leak Size	TEDE Dose	Thyroid Dose	Margin Factor
1a	3 mm	16 mrem	400 mrem	12.5 (Th)
1b	3 mm	16 mrem	400 mrem	12.5 (Th)
2	3 mm	1 mrem	20 mrem	250 (Th)
3a	3 mm	1 mrem	20 mrem	250 (Th)
4a	3 mm	1.4 mrem	33 mrem	150 (Th)

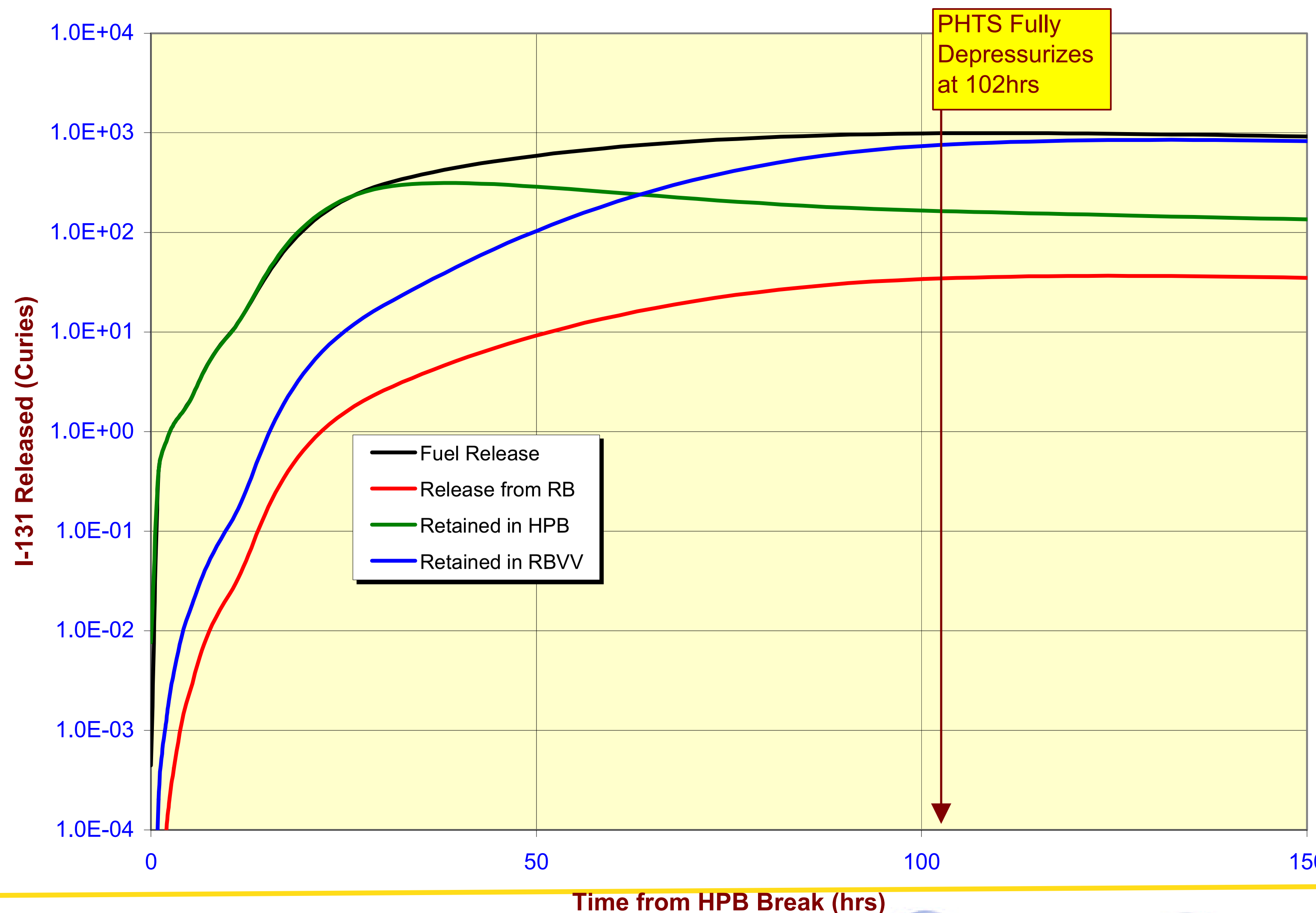
Results - Sensitivity Cases for RB Alternatives

Configuration	RBVV (m3)	Leak Size	TEDE Dose	Thyroid Dose	Margin Factor
3b	50,000	3 mm	.2 mrem	4 mrem	1250
4b	50,000	3 mm	1.3 mrem	30 mrem	167
3b	100,000	3 mm	.09 mrem	2 mrem	2500
4b	100,000	3 mm	1.1 mrem	25 mrem	200
4c (puff)	10,000	3 mm	723 mrem	18 rem	0.28
4c (puff)	100,000	3 mm	943 mrem	23 rem	0.22

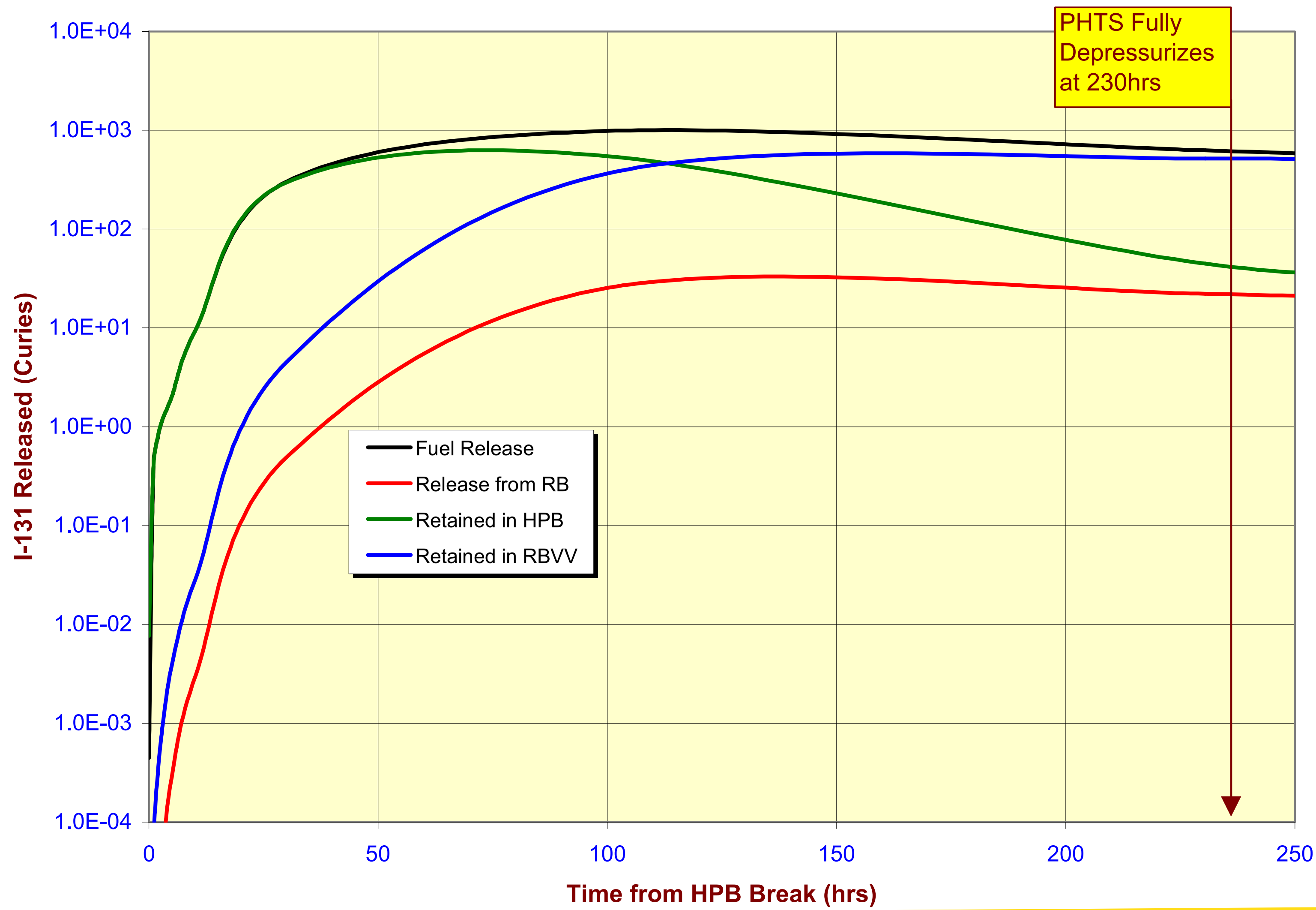
Dose Behavior vs. Break Size for Alternative 1a RBVV = 10%



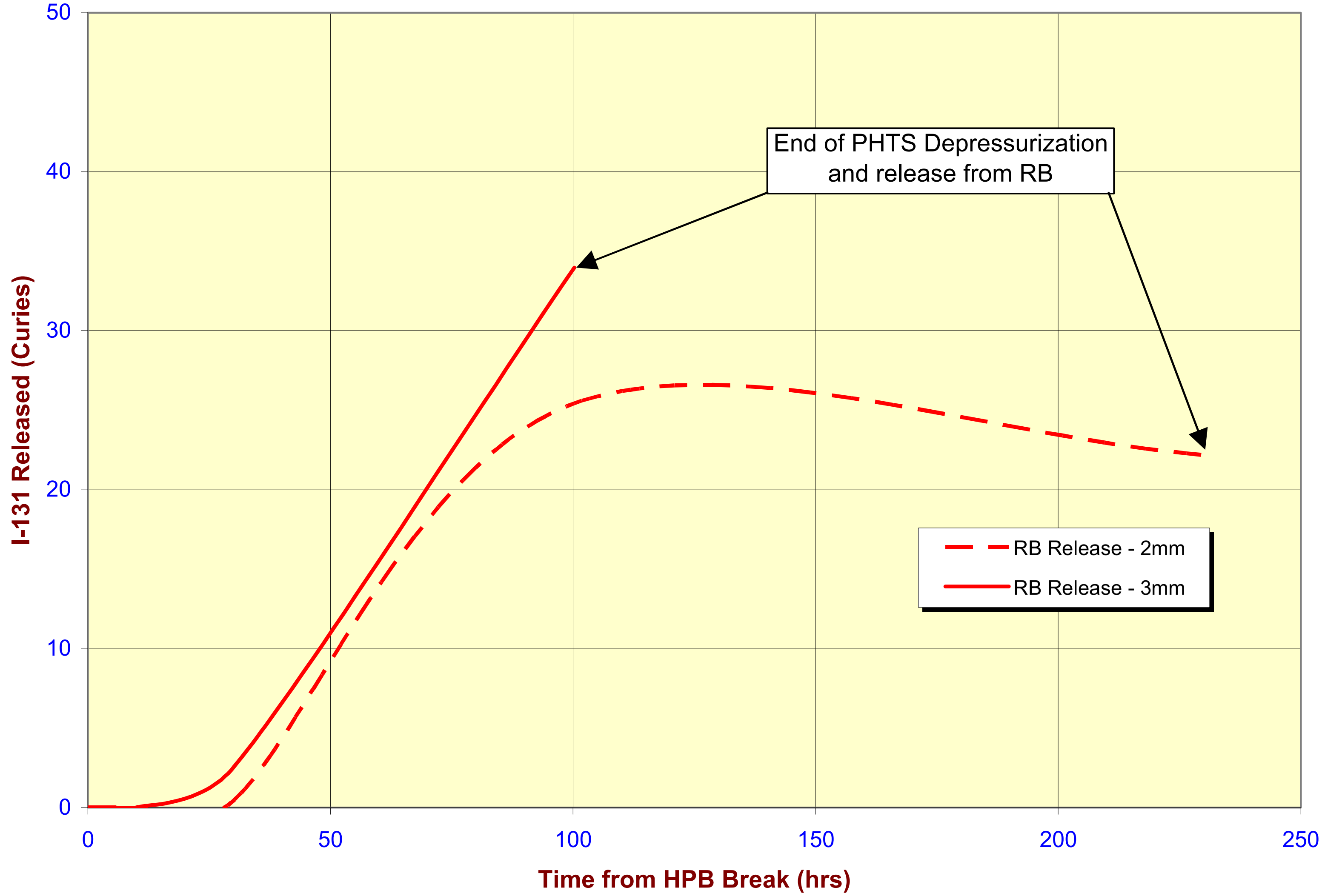
Time Dependent Releases of I-131
3mm HPB Break in CIP – Alternative 1a
RBVV = 10%



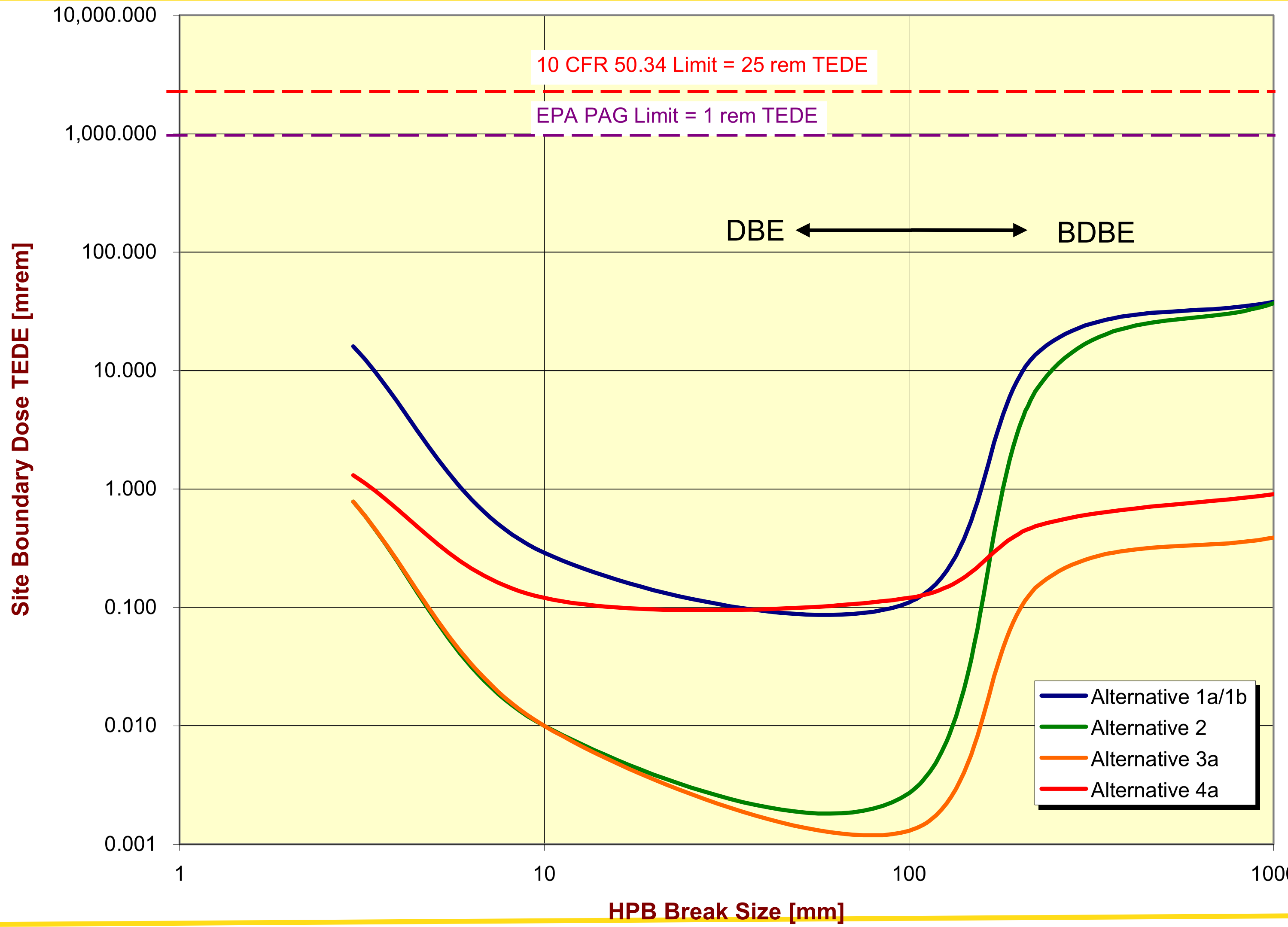
Time Dependent Releases of I-131
2mm HPB Break in CIP – Alternative 1a
RBVV = 10%



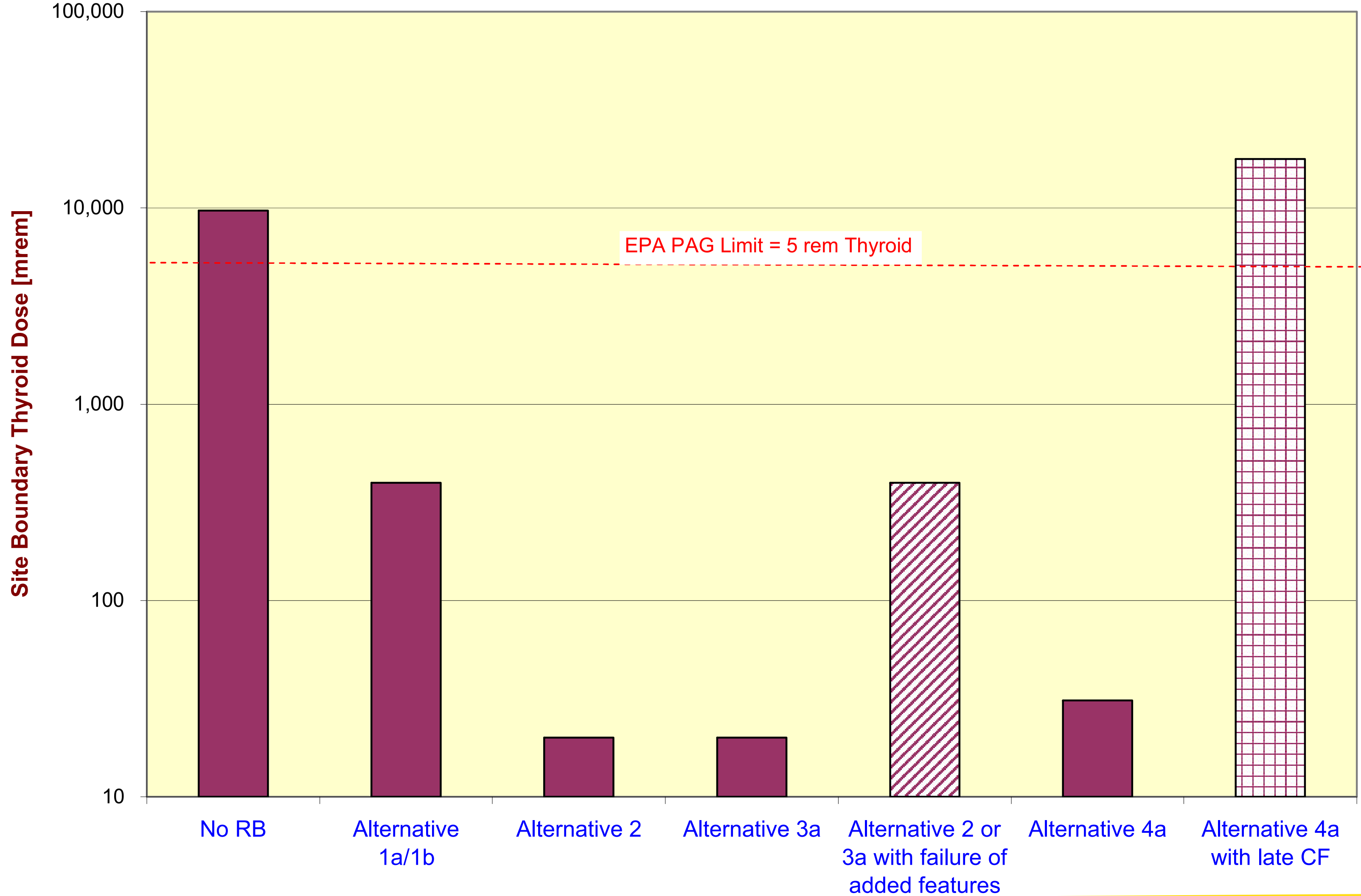
Comparison of I-131 Releases: 2mm vs 3mm HPB Breaks –Alternative 1a



Comparison of Dose vs. Break Size for Design Alternatives with Same RBVV = 10%



Comparison of Thyroid Dose vs. Design Options with Same RBVV = 10%



Conclusions Regarding Radiological Retention Capabilities of the Evaluated Alternatives

- Some mitigation by a RB is required to meet Thyroid PAG
- All evaluated RB alternatives provide ample margin to meet 10CFR50.34 and PAGs
- Alt 3a provides the most effective radiological retention for DLOFC across the break spectrum for the RBVV = 10% cases (doses for 3b somewhat lower than 3a due to increased volume)
- Alt 2 provides comparable radiological retention to Alt 3a and 3b for DLOFC for design basis break sizes
- Alt 1a and 1b provide an indication of the consequences of events where design features added in Alts 2, 3a and 3b fail (BDBEs)
- **Alt 4a and 4b provide less effective retention than Alt 3a and 3b across the full break spectrum and less than Alt 2 across the DBE spectrum because the RB is pressurized to drive out radionuclides for the duration of dose calculation despite a relatively low leak rate**
- Alt 4c was defined to simulate consequences of assumed delayed failure of RB capability (BDBE)
- **Alt 4c exceeds the dose limits for all the cases analyzed including the case with no reactor building, since PHTS is still pressurized at time of RB failure greater transport for radioactivity inside the RB and the HPB**

Case 1a Results - Details

RB Option	Vent Volume Fraction	Leak Size (mm)	Thyroid Dose (mrem)	TEDE Dose (mrem)	Max RBVV Pressure (bar)	Time to Depressurize to 1 atm (s or hr)	Time to Start Cooldown (hr)
1a	0.1	1000	7.7	38	4.3	16.2 s	43
1a	0.1	230	6	15	1.8	63 s	43
1a	0.1	100	0.11	0.11	1.125	334 s	43
1a	0.1	10	4.9	0.29	1.013	9.3 hr	43
1a	0.1	3	398	16	1.013	102 hr	102
1a	0.1	2	326	14	1.013	230 hr	230
1a	0.2	1000	6.5	32	2.8	28.6 s	43
1a	0.2	100	0.072	0.078	1.115	334 s	43
1a	0.2	3	206	8.5	1.013	102	102

Insights – Alternative 1a Results

- Initial inventory for release in blowdown is small
- Major activity source is activity release during heatup
- Release from fuel reaches maximum at 72 hours for I-131 and 100 hours for Cs-137
- 1% of total release from fuel is reached at 8 hours for I-131 and 15 hours for Cs-137
- For 10 mm leak size depressurization is complete at 9.3 hours
- For greater than 10 mm leak sizes the depressurization is complete before the major release from the fuel occurs.
- For greater than 10 mm leak sizes the doses are low because the release from fuel is stagnant in the hot volume without a driving force to the environment except expansion until cooldown starts.
- For greater than 10 mm leak sizes the cooldown starts at 43 hours and is not perturbed by blowdown.

Insights - Alternative 1a Results (Continued)

- For less than 10 mm leak sizes the blowdown extends into the heatup phase and delays the cooldown to 102 hours for a 3 mm leak and 230 hours for a 2 mm leak.
- For less than 10 m leak sizes the extended blowdown provides a significant convective mechanism to drive radionuclides out of the HPB during the major release phase.
- The major doses result for leak sizes less than 10 mm because of the extended blowdown transport of radionuclides to the RBVV.
- The slow blowdown rates allow major retention of radionuclides in the RBVV.
- Doubling the size of the RBVV reduces the maximum dose by almost a factor of two.

Design Basis Peak Pressures [100mm break size]

Alternative	RBVV %	Peak Pressure (Bar)
1a	10	1.125
1a	20	1.115
1b	10	1.138
1b	20	1.131
2	10 and 20	1.213 [Note 1]
3a	10	1.125
3a	20	1.116
3b	50	1.097
3b	100	1.079
4a	10	5.1
4a	20	3.1
4b	50	1.9
4b	100	1.4

Notes: 1. Reclosable damper opening pressure

Peak RBVV Pressure (continued)

- For DBEs the RBVV peak pressure is controlled by the 100 mm leak and by the RBVV leak rate.
- For Alts 1a and 1b (100%/d at 0.1 bard), and Alts 3a and 3b (50%/d at 0.1 bard) the RBVV peak pressure is below 0.125 bard.
- For Alt 2 the RBVV peak pressure is controlled by the opening pressure of the recloseable damper.
- For Alts 4 a and 4 b (1%/d at 10 bar) the peak pressure ranges from 1.4 bar to 5.1 bar. It decreases with increasing RBVV volume. Significant tradeoff between design pressure and volume.

Beyond Design Basis Event Margins

- 1000 mm Cold leg HPB break DLOFC is controlled by TEDE Dose and meets PAG limit with Margin Factor of 26.
- 1000 mm Cold leg HPB break DLOFC shows high RBVV pressure, up to 4.3 bar for Configurations 1a, 2 and 3a with 10% RBVV.
- With a low design pressure to meet DBE needs the BDBEs might behave more like a No RB Configuration.
- Still meets 10CFR50.34 Limits (25 Rem TEDE, 75 Rem Thyroid) by Margin Factor of 8 (Thyroid)

Air Ingress and CO Formation during Cold Leg HPB Break DLOFC Cooldown Phase

- Air Ingress into hot Graphite region can cause Oxidation of Graphite at elevated temperatures:
  - $2C+O_2 \Rightarrow 2CO + 0.1105$ kJ/KMol-O_2
- Air can be drawn into the reactor vessel by:
  - Buoyancy driven flow:
    - Hot Helium in core wants to rise and push cold Helium out through the break. Draw cold air-helium mixture from he RB into the vessel through the break.
    - Closed Circulator discharge check valve forces hot helium and cold air to flow in counter-current streamline flow past each other without mixing.

Air Ingress and CO Formation during DLOFC Cooldown Phase (continued)

- Flow path is very complex:
  - For breaks downstream of check valve, helium has to flow from the hot core up to the core inlet plenum, then down through the outer reflector to the cold lower plenum, then up along the core barrel to the inlet plenum at the top, then out through the cold inlet pipe to the break before the check valve. Cold helium-air mixture must be drawn into through the break and flow counter-current to the hot helium without mixing
  - For breaks upstream of check valve, helium has to flow against buoyancy down from the hot core to the hot lower core plenum, out through the hot outlet pipe to the break between the heat exchangers or between the second heat exchanger and the circulator. Cold helium-air mixture must be drawn in through the break and flow counter-current to the hot core without mixing.
  - Streamline counter-current flow without mixing in these complex geometrie are very unlikely. Buoyancy force in hot helium is 1/7th of same temperature buoyancy in air because of low he density.

Air Ingress and CO Formation during Small Cold Break DLOFC Cooldown Phase (continued)

- Contraction during cooldown phase
  - Cooldown phase does not start until 43 hours, longer for breaks < 10 mm.
  - Contraction of atmospheric pressure Helium in PHTS draws cold Air-Helium mixture into cold PHTS volume through break and mixes with Helium in cold PHTS volume.
  - Contraction of hot PHTS volume draws cold volume Air-Helium mixture into hot volume.
  - Oxygen reaching hot core can react with hot graphite.
  - Calculations with 3 volume model show that less than 2 Mol of CO is produced by 300 hours due to contraction effects alone.

Air Ingress through Fuel Inlet Pipe Break at Top of Reactor Vessel

- Configuration and Solution
  - One Fuel Tube per 120 º sector of core (3 tubes)
  - 65 mm diameter with ribs to guide 60 mm diameter pebbles
  - Postulate Fuel Tube Break at Top of Vessel
  - Blowdown of cold PHTS volume through core will tend to cool core down. PHTS is depressurized at 1000 seconds.
  - Hot Helium in core will rise through Fuel Tube and draw cold Air-Helium mixture in from RB.
  - Counter-current flow in straight vertical tube
  - Force Balance: Buoyancy force = Friction losses
  - Volume Flow balance: Volumetric flow rate of Helium out = Air in
  - Solved with double iteration on friction factors for Helium and for Air

Air Ingress through Fuel Tube Break at Top of Vessel (continued)

- Assumptions
  - At end of Blowdown assume Helium temperature in core and Air-Helium mixture in RB are both at 1000 ºC
  - Assume pure air in RB (conservative, maximizes buoyancy)
  - Assume 8 ribs of 2 mm height in fuel tube
  - Assume all air entering core reacts with Graphite to form CO
- Results
  - Helium upflow is laminar and occupies 86% of fuel tube flow area
  - Air downflow is turbulent and occupies 14% of fuel tube flow area
  - Volumetric upflow = downflow = 0.00151 m3/s
  - Air mass flow = air ingress rate = 4.18E-04 kg/s
  - Replaces Helium in core and upper and lower plenum in 10.3 hours
  - CO formation rate = 0.0283 Mol CO/s
  - Reaction energy = 1.56 mW (Compare: Decay Heat = 5 MW at 3 hrs)

Air Ingress through Fuel Tube Break at Top of Vessel (continued)

- Conclusions
  - Air Ingress through ruptured fuel tube is small
  - CO reaction energy is negligible compared to decay heat
  - Analysis is conservative (pure air in RB, equal temperatures, all O2 reacts to form CO)

Insights Regarding Pressure Capacities

- RB should withstand the higher of the blow-down pressure of 1.15 bar or the opening pressure of the re-closeable damper or filter damper for DBEs.
- RB should withstand about 5 bar if required to be intact for BDBEs.

Limitations and Areas for Further Research

- Conclusions regarding radiological retention capability of evaluated RB options are subject to limitations due to
  - Lack of design details and associated full scope PRA model
  - Need to evaluate different HPB break locations and a fuller set of licensing basis events
  - Need to consider the impact of natural convection on core temperatures during small leaks (2-10mm)
  - Lack of a fully integrated mechanistic source term model
  - Need to consider the quantitative failure probabilities of various design features as well as RB structural capability to withstand loads from a full set of licensing basis events
  - Lack of a full uncertainty analysis in the source term and consequence modeling
- These limitations should be addressed in the Conceptual and Preliminary Design Stages of the NGNP
- A detailed mechanistic code with integrated PHTS and RB models, in place of the simplified 3 volume model, will be required for the next phase of the analysis.

Integrated Evaluation of Alternative Reactor Building Designs

Approach to Evaluation of Alternatives

- Every alternative is presumed to be capable of meeting all non-safety functional requirements (mostly geometry and strength)
- Approximate pressure transients are calculated for several break sizes and locations
- Engineered features (rupture panels, filters, etc) and possible changes in building geometry and strength requirements are estimated for each alternative
- Radionuclide retention by inherent and engineered features is estimated for each alternative, and site boundary doses are estimated
- Capital costs are estimated for each alternative

Evaluation for Normal Operation Criteria

- Normal Operation Goals
  - Good operator access during operation and maintenance
  - Minimum operator exposure to radiation and other hazards
  - Minimum normal operating release of radionuclides to offsite public
- Evaluation of Alternatives
  - 1a: Adequate access and control of exposure; May not provide control of air activation products, and may require large HVAC flow -> score: 9
  - 1b: Adequate access and control of exposure; vent path normally isolated from environment improves control of air activation products and does not require large HVAC flow -> score: 10
  - 2: Same as 1b -> score: 10
  - 3a: Reduced leak rate requires airlocks and other impediments to operator access, otherwise, similar to 2 -> score: 8
  - 3b: Same as 3a -> score: 8
  - 4a and 4b: pressure retention requires additional features that impede operator access -> score: 3

Evaluation for Investment Protection Criteria

- Investment Protection Goals
  - Low forced outage rate and durations
  - Low risk of events that can cause damage to plant systems, structures, and components (SSCs)
  - Low risk of events that can result in significant plant outage time
  - Acceptably low risk of plant write-off
- Evaluation of Alternatives
  - Those with more systems and components have an increase in forced outages and an increase in downtime
  - Those with more stringent leakage requirements have an increase in outage durations
  - Leak tight options take more time to recover from radionuclide releases

Evaluation for Safety Evaluation for Safety Criteria: HPB Leaks/Breaks Response to Pressure Transient

- HPB Leaks and Breaks Design Goal
  - Maintain structural geometry of reactor and RCCS during pressure transient
- Evaluation of Alternatives
  - 1a: Vented design results in survivable pressure transient with high reliability-> score: 10
  - 1b: Rupture panels increase pressure transient slightly, help to isolate and protect HVAC -> score: 9
  - All other alternatives same score as 1b

Evaluation for Safety Criteria: HPB Leaks/Breaks Relative to Radionuclide Retention

- HPB Leaks and Breaks Design Goal
  - Provide retention of radionuclides released from HPB and fuel
- Evaluation of Alternatives
  - Scores correlate to radionuclide releases and offsite doses over spectrum of LBEs
  - 3a and 3b receive highest scores based on the results of the radiological release study, although they will have sequences in which the filters do not operate with the higher doses of 2, 1a, and 1b
  - 4a and 4b have events with release of both prompt and delayed fuel source term driven by pent-up non-condensable helium

Evaluation for Evaluation for Safety Criteria: Seismic

- Seismic Design Goal
  - Maintain geometry during design basis seismic events and capability for seismic BDBEs
- Evaluation of Alternatives
  - All alternatives designed for seismic with margin
  - All must respond with capability for beyond SSE events
  - Those with fewer filters and damper graded higher
  - Those with higher leak tightness more susceptible to seismic-induced cracks and leakage

This criteria impacted by degree of embedment

Evaluation for Evaluation for Safety Criteria: Hydrogen / Process Hazards

- Hydrogen Process Hazard Design Goals
  - Protect RB internals from shock/blast waves
  - Protect RB internals from chemical clouds/toxicity
- Evaluation of Alternatives
  - 1a: External pressure loading due to hydrogen explosion could be greater than pressure transient loading; Open vent path may weaken resistance to hydrogen explosion -> Score: 5
  - 1b: Same as 1a -> Score: 5
  - 2: Greater than open alternatives 1a and 1b -> Score: 8
  - 3a and 3b: More robust building means hazards not likely to control or impact design -> Score: 9
  - 4a and 4b: Even more robust building, without vent or filters, offers greatest resistance to hydrogen event hazards -> Score: 10

This criteria impacted by degree of embedment

Evaluation for Security / Airplane Crash Criteria

- Security Goals
  - Prevent malevolent intervention from impacting plant safety or operation
  - Provide protection of RB internals to airplane crash
- Evaluation of Alternatives
  - 1a and 1b have open vent pathway -> score: 5
  - All other alternatives -> Score: 10

This criteria may be impacted by degree of embedment

Evaluation for Cost Criteria

- Cost Goals
  - Minimize plant construction cost
  - Minimize operating costs
- Evaluation of Alternatives
  - 1a: Minimal features and loads -> Score: 10
  - 1b: Modest additional features -> Score: 9
  - 2: Additional damper and filter, not significant cost drivers -> Score: 8
  - 3a: Significant cost penalty for reduced leak rate. Structural design probably not controlled by pressure loads -> Score: 5
  - 3b: Similar to 3a, but with additional cost for expansion volume -> Score: 4
  - 4a: Large cost increment for low leak rate, pressure retaining capability -> Score: 2
  - 4b: Similar to 4a, but with modest reduction in cost due to lower pressure load; additional cost for expansion volume -> Score: 1

This criteria may be impacted by degree of embedment

Evaluation for Licensing Criteria

- Licensing Goals
  - Meet statutory limits for LBEs at site boundary
  - Meet PAG for LBEs for no evacuation at site boundary
- Criteria is not totally under our control and judgment at this time is highly subjective
- Links to the margins of the safety criteria and our ability to credibly present and defend our non-LWR safety design approach
- Evaluation of Alternatives
  - 1a: In part because of events with prior graphite moderated reactors, air ingress will be an issue with this open design
  - All other vented options are ranked based on their dose margins
  - Options 4a and 4b have the potential for the highest consequence sequences

Evaluated Score of Alternatives

Alternative	1a	1b	2	3a	3b	4a	4b
Normal operating requirements	9	10	10	9	8	3	3
Investment protection requirements	9	10	8	6	6	4	4
Safety requirements							
HBP leaks/breaks pressure response	10	9	9	9	9	9	9
HBP leaks/breaks radionuclide release	7	7	8	10	10	5	5
Seismic	10	10	9	8	8	6	6
Hydrogen/Process Hazards	5	5	8	9	9	10	10
Security / aircraft crash	5	5	10	10	10	10	10
Capital and operating cost	10	9	8	5	4	2	1
Licensability	4	4	9	10	10	6	6

Weighted Score of Alternatives

Alternative	Weight	1a	1b	2	3a	3b	4a	4b
Normal operating requirements	10	90	100	100	90	80	30	30
Investment protection requirements	5	45	50	40	30	30	20	20
Safety requirements								
HBP leaks/breaks pressure response	12	120	108	108	108	108	108	108
HBP leaks/breaks radionuclide release	8	56	56	64	80	80	40	40
Seismic	5	50	50	45	40	40	30	30
Hydrogen/Process Hazards	10	50	50	80	90	90	100	100
Security / aircraft crash	10	50	50	100	100	100	100	100
Capital and operating cost	25	250	225	200	125	100	50	25
Licensability	15	60	60	135	150	150	90	90
Total	**100**	**771**	**749**	**872**	**813**	**778**	**568**	**543**

Evaluation Summary of Vented vs Pressure-Retaining RB Options

- In general the vented options are superior to the pressure-retaining options based on the following considerations;
  - Greater compatibility with a non-condensable and inert primary coolant
  - Venting of the primary coolant inventory to atmosphere with or without filtration eliminates a driving force for subsequent radionuclide transport of the delayed fuel release
  - Better or comparable radionuclide retention capability for alternatives with range of design features
  - Lower costs for mitigation provided
  - Easier and less costly to engineer interfaces with RCCS, SHTS, FHSS, HSS, and other NHSS and auxiliary systems

Evaluation Summary of Vented RB Options

- Highest rating of integrated evaluation for alternatives examined was Alt 2 followed by Alt 3a
- Both alternatives provide superior radionuclide retention capability for design basis HPB breaks with DLOFC than pressure retaining alternatives; Alt 3b closely followed by 3a is superior in terms of radionuclide retention to all evaluated alternatives across the entire HPB break spectrum including AOOs, DBEs, and BDBEs
- Alts 2, 3a, and 3b are expected to have greater licensability than either of the open vented options 1a and 1b
- Another alternative for future study is a passive reclosable damper without a filter (Alt 1c). This is expected to have delayed fuel release retention capabilities approaching that of Alt 2

Radionuclide Retention Allocations for Recommended RB Alternative

- Results of the radionuclide retention study show that all the evaluated alternatives provide sufficient margins to offsite dose limits based on passive radionuclide retention within the fuel and HPB
- Added engineered features such as filters and re-closable dampers add additional margins
- This study confirms that radiological retention is not a required safety function but rather a supportive safety function for the NGNP reactor building
- It is recommended that a RB design goal be set for radionuclide retention capability of a factor of 10 reduction in releases from the HPB for I-131 and Cs-137 for DBE HPB breaks

Open Issues and Additional Engineering Studies

- Conclusions regarding radiological retention capability of evaluated RB options are subject to limitations due to
  - Lack of design details and associated full scope PRA model
  - Need to evaluate different HPB break locations and a fuller set of licensing basis events
  - Need to consider the impact of natural convection on core temperatures during small leaks (2-10mm)
  - Lack of a fully integrated mechanistic source term model
  - Need to consider the failure probabilities of various design features as well as RB structural capability to withstand loads from a full set of licensing basis events
  - Lack of a full uncertainty analysis in the source term and consequence modeling
- These limitations should be addressed in the Conceptual and Preliminary Design Stages of the NGNP supported by the NGNP PRA
- Key challenges identified for reactor building design
  - Optimization of RV vented volume dimensions vs performance and cost
  - Unknowns regarding needed protection against hydrogen process hazards
  - Unknowns regarding needed protection against physical security threats
  - Systems interactions issues associated with SHTS piping penetration RB walls
- Pending a more thorough design iteration, DDNs will be formulated leading to potential technology development primarily in the fuel, reactor, and HPB

Future Consideration of Building Interactions

- Key requirement for the RB is to provide physical separation of the NHSS from events and hazards associated with the HPS, PCS, and BOP facilities
- SHTS piping provides structural linkage between RB and adjacent buildings
- Need to investigate “systems interactions” involving
  - Faults in HPS or PCS propagating into RB
  - External events for which RB is protected but other buildings are not causing adverse interactions
- Need to provide high confidence of no adverse interactions for design basis events
- May lead to identification multiple large HPB breaks for BDBEs during the Conceptual Design PRA
- Key challenge in the next phase of the design
- Issue may complicate the approach to embedment