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Engineering Services for the Next Generation Nuclear Plant (NGNP) with Hydrogen Production

Reactor Containment, Embedment Depth, and Building Functions Study

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SUMMARY

The Next Generation Nuclear Plant (NGNP) is based on previous Modular High Temperature Gas Cooled Reactor (MHTGR) concepts that relied on a design strategy that is more based on the prevention of challenging events rather on their mitigation. The selection of key materials and parameters lead to a low power density reactor with a very large heat capacity, which results in slow transients that occur over time periods of days. The use of a graphite moderator, an uninsulated steel reactor vessel, and a passive decay heat rejection system make the system tolerant of loss-of-flow and loss-of-coolant accidents without compromising safety. The selection of ceramic coated particle fuel and an inert coolant eliminate the possibility of core meltdown. The simplicity and robustness of the design eliminates the need for operator actions during design basis events (DBEs) and is tolerant of operator mistakes.

For the NGNP, barriers to the release of radioactivity include the coated particle fuel, the fuel element structural graphite, the primary coolant pressure boundary, and the containment building. The General Atomics (GA) NGNP concept utilizes a Vented Low Pressure Containment (VLPC) building. The VLPC concept evaluated in this study is based on the Reactor Building (RB) design developed for the 450 MWt MHTGR steam-cycle plant. The VLPC has been a design choice for Modular Helium Reactors (MHRs) for over 20 years. As a result of the required very high radionuclide retention by the fuel, there is no need to have a high-pressure containment as required for Light Water Reactors (LWRs). The venting of pressure from the VLPC reduces the design requirements and cost of the building without compromising public safety.

This study is focused on the following areas:

1. RB design basis, including supporting the development of Technical and Functional Requirements (T&FRs).
2. Impacts of RB embedment on NGNP design and construction, including assessments of design, construction, functional and licensing considerations for different embedment options.
3. Radionuclide source terms.
4. Response of the VLPC to key events, including beyond DBEs that involve large-scale ingress of air.
5. Reactor building design alternatives, with the emphasis on design options that can reduce offsite doses during accidents and provide additional margins to account for uncertainties in radiological source terms.

Based on this study and studies performed for previous MHTGR concepts, the VLPC concept is recommended for the NGNP RB design. The different effects that embedment of the NGNP RB can have on the design, construction, maintenance and operation of the plant are evaluated and

discussed considering three possible alternatives where the RB is either fully embedded, partially embedded, or partially embedded with backfill. The most economical solution for the RB design is dependant on site specific conditions such as depth of rock, seismic conditions and elevation of water table. It is recommended that site investigations be performed at potential candidate sites so the RB embedment can be designed based on site-specific conditions. A greater embedment depth reduces the overall height of the RB above grade, which is driven by requirements for refueling equipment. Deeper embedment also provides greater protection against natural hazards and external threats.

VLPC design alternatives that can reduce doses at the Exclusion Area Boundary include filtered pathways on the RB and on the primary coolant pressure relief line, an elevated stack, and an extension of the Exclusion Area Boundary (EAB). Neither the capital costs, nor the O&M implications of VLPC design alternatives are expected to have a significant negative impact on the operability or cost of the NGNP.

If the NGNP includes a steam generator in the primary loop (as assumed for this study), accidents involving water-ingress are expected to result in the most severe radiological consequences. However, the doses at the EAB for these events are expected to be below the EPA Protective Action Guideline (PAG) limits and the radiological consequences can be further mitigated if one or more of the alternatives identified above are incorporated into the VLPC design. In particular, a filtered pathway on the primary coolant pressure relief line offers several advantages with only modest cost implications. In addition to significantly reducing radionuclide release to the environment during water ingress events, this design option can also improve worker safety by eliminating the possibility of discharge into the RB.

Because of the previous severe accidents at the Chernobyl and Windscale reactors, demonstrating the safety case for beyond DBEs with air ingress may become an important issue for the NGNP. As part of this study, independent assessments of a cross-vessel rupture event were performed by Fuji Electric Systems and Korea Atomic Energy Research Institute. Both of these assessments show that air ingress does not affect peak fuel temperatures reached during the accident and has a relatively small impact on the overall temperature response of the core during the accident. The total amount of graphite oxidation is limited to a few percent and is confined to the lower graphite structures and bottom-most layer of the active core. For these reasons, the incremental radiological consequences associated with air ingress and graphite oxidation should be small compared to heatup of the core, which is largely driven by decay heat. However, this event should continue to be analyzed in increasing detail, including more detailed modeling of oxidation in the lower graphite structures and assessments of the impacts of oxidation on structural integrity. Design measures to mitigate air ingress, including a counter-diffusion concept developed by Japan Atomic Energy Agency, should be evaluated in more detail.

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ACRONYMS AND ABBREVIATIONS

ALARA	As Low As Reasonably Achievable
ALWR	Advanced LWR
ANL	Argonne National Laboratory
AOO	Anticipated Operational Occurrence
ASME	American Society of Mechanical Engineers
ASTM	American Society of Testing and Materials
B&PV	Boiler and Pressure Vessel
BEA	Battelle Energy Alliance
CFD	Computational Fluid Dynamics
DBE	Design Basis Event
DCF	Dose Conversion Factor
D-i-D	Defense-in-Depth
DOE	U.S. Department of Energy
EAB	Exclusion Area Boundary
EPA	Environmental Protection Agency
EPRI	Electric Power Research Institute
EPZ	Emergency Planning Zone
FES	Fuji Electric Systems
FOAK	First Of A Kind
GA	General Atomics
GDC	10CFR50 Appendix A, General Design Criteria
GT-MHR	Gas Turbine Modular Helium Reactor
HPCC	High Pressure Conduction Cooldown
HTGR	High-Temperature, Gas-Cooled Reactor
HTTR	High Engineering Test Reactor
HVAC	Heating, Ventilation, and Air Conditioning
IHX	Intermediate Heat Exchanger
INL	Idaho National Laboratory

JAEA	Japan Atomic Energy Agency
KAERI	Korea Atomic Energy Research Institute
LANL	Los Alamos National Laboratory
LBE	Licensing Basis Event
LOCA	Loss of Coolant Accident
LOFA	Loss of Flow Accident
LPCC	Low Pressure Conduction Cooldown
LPZ	Low Population Zone
LWR	Light Water Reactor
MHR	Modular Helium Reactor
MHTGR	Modular HTGR
MWt	Megawatt thermal
NGNP	Next Generation Nuclear Plant
NPH	Natural Phenomenon Hazards
NRC	Nuclear Regulator Commission
O&M	Operations & Maintenance
OBE	Operating Basis Earthquake
PAG	Protective Action Guideline
PCS	Power Conversion System
PLT	Plant Level Requirement (defined in SRM)
PRA	Probabilistic Risk Assessment
PSID	Preliminary Safety Information Document
PSR	Permanent Side Reflector
QC	Quality Control
R&D	Research and Development
RAMI	Reliability, Availability, Maintainability, Inspectability
RB	Reactor Building
RCCS	Reactor Cavity Cooling System

RCPB	Reactor Coolant Pressure Boundary
RG	NRC Regulatory Guide
RPV	Reactor Pressure Vessel
RTNSS	Regulatory Treatment of Non-Safety Systems
SCAD	Sustained Counter Air Diffusion
SCS	Shutdown Cooling System
SG	Steam Generator
SRDC	Safety Related Design Condition
SRM	Systems Requirement Manual
SRP	NUREG-0800 Standard Review Plan
SSCs	Structures, Systems, and Components
SSE	Safe Shutdown Earthquake
T&FR	Technical and Functional Requirements
URS-WD	URS – Washington Division
URD	EPRI Utility Requirements Document
VCS	Vessel Cooling System
VHTR	Very High Temperature Reactor

1. INTRODUCTION AND BACKGROUND

The Next Generation Nuclear Plant (NGNP) is based on previous Modular High Temperature Gas Cooled Reactor (MHTGR) concepts that relied on a design strategy that is more based on the prevention of challenging events rather on their mitigation. The selection of key materials and parameters lead to a low power density reactor with a very large heat capacity, which results in slow transients that occur over time periods of days. The use of a graphite moderator, an uninsulated steel reactor vessel, and a passive decay heat rejection system make the system tolerant of loss-of-flow and loss-of-coolant accidents without compromising safety. The selection of ceramic coated particle fuel and an inert coolant eliminate the possibility of core meltdown. The simplicity and robustness of the design eliminates the need for operator actions during design basis events (DBEs) and is tolerant of operator mistakes.

The NGNP design is required to meet Nuclear Regulator Commission (NRC) licensing requirements and user/utility requirements, including the requirement that the plant avoid any intrusion into the normal day-to-day activities of the public, even during DBEs. In order to meet this requirement, offsite doses during any DBE must be less than the lower threshold dose limits at which the public would be required to evacuate or take shelter. The U.S. Environmental Protection Agency (EPA) has established Protective Action Guidelines (PAGs) that set these does limits at 5 rem to the thyroid and 1 rem to the whole body. This requirement has also been interpreted to mean that the frequency for any event that would require public evacuation or sheltering must be very low and below the cutoff frequency for DBEs, which has been set at 5×10^{-7} per plant year. In order to eliminate the requirement for public evacuation or sheltering for the NGNP, the Emergency Planning Zone (EPZ) and Exclusion Area Boundary (EAB) have been set to the plant boundary. The NGNP EAB has been set to 425 m [SRM 2007].

For the NGNP, barriers to the release of radioactivity include the coated particle fuel, the fuel element structural graphite, the primary coolant pressure boundary, and the containment building. The General Atomics (GA) NGNP concept utilizes a Vented Low Pressure Containment (VLPC) building. The VLPC concept evaluated in this study is based on the Reactor Building (RB) design developed for the 450 MWt MHTGR steam-cycle plant [Bechtel 1993], [Dilling 1993]. The VLPC has been a design choice for Modular Helium Reactors (MHRs) for over 20 years. As a result of the required very high radionuclide retention by the fuel, there is no need to have a high-pressure containment as required for Light Water Reactors (LWRs). The venting of pressure from the VLPC reduces the design requirements and cost of the building without compromising public safety.

This study is focused on the following areas:

1. RB design basis, including supporting the development of Technical and Functional Requirements (T&FRs).

2. Impacts of RB embedment on NGNP design and construction, including assessments of design, construction, functional and licensing considerations for different embedment options.
3. Radionuclide source terms.
4. Response of the VLPC to key events, including beyond DBEs that involve large-scale ingress of air.
5. Reactor building design alternatives, with the emphasis on design options that can reduce offsite doses during accidents and provide additional margins to account for uncertainties in radiological source terms.

The T&FRs are developed based on current NRC regulations and guidance, and on industry codes and standards applicable to reactor containment structures, systems and components (SSCs). Consideration is given to unique design features of the NGNP as compared to current domestic operating reactors, and anticipated changes in licensing policy reflected in NRC activities to develop a regulatory framework for advanced reactors.

The main factors considered in development of RB T&FRs are as follows:

1. Environmental conditions inside the RB (i.e., pressure, temperature, humidity, radiation, etc.) under normal, off-normal, and DBE conditions.
2. Containment effects on radiological source terms.
3. Effects of air ingress on calculated radiation dose rates during beyond DBEs and potential design features to mitigate air ingress.
4. Filtration and ventilation requirements.
5. Impacts of design basis threats and hazards on RB structure and configuration.
6. With regard to embedment, impact on:
    a. Impacts on operations (reactor protection, access for refueling, etc.).
    b. Reactor design.
    c. Ultimate heat sink.
    d. Site location, including consideration of geo-technical constraints, water tables, etc.
    e. Construction complexity.
    f. Cost.
    g. Design basis threats and Natural Phenomenon Hazards (NPH).
    h. Seismic performance/effects.

i. Others (e.g., RAMI, component repair/replacement, ALARA, fire protection, flooding, manual actions, interactive effects, etc.).

Major factors considered in this study that may affect the building functions and embedment considerations include:

1. Operational considerations including general access for maintenance and refueling.
2. Operating and accident conditions within the RB.
3. RB cooling systems.
4. Containment integrity – containment isolation and penetration.
5. Pre-service and in-service inspection requirements.
6. Access to equipment for maintenance and removal.
7. Key INL site parameters that may influence the cost and feasibility of embedment depths.
8. Construction aspects that may significantly impact the embedment considerations.
9. Heavy component transportation.
10. Site related water table considerations that may impact construction and installation.
11. Site specific geotechnical considerations that may significantly affect embedment.
12. Evaluation of malevolent hazards, such as aircraft impact, missiles, etc.
13. Natural Phenomenon Hazards (NPH) such as seismic, flooding, tornadoes, fires, etc. that may affect the building design and depth of the embedment.
14. Radiation shielding considerations.
15. Cost considerations, including impact of embedment depth, site location and geo-technical constraints on construction costs.

In some cases, the applicability of design considerations addressed in this study may be different for the first-of-a-kind (FOAK) NGNP and future commercial applications of the design. Such differences are noted in the study.

2. RB DESIGN BASIS AND TECHNICAL AND FUNCTIONAL REQUIREMENTS

2.1 Plant Configuration

For this study, the plant configuration is based on a 600 MWt steam-cycle plant that can produce steam for both electricity generation and process heat applications. The NGNP plant will also include a small (65 MWt) loop for producing very high temperature heat to support a nuclear hydrogen production mission. It is assumed that economic considerations will result in a plant configuration with a steam generator (SG) in the primary loop, which is consistent with previous HTGR plant designs, including Peach Bottom and Ft. St. Vrain. A schematic of the plant configuration assumed for this study is shown in Fig. 2-1. This configuration is described in more detail in [GA 2008a].

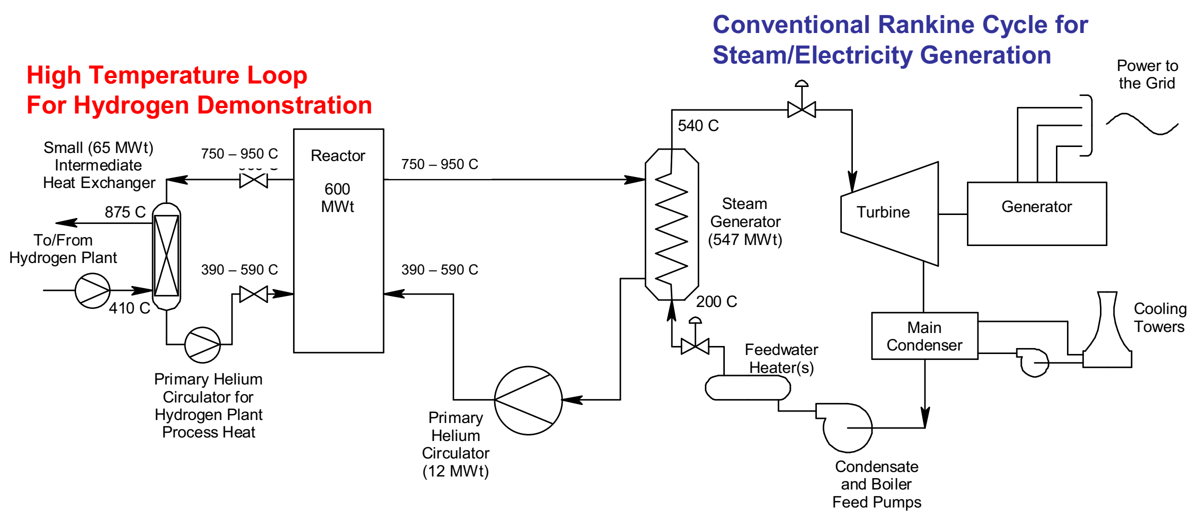

Figure 2-1. Plant Configuration Assumed for RB Study

2.2 VHTR Radionuclide Containment System

The VHTR radionuclide control philosophy and containment t system described below is taken from [Hanson 2008].

The dominant source of radionuclides in a VHTR is the fission product inventory in the reactor core. For modular HTGR designs, a hallmark philosophy has been adopted since the early 1980s to design the plant such that radionuclides would be retained in the core during normal operation and postulated accidents. The key to achieving this safety goal is reliance on TRISO-coated fuel particles for primary fission product containment at their source, along with passive cooling to assure the integrity of the coated particles is maintained even if the normal cooling systems were permanently disrupted.

In response to the above stated safety goal, a radionuclide containment system for a VHTR, which reflects a defense-in-depth philosophy, has been designed to significantly limit radionuclide release from the core to the environment during normal operation and for a spectrum of postulated accidents. A fundamental design requirement is to establish allowable limits on core releases during normal operation and postulated accidents such that all regulatory and user/utility requirements are satisfied.

As shown schematically in Figure 2-2, the five principal release barriers in an MHR radionuclide containment system are: (1) the fuel kernel, (2) the particle coatings (particularly the SiC coating), (3) the fuel element structural graphite, (4) the primary coolant pressure boundary, and (5) the VLPC. The effectiveness of these individual barriers in containing radionuclides depends upon a number of fundamental factors including the chemistry and half-lives of the various radionuclides, the service conditions, and irradiation effects. The effectiveness of these release barriers is also event specific.

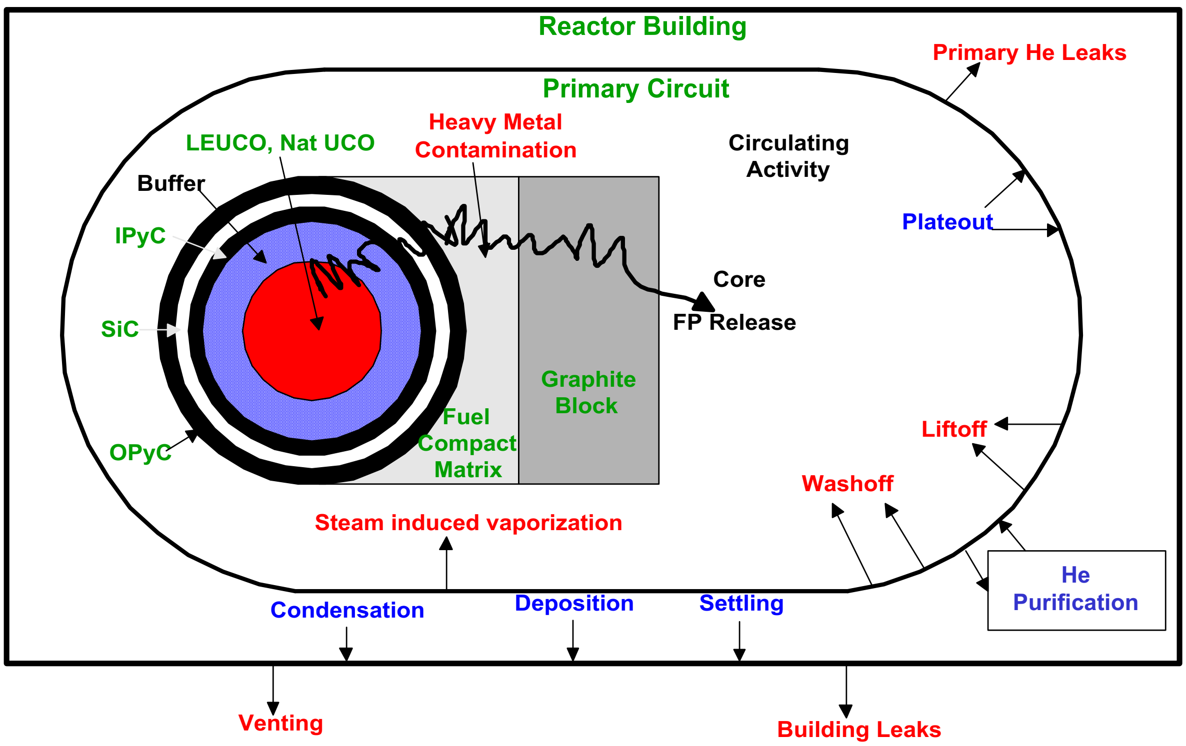

Figure 2-2. VHTR Radionuclide Containment System

The first barrier to fission product release is the fuel kernel itself. Under normal operating conditions, the kernel retains >95% of the radiologically important, short-lived fission gases such as Kr-88 and I-131. However, the effectiveness of a UCO kernel for retaining gases can be reduced at elevated temperatures or if an exposed kernel is hydrolyzed by reaction with water vapor (trace amounts during normal operation and much higher concentrations during water-ingress accidents). The retentivity of oxidic fuel kernels for long-lived, volatile fission metals such as Cs, Ag, and Sr is strongly dependent upon the temperature and the burnup.

The second - and most important - barrier to fission product release from the core is the silicon carbide and pyrocarbon coatings of each fuel particle. Both the SiC and PyC coatings provide a barrier to the release of fission gases. The SiC coating acts as the primary barrier to the release of metallic fission products because of the low solubilities and diffusion coefficients of fission metals in SiC; the PyC coatings are partially retentive of Cs at lower temperatures but provide little holdup of Ag and Sr.

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

The fuel-element graphite, which is denser and has a more ordered structure than the fuel-compact matrix, is somewhat less sorptive of the fission metals than the matrix, but it is more effective as a diffusion barrier than the latter. The effectiveness of the graphite as a release barrier decreases as the temperature increases. Under typical core conditions, the fuel element graphite attenuates the release of Cs from the core by an order of magnitude, and the Sr is essentially completely retained. The extent to which the graphite attenuates Ag release is not nearly as well characterized, and there is some evidence that the retention of Ag by graphite increases as the total system pressure increases (implying gas-phase transport through the interconnected pore structure of the graphite).

Typically, the two dominant sources of fission product release from the core are (1) as-manufactured heavy metal contamination (i.e., heavy metal outside the coated particles) and (2) particles whose coatings are defective or fail in service. In addition, the volatile metals (e.g., Cs, Ag, Sr) can, at sufficiently high temperatures for sufficiently long times, diffuse through the SiC coating and be released from intact TRISO particles; however, diffusive release from intact particles during normal operation is only significant compared to other sources for silver and tritium release. Fission products resulting from fissions in HM contamination outside of the particles are obviously not attenuated by the kernels or coatings, nor are the fission products produced in the kernels of failed particles appreciably attenuated by the failed coatings. In these cases, the fission products must be controlled by limiting the respective sources and by the fuel-element graphite in the case of the fission metals and actinides.

The fourth release barrier is the primary coolant pressure boundary. Once the fission products have been released from the core into the coolant, they are transported throughout the primary circuit by the helium coolant. The helium purification system efficiently removes both gaseous and metallic fission products from the primary coolant at a rate determined by the gas flow rate through the purification system (the primary purpose of the HPS is to control chemical impurities in the primary coolant). However, for the condensable fission products, the dominant removal mechanism is deposition ("plateout") on the various helium-wetted surfaces in the primary circuit (i.e., the deposition rate far exceeds the purification rate).

The plateout rate is determined by the mass transfer rates from the coolant to the fixed surfaces and by the sorptivities of the various materials of construction for the volatile fission products and by their service temperatures. Condensable radionuclides may also be transported throughout the primary circuit sorbed on particulates ("dust") which may be present in the primary coolant; the plateout distribution of these contaminated particulates may be considerably different than the distribution of radionuclides transported as atomic species.

The circulating and plateout activities in the primary coolant circuit are potential sources of environmental release in the event of primary coolant leaks or as a result of the venting of primary coolant in response to overpressuring of the primary circuit (e.g., in response to significant water ingress in a steam-cycle plant). The fraction of the circulating activity lost during such events is essentially the same as the fraction of the primary coolant that is released, although the radionuclide release can be mitigated by pump down through the Helium Purification System if the leak rate is sufficiently slow.

A small fraction of the plateout may also be re-entrained, or "lifted off," if the rate of depressurization is sufficiently rapid. The amount of fission product liftoff is expected to be strongly influenced by the amount of dust in the primary circuit as well as by the presence of friable surface films on primary circuit components which could possibly spall off during a rapid depressurization.

Other mechanisms which can potentially result in the removal and subsequent environmental release of primary circuit plateout activity are "steam-induced vaporization" and "washoff." In both cases, the vehicle for radionuclide release from the primary circuit is water which has entered the primary circuit. In principle, both water vapor and liquid water could partially remove plateout activity. However, even if a fraction of the plateout activity were removed from the fixed surfaces, there would be environmental release only in the case of venting of helium/steam from the primary circuit. For all but the largest water ingress events the pressure relief valve does not actuate. Moreover, the radiologically important nuclides, such as iodine and cesium, are expected to remain preferentially in the liquid water which remains inside the primary circuit.

The VLPC is the fifth barrier to the release of radionuclides to the environment. Its effectiveness as a release barrier is highly event-specific. The VLPC may be of limited value during rapid depressurization transients; however, it is of major importance during longer term, core conduction cool-down transients during which forced cooling is unavailable. Under such conditions, the natural removal mechanisms occurring in the VLPC, including condensation, fallout and plateout, serve to attenuate the release of condensable radionuclides, including radiologically important iodines.

2.3 Radionuclide Design Criteria

The GA philosophy for developing radionuclide design criteria is described below and taken from [Hanson 2008].

Standard GA design practice is to define a two-tier set of radionuclide design criteria, - referred to as “Maximum Expected” and “Design” criteria, - (or allowable core releases for normal operation and Anticipated Operational Occurrences); this practice has been followed since the design of the Peach Bottom 1 prototype U.S. HTGR up through the current commercial GT-MHR [Hanson 2002]. The “Design” criteria are derived from externally imposed requirements, such as site-boundary dose limits, occupational exposure limits, etc.; in principle, any of these radionuclide control requirements could be the most constraining for a given reactor design. The off-site PAG dose limits proved to be the most constraining for the 350 MW(t) steam-cycle MHTGR, and they will probably also be the most constraining for the NGNP.

Once the “Design” criteria have been derived from the radionuclide control requirements, the corresponding “Maximum Expected,” criteria are derived by dividing the “Design” criteria by an uncertainty factor, or design margin, to account for uncertainties in the design methods. This uncertainty factor is typically a factor of four for the release of fission gases from the core and a factor of 10 for the release of fission metals. The fuel and core are to be designed such that there is at least a 50% probability that the fission product release will be less than the “Maximum Expected” criteria and at least a 95% probability that the release will be less than the “Design” criteria. The GA approach to implementing such radionuclide design criteria is illustrated in Figure 2-3. (No particular scale is implied in this figure; it is simply a conceptual illustration of the approach.)

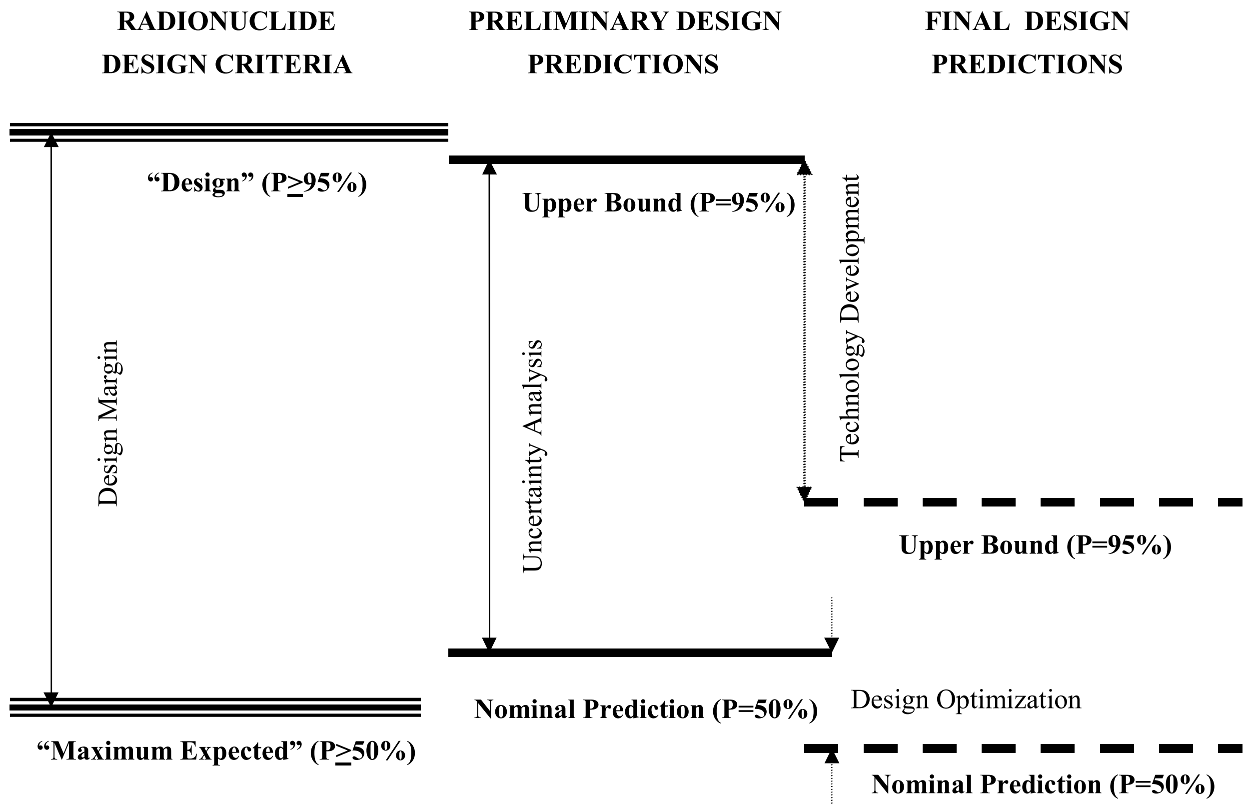

Figure 2-3. Radionuclide Design Criteria

As part of the design process, performance requirements must be derived for each of the five release barriers described in Section 2.2. Of these barriers, the particle coatings are the most important. Moreover, the in-reactor performance characteristics of coated-particle fuel are strongly influenced by its as-manufactured attributes. Consequently, the fuel performance requirements and fuel quality requirements must be systematically defined and controlled.

When the fuel requirements presented herein were derived, credit was taken for radionuclide retention by each of the release barriers. Barrier performance requirements are specified such that only the particle coatings are needed to meet 10CFR100 off-site dose limits; however, credit for the additional barriers is taken to meet the more stringent EPA PAG dose limits.

Overall, the most constraining radionuclide control requirement for the steam-cycle MHTGR was to comply with the dose limits specified in the EPA PAGs at the 425-m EAB so that the Emergency Planning Zone (EPZ) could be located at the EAB to preclude the need for public evacuation plans. The PAGs limit both whole body and thyroid doses; these dose limits were used to derive allowable environmental releases of noble gases and iodines, respectively, during Licensing Basis Events (LBEs). The limit on iodine-131 (the dominant iodine isotope) release from the plant was used to derive the limit on I-131 release from the core which, in turn, was used to set the limit on in-service fuel failure. Finally, this limit on in-service coating failure was used to derive the limits on certain as-manufactured defects, including the missing-buffer layer fraction.

The second, most constraining, top-level radionuclide control requirement for the steam-cycle MHTGR was to limit the occupational exposure to ≤10% of 10CFR20 (i.e., a factor of 10 ALARA margin was imposed on the design). A detailed occupational exposure assessment has not been performed for the GT-MHR (or for the NGNP). Hence, in deriving limits on plateout activity consistent with the subject goal, it was necessary to rely heavily upon previous occupational exposure assessments for earlier steam-cycle HTGR designs and upon engineering judgment. On that basis, it was projected that the ≤10% of 10CFR20 goal would be met if the radiation fields around the primary circuit due to fission product plateout were limited to ≤10 mR/hr for scheduled maintenance activities (e.g., IHX and circulator ISI, etc.) and to ≤100 mR/hr for unscheduled maintenance activities (e.g., steam-generator tube plugging, etc.). These limits on dose rates were in turn used to set limits on the primary circuit plateout inventories, in particular, limits on the releases of metallic fission products from the core, including Ag-110m, Cs-134, and Cs-137. Finally, the limits on Cs release from the core were used to derive limits on as-manufactured SiC defects.

For the NGNP conceptual design phase, it is recommended herein that the allowable in-service fuel failure limit and as-manufactured fuel quality requirements be maintained at the commercial GT-MHR limits despite the anticipated higher core outlet temperature for the NGNP (900-950 C versus 850 C). The primary reason for this recommendation is the necessity to limit the release of I-131 in order the meet the PAGs during depressurization and water-ingress accidents. However, it is recommended to increase the allowable Ag-110m fractional release by a factor of 2.5 (back to the steam-cycle MHTGR limit of 5.0×10^{-4}). The latter recommendation is based upon review of the available data on Ag release and the predicted Ag-110m release from a direct-cycle GT-MHR with Pu fuel and 850 C core outlet temperature. The preliminary radionuclide design criteria for the NGNP and commercial VHTR concepts are summarized in Table 2-1.

Table 2-1. Provisional VHTR Radionuclide Design Criteria

Parameter	Commercial GT-MHR		VHTR	
	$\geq$50% Confidence	$\geq$95% Confidence	$\geq$50% Confidence	$\geq$95% Confidence
As-Manufactured Fuel Quality				
HM contamination	$\leq 1.0 \times 10^{-5}$	$\leq 2.0 \times 10^{-5}$	[$\leq 1.0 \times 10^{-5}$]	[$\leq 2.0 \times 10^{-5}$]
Missing or defective buffer	$\leq 1.0 \times 10^{-5}$	$\leq 2.0 \times 10^{-5}$	[$\leq 1.0 \times 10^{-5}$]	[$\leq 2.0 \times 10^{-5}$]
Defective SiC	$\leq 5.0 \times 10^{-5}$	$\leq 1.0 \times 10^{-4}$	[$\leq 5.0 \times 10^{-5}$]	[$\leq 1.0 \times 10^{-4}$]
In-Service Fuel Performance				
Normal operation	$\leq 5.0 \times 10^{-5}$	$\leq 2.0 \times 10^{-4}$	[$\leq 1.0 \times 10^{-4}$]	[$\leq 4.0 \times 10^{-4}$]
Core heatup accidents	[$\leq 1.5 \times 10^{-4}$]	[$\leq 6.0 \times 10^{-4}$]	[$\leq 3.0 \times 10^{-4}$]	[$\leq 1.2 \times 10^{-3}$]
Metallic Core Release Limits				
Cs-137 core fractional release	1.0×10^{-5}	1.0×10^{-4}	[1.0×10^{-5}]	[1.0×10^{-4}]
Ag-110m core fractional release	2.0×10^{-4}	2.0×10^{-3}	[5.0×10^{-4}]	[5.0×10^{-3}]

2.4 Overview of VLPC Design Concept

The VLPC concept evaluated in this study is based on the RB design developed for the 450 MWt MHTGR steam-cycle plant [Bechtel 1993], [Dilling 1993]. The VLPC has been a design choice for Modular Helium Reactors for over 20 years. As a result of the required very high radionuclide retention by the fuel, there is no need to have a high-pressure containment as required for LWRs. The venting of pressure from the VLPC reduces the design requirements and cost of the building without compromising public safety. Figure 2-4 shows a cutaway view of the VLPC concept.

The VLPC is a multicelled, embedded structure constructed of cast-in-place reinforced concrete. The degree of embedment was selected to serve a number of objectives, which include ease of operation, minimization of shielding, good seismic performance, and reduced risk of sabotage. The operating floor of the reactor is set at site grade, with a common maintenance enclosure covering the operating area traversed by refueling equipment. There are two floors below grade with a rectangular footprint which are used to house mechanical, electrical, and instrumentation systems. A number of additional mechanical and electrical systems which do not require radiation shielding or protection from external hazards are designed to be delivered to the site as prefabricated modules and located at grade outside the maintenance enclosure.

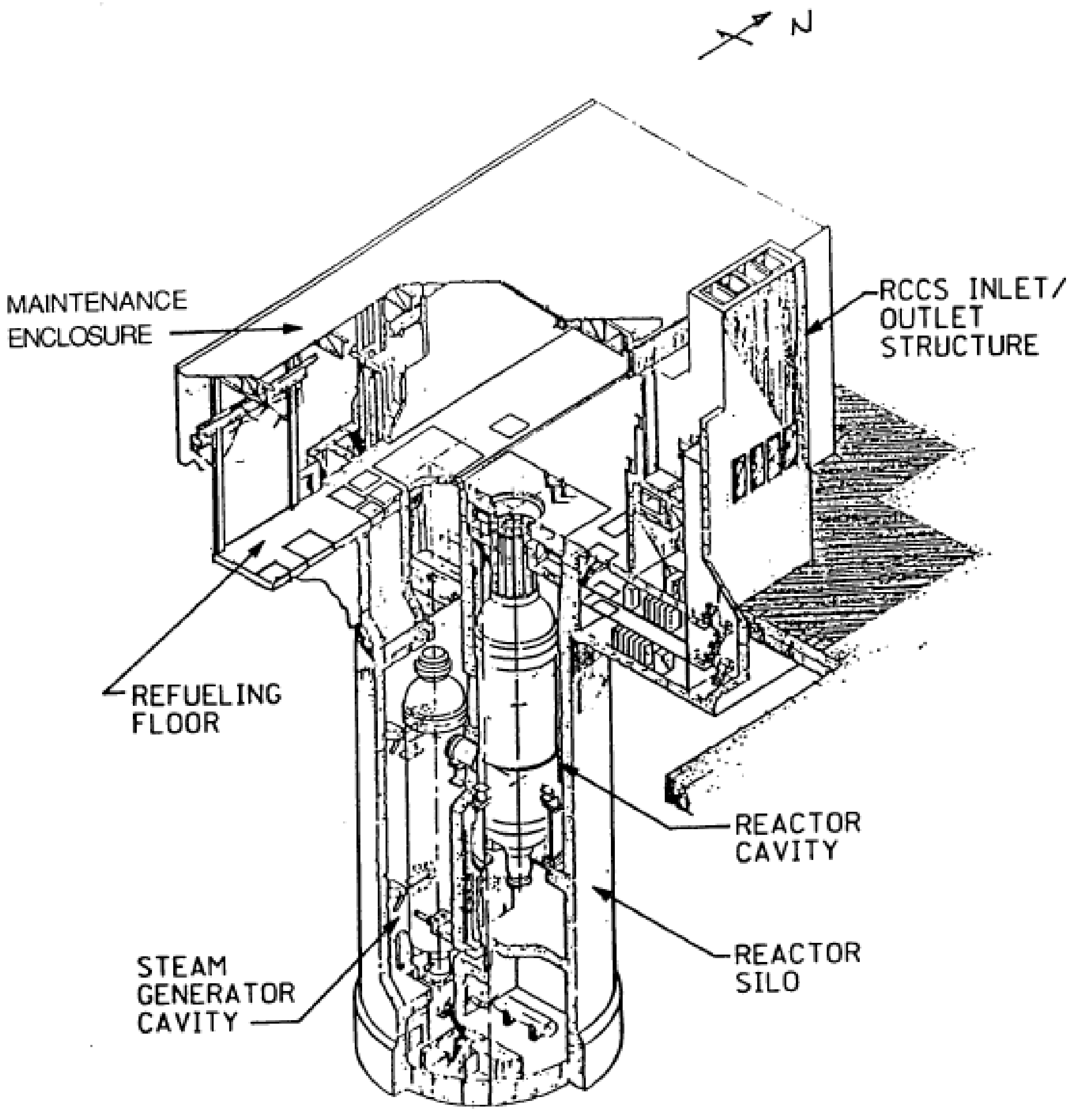

Figure 2-4. Cutaway View of VLPC Concept

The RB below elevation -30 ft is configured as a cylinder to enable it to resist soil and groundwater pressure. As shown in Fig. 2-5, the reactor, vessel, and heat transport systems are located within this space. The length of the steam generator generally controls the embedment depth of the RB. Assuming a single SG for a 600 MWt steam-cycle plant, the maximum embedment depth is expected to be in the range 140 – 150 ft. The silo depth must also accommodate the machinery used to service the shutdown cooling circulator and heat exchanger. Access to and from the cylindrical portion of the building for piping, electrical services, personnel, and the concentric Reactor Cavity Cooling System (RCCS) ducting is made from the rectangular portion of the building between elevations -30 ft and grade. Access for refueling and for major maintenance activities is from the operating floor. There are two extensions of the reinforced concrete RB above grade. On the west side of the RB, adjacent to the steam generator, an elevated portion of the building provides protection for the main steam relief valve discharge stacks and is part of the RB vent path. On the east side of the RB, the reinforced concrete portion of the building extends to elevation +95 ft 6 in. to serve as the RCCS elevated inlet-outlet structure.

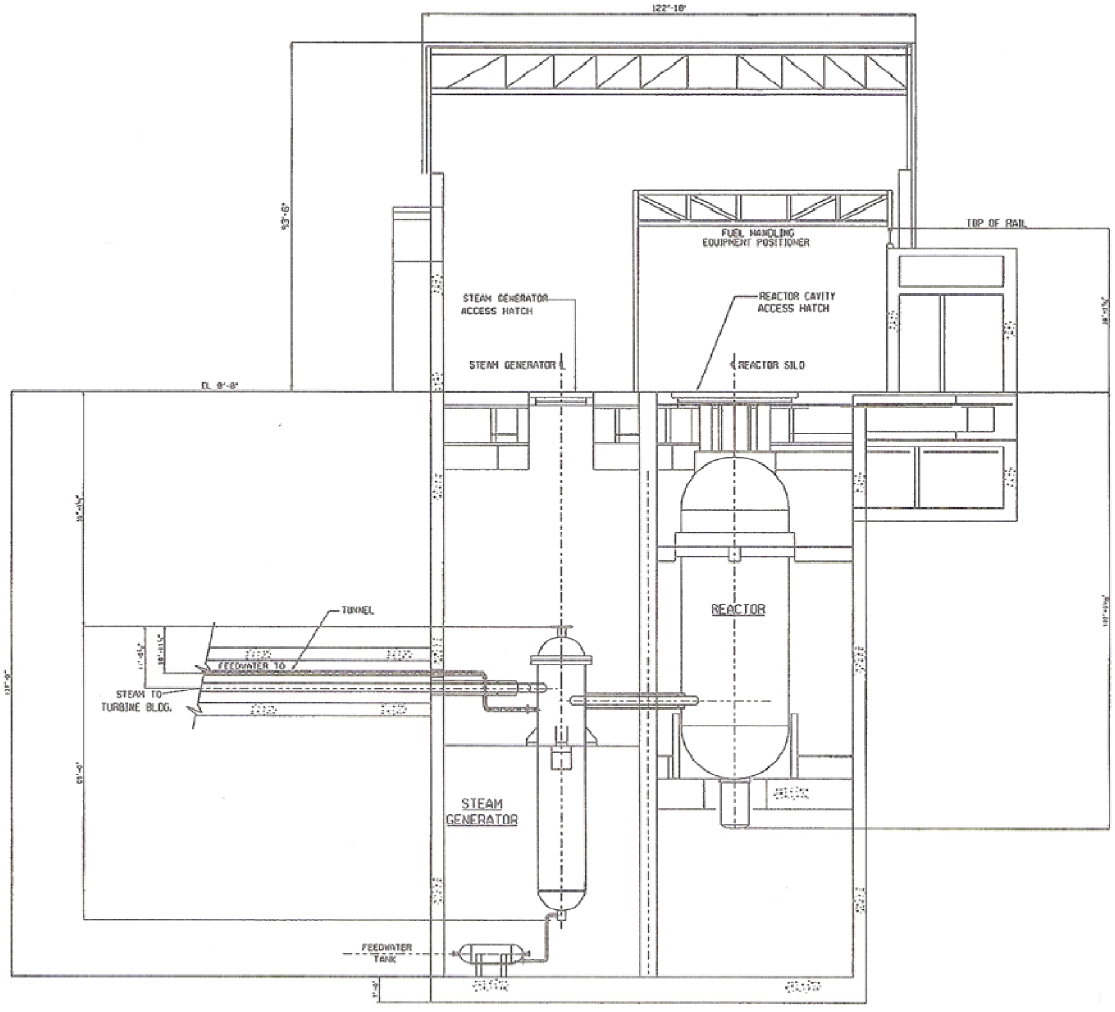

Figure 2-5. Reactor Building Elevation View

As shown in Fig. 2-6, the RB has been divided into two distinct zones for purposes of the heating, ventilating, and air conditioning (HVAC) design. The cells containing the Helium Purification Train, the vent path sections above grade west of the maintenance enclosure, and most of the cells in the cylindrical portion of the building have been designed to form a closed, interconnected space which is normally isolated from the environment. Air is recirculated internally and heat is removed by chilled water-cooled air handling units. The balance of the rectangular portion of the building, the personnel access stairways, the personnel elevator shaft into the silo portion of the building, and the space below the reactor vessel have been designed to be conditioned by a once-through flow of heated or cooled air.

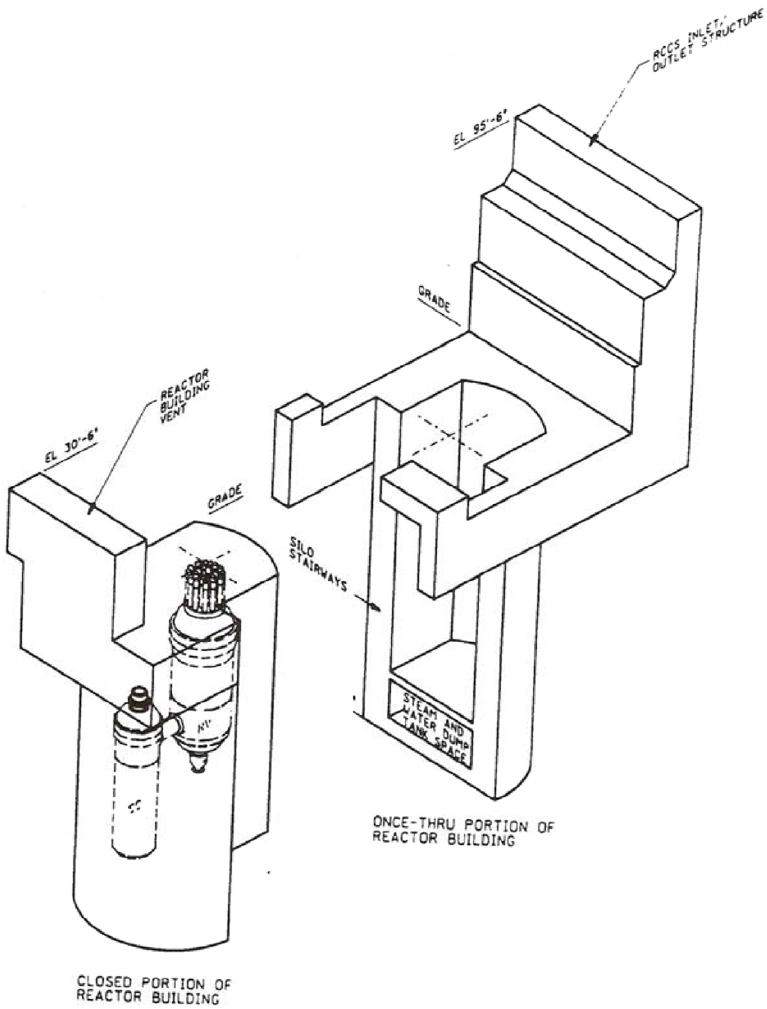

Figure 2-6. Reactor Building HVAC Strategy

The RCCS panels within the closed portion of the RB are regarded as part of the VLPC boundary. Air flowing inside the RCCS ducts and panels is considered to be outside the RB boundary. The walls, doors, plugs, and other barriers which separate the closed, recirculated portion of the building from the once-through cooled portion of the building or from the outside environment (including the RCCS panels and ducts) constitute the fifth containment barrier. Leakage from within this portion of the RB to the other part of the RB or to the environment has the potential to transport fission products from the containment to the environment. This space is also the portion of the RB which is affected by the specified building leak rate. The net free volume within this space is approximately 260,000 ft^3. This space is designed to have a leak rate of no greater than 1 volume per day at an internal pressurization of 1 psid, and to vent whenever the internal pressure exceeds 1 psid. It is expected that essentially none of the leakage which occurs will be from the surfaces of the building which are in contact with the soil, and that the specified leak rate represents an upper bound on the exchange which could occur between the building interior and the environment, since the pressure (and therefore the leakage) will normally decrease over the course of an accident. Architectural features such as doors, gaskets around floor plugs, and penetrations are important to establishing the building leak rate, and these features can be designed to achieve the specified leak rate.

The temperatures in the RB during normal operation are generally kept at the same level for all plant operating modes, except for the reactor cavity. HVAC systems maintain the temperature of the building. The reactor cavity is not ventilated during reactor operation, and heat is removed from the cavity by the RCCS. Concrete surfaces not protected by the RCCS panels or inlet/outlet plenums are insulated. During reactor shutdown, the reactor cavity is only ventilated when required for operator entry.

In the event of a large primary or secondary coolant leak within the closed portion of the RB, the internal pressure will exceed 1 psid. Gases are able to flow from any compartment through the building and out the vent relief valves or dampers to the atmosphere. If a break were to occur in the reactor cavity, helium would be able to flow through a shielded labyrinth into the steam generator compartment, via a one-way damper. Energetic pipe failures are more likely in the steam generator compartment. If a blowdown occurs in the steam generator area, gasses are able to flow downward to the bottom of the steam generator compartment, and then to the north side of the silo. The gases then follow the main feedwater and main steam lines upward to the building vent. The vent dampers are maintained in a closed position by gravity, and the weight of the damper plate determines the relief setpoint pressure, which is the internal pressure needed to open the damper. The relief setpoint pressure affects both the nominal building leak rate and the building pressure transients following a large primary or secondary coolant leak. The building relief setpoint pressure and vent opening area can both be adjusted if needed to obtain satisfactory performance during a pressure transient. The reinforced concrete building and RCCS panels have been designed to withstand pressure transient loadings of 10 psid.

Both the leakage across the RB boundary and the gases which are vented to the atmosphere via the RB vent are considered ground level releases.[1] Radionuclides released from the RB are assumed to travel a minimum of 425 meters to the EAB. As a consequence, the concentration of radionuclides is reduced in transit by atmospheric dispersion and, to a lesser extent, by

[1] One of the building alternatives under consideration is to utilize an elevated stack to enhance atmospheric dispersion before radionuclides reach the EAB.

deposition on the ground and radioactive decay. These effects are included in the assessments of radiological risks to the public during postulated accidents.

2.5 RB Technical, Safety, and Licensing Issues

2.5.1 Technical Functions and Requirements

High-level functions and requirements for the NGNP Reactor Complex (which includes the RB) are given in the NGNP System Requirements Manual [SRM 2007]. For the 450 MWt steam cycle plant, functions and requirements for the Reactor Complex are specified in more detail in [Bechtel 1993].

For this study, Washington Division (URS-WD) performed a systematic review of NRC regulations and the EPRI Utility Requirement Document [URD 1999] to assess technical, safety, and licensing issues that need to be addressed in order to develop the Technical and Functional Requirements (T&FRs) for the NGNP RB. As discussed in Section 2.5.4, design features have been correlated with issues and regulations/requirements into matrices that are included as Tables 2-2 and 2-3 at the end of Section 2.5 and recommended T&FRs based on these issues are presented in Table 2-3. Appendix A provides a summary of applicable federal regulations and guidance.

2.5.2 NRC Regulations Applicable to the RB

Technical and Functional Requirements (T&FRs) for the RB are developed with consideration given to NRC regulations applicable to the NGNP design. Title 10 of the Code of Federal Regulations (10CFR) is the governing set of regulations for licensing domestic nuclear reactors, including Class 103 licenses and certifications for commercial reactors. The principal NRC regulations that influence the RB design are found in

- 10CFR20, Standards for Protection against Radiation
- 10CFR50, Domestic Licensing of Production and Utilization Facilities
- 10CFR51, Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions
- 10CFR52, Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants
- 10CFR73, Physical Protection of Plants and Materials
- 10CFR100, Reactor Site Criteria

Detailed regulatory requirements considered applicable to the RB design are referenced in the Design Features-Issues matrix presented in Table 2-3.

At the time of this writing, 10CFR53, with which NRC plans to establish a risk-informed and performance-based licensing process for advanced reactors, remains reserved for future use. NRC regulations currently applicable to commercial nuclear reactors are focused on LWR designs. NRC efforts to establish a regulatory framework for licensing advanced non-LWR reactors are ongoing. Current NRC thinking with respect to licensing and regulation of advanced nuclear power plants is reflected in its draft statement of policy dated May 9, 2008 (73 FR 26349). The overarching NRC expectations for advanced reactors are consistent with the July 8, 1986 advanced reactor policy (51 FR 24643); specifically, the Commission expects at least the same degree of protection of the environment and of public health and safety, and common defense and security that is required for current generation LWRs, and expects advanced reactors to have "enhanced margins of safety and/or simplified, inherent, passive or other means to accomplish their safety and security functions." Defense-in-depth (D-i-D) remains a governing principle that guides the NRC development of advanced reactor licensing policy. D-i-D is expected to be a fundamental part of a risk-informed and performance-based non-LWR regulatory framework, as described in detail in NGNP IHX and Secondary Heat Transport Loop Alternatives Study [GA 2008b].

In its joint Report to Congress dated August 15, 2008, DOE and NRC recommended adoption of current NRC requirements with modification to support licensing of advanced reactors. NUREG-1860 is the NRC staff's feasibility study for the risk-informed and performance-based regulatory structure for future plant licensing. It includes an assessment of 10CFR50 applicability to non-LWR designs, with examples of proposed alternatives to the current regulations and use of the 10CFR52 combined license process. Although it does not represent a formal NRC position or consensus opinion, it provides useful insight to potential implementation of the regulatory framework under which the NGNP will be licensed. NUREG-1860 is referenced in the Table 2-2 Design Features-Issues Matrix, in cases where it provides examples of alternatives to existing regulations that may affect RB design.

2.5.3 Previous Licensing Experience for Modular Steam-Cycle Plants

Conceptual design information for the 350 MWt MHTGR was submitted to the NRC via the Preliminary Safety Information Document [PSID 1992]. The PSID was submitted for pre-application review, and did not result in any final licensing determinations for the MHTGR. However, NRC documented the results of its review of the PSID and subsequent DOE transmittals describing 350 MWt and 450 MWt MHTGR design concepts, in a Preliminary Safety Evaluation Report [PSER 1996]. The PSER identifies the following "licensability issues" as issues whose resolution may fundamentally alter the plant design:

1. Fuel Performance
2. Fission Product Transport Computer Codes
3. Source Term
4. Unconventional Containment
5. Safety Classification and Regulatory Treatment of Non-Safety-Grade Systems
6. Completely Passive System for Ultimate Heat Sink
7. Reactor Vessel Neutron Fluence Embrittlement

8. Reactor Vessel Elevated Temperature Service
9. Applied Technology Designation

Interdependence of Dose-Related Licensability Issues (Issues 1 through 5)

Licensability Issues 1 through 4 above are inter-related, and their ultimate resolution for the NGNP is critical to defining the RB T&FRs because they will determine occupational doses as well as the public dose consequences of radiological accidents. As described in [PSID 1992] and acknowledged in [PSER 1996] for the MHTGR design, the safety characteristics and mechanistic source terms for the Licensing Basis Events (LBEs) of the NGNP make the LWR-type containment unnecessary. The MHTGR RB provides an enclosure that can be vented in a controlled manner, acting as an additional attenuating barrier to radionuclide releases. Releases are filtered or contained during normal operation and in the long-term post-accident, but are released to the atmosphere when dampers open to relieve the pressure pulse following a helium or steam-line break. The VLPC envisaged for the NGNP [GA 2007] is similar to that of the previous MHTGR containment concept. Licensing of the NGNP VLPC design is expected to emphasize performance-based design criteria. The following criteria were deemed acceptable by NRC as stated in Section 4.2.4 of [PSER 1996]:

- Containment designs must be adequate to meet the onsite and offsite radionuclide release limits for {the event categories developed for accident selection and evaluation}.
- For approximately 24 hours following the onset of core damage, the specified containment challenge event results in no greater than the limiting containment leak rate used in evaluation of the event categories, and structural stresses are maintained within acceptable limits (i.e., American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code) Service Level C or D requirements or equivalent). After this period, the containment must prevent uncontrolled releases of radioactivity.

Successful demonstration of conformance to performance-based containment criteria for NGNP depends upon the use of mechanistic analysis of source terms, which, as stated in Section 5.2.8 of [PSER 1996] requires that:

- The performance of the reactor and fuel during normal and off-normal events are sufficiently well understood to permit mechanistic analysis.
- Transport of fission products can be adequately modeled for all pathways and barriers, including containment.
- Events considered in the analysis to develop the source terms bound severe accidents and design-dependent uncertainties.

Safety Classification

Licensability Issue 5 in [PSER 1996] pertains to safety classification of Structures, Systems and Components (SSCs). Criteria for determining SSCs' safety classification, as cited in Section 5.2.7 of the PSER, are contained in Section VI.a of 10CFR100, Appendix A. SSCs are safety-related, and designed to withstand the Safe Shutdown Earthquake (SSE), if they are necessary to assure

(i) the integrity of the reactor coolant pressure boundary,

(ii) the capability to shut down the reactor and maintain it in a safe condition, or

(iii) the capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the guideline exposures of this part.

The MHTGR SSC classifications proposed in [PSID 1992] focused only on criterion (iii), prevention or mitigation of accident dose consequences at the site boundary. Reactor coolant pressure boundary and containment functions were not identified as safety-related. NRC refers to defense-in-depth principles and licensing policy, as well as the need to review more detailed design information (e.g., as part of a pre-application license review or 10CFR52 application), as factors to address in order to resolve licensability issues associated with safety classification.

In its feasibility study for advanced reactor licensing (NUREG-1860), NRC staff provide insight to application of 10CFR50 criteria to non-LWRs. In the NUREG-1860 example of the proposed regulatory framework, General Design Criterion 1 in 10CFR50 Appendix A (GDC 1) would apply to advanced reactors, with minor modification to terminology. GDC 1 states the following:

> "*Criterion 1--Quality standards and records*. Structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. A quality assurance program shall be established and implemented in order to provide adequate assurance that these structures, systems, and components will satisfactorily perform their safety functions. Appropriate records of the design, fabrication, erection, and testing of structures, systems, and components important to safety shall be maintained by or under the control of the nuclear power unit licensee throughout the life of the unit."

Based on NRC licensability issues pertaining to safety classification of SSCs in the 350 MW MHTGR PSID, and the expected application of GDC 1 as suggested by NUREG-1860, this study assumes the NGNP will be subject to safety classification criteria that are equivalent to current LWR requirements.

2.5.4 Correlation of Design Features with Issues and Regulations

In order to develop RB T&FRs, reviews of current NRC regulations and guidance, including Revision 8 of the EPRI Utility Requirements Document for Advanced LWRs [URD 1999], are documented in Tables 2-2 and 2-3.

Table 2-2 correlates key RB civil and structural design features or parameters with URD requirements, with modifications recommended by URS-WD to comply with later NRC regulations and guidance and to accommodate a standard design that can be constructed in most of the central and eastern US.

Table 2-3 provides a Design Features-Issues Matrix with recommended RB T&FRs, based on a systematic review of applicable NRC regulations and guidance. Where applicable, references to advanced reactor licensing policy or forthcoming rule changes are identified for potential impact to T&FRs.

Table 2-2. Design Features Matrix Correlating Key RB Civil Structural Parameters to EPRI Utility Requirements Document

Key RB Civil/Structural Design Parameter	Based on URD	Based on URS-WD recommendations considering NRC regulations and guidance
MAXIMUM GROUND WATER LEVEL:	2 feet below grade	1 foot below grade
MAXIMUM FLOOD (OR TSUNAMI) LEVEL:	1 foot below plant grade	1 foot below plant grade
PRECIPITATION (FOR ROOF DESIGN):	• Maximum rainfall rate: 19.4 in/hr 6.2 in/5 min. • Maximum snow load: 50 lb/sq ft	• Maximum rainfall rate: 19.4 in/hr 6.3 in/5 min. • Maximum snow load: 75 lb/sq ft
AMBIENT DESIGN TEMPERATURES:	• 5% Exceedance Values Maximum: 95°F dry bulb/77°F coincident wet bulb 79°F wet bulb (non-coincident) Minimum: -5°F • 1% Exceedance Values Maximum: 100°F dry bulb/77°F coincident wet bulb 80°F wet bulb (non-coincident) Minimum: -10°F • 0% Exceedance Values (historical limit excluding peaks < 2 hours) Maximum: 115°F dry bulb/80°F coincident wet bulb 81°F wet bulb (non-coincident) Minimum: -40°F dry bulb	• 5% Exceedance Values Secondary HVAC Maximum: 95°F dry bulb/77°F coincident wet bulb 79°F wet bulb (non-coincident) Normal Plant Heat Sink Maximum: 95°F dry bulb/75°F coincident wet bulb 76°F wet bulb (non-coincident) Minimum: -5°F • 1% Exceedance Values Maximum: 100°F dry bulb/77°F coincident wet bulb 81°F wet bulb (non-coincident) Minimum: -10°F • 0% Exceedance Values (historical limit excluding peaks < 2 hours) Maximum: 115°F dry bulb/80°F coincident wet bulb 86°F wet bulb (non-coincident) Minimum: -40°F dry bulb
EXTREME WIND:	Basic wind speed: 110 mph Importance factors: 1.0 non safety structures 1.11 safety structures	Basic wind speed: 155 mph 3 sec gust at 33 ft above ground for exposure C Importance factors: 1.15

<table>
<tr><th>Key RB Civil/Structural Design Parameter</th><th>Based on URD</th><th>Based on URS-WD recommendations considering NRC regulations and guidance</th></tr>
<tr><td>TORNADO:</td><td>• Maximum Tornado Wind Speed: 300 mph
• Maximum Rotational Speed: 240 mph
• Maximum Translational Speed: 60 mph
• Radius of Maximum Rotational Speed: 150 ft
• Maximum Pressure Drop: 2.0 psi
• Rate of Pressure Drop: 1.2 psi/sec
• Missile Spectrum: Spectrum II of SRP 3.5.1.4
• Missile Velocities: Corresponding to Region II of Spectrum II</td><td>• Maximum Tornado Wind Speed: 230 mph
• Maximum Rotational Speed: 184 mph
• Maximum Translational Speed: 46 mph
• Radius of Maximum Rotational Speed: 150 ft
• Maximum Pressure Drop: 1.2 psi
• Rate of Pressure Drop: 0.5 psi/sec
• Missile Spectrum: Table 2 of RG 1.76, Rev. 1 (The specified missiles are assumed to have a vertical speed component equal to 2/3 of the horizontal speed.)
• Missile Velocities: Corresponding to Region I most severe</td></tr>
<tr><td>FOUNDATION SUPPORTING MEDIA:</td><td>• Maximum static bearing pressure demand: 15 ksf

• Foundation support conditions used for standard plant design should encompass the following site categories:

<table><tr><th>Site Category</th><th>Average Depth to Bedrock (feet)</th><th>Range of Depth to Bedrock (feet)</th></tr><tr><td>I</td><td>20</td><td>10-30</td></tr><tr><td>II</td><td>50</td><td>30-80</td></tr><tr><td>III</td><td>120</td><td>80-180</td></tr><tr><td>IV</td><td>250</td><td>180-400</td></tr><tr><td>V</td><td>500</td><td>>400</td></tr><tr><td>VI</td><td></td><td>0-10</td></tr></table></td><td>• Maximum average static bearing pressure of site specific demand or 15 ksf
• Maximum average dynamic bearing pressure of site specific demand or 95 ksf
• Foundation support conditions used for standard plant design should be verified by a site specific SSI analysis to properly represent the site specific soil conditions.</td></tr>
</table>

Key RB Civil/Structural Design Parameter	**Based on URD**	**Based on URS-WD recommendations considering NRC regulations and guidance**
FOUNDATION SUPPORTING MEDIA (Cont.):	Soil profiles designated below correspond to soil profile cases from EPRI report TR-102293, guidelines for Determining Design Basis Ground Motion, Volumes 1 through 4. Site Category Soil — Shear Wave Velocity Profile Cases I — Nos. 26, 32, 42, 43, 47, & 48 II — Nos. 7, 8, 19, 29, 30, & 42 III — Nos. 1, 9, 15, 20, 22, & 28 IV — Nos. 8, 18, 19, 22, 36, & 40 V — Nos. 16, 18, 24, 32, 34, & 49 • Representative mid range bedrock shear wave velocity: 6,000 fps • Upper range for rock shear wave velocity: 12,000 fps • Liquefaction potential: None (at Site-Specific SSE level)	The assumed soil conditions are suggested as follows for the central and eastern US: • Subsurface stability – minimum shear wave velocity of subgrade of 1,000 ft/s • Subsurface stability – shear wave velocity for defining rock 3,500 ft/s • Subsurface stability – shear wave velocity for defining firm to hard rock 6,500 ft/s • Subsurface stability – shear wave velocity for defining hard (competent) rock 9,200 ft/s • Subsurface stability – liquefaction potential None (for seismic category I structures)
SEISMOLOGY	• SSE PGA: 0.30g • SSE Design Response Spectra: per Reg. Guide 1.60 • SSE Time History: Envelope SSE Response Spectra	• SSE PGA: 0.30g defined at bottom of excavation • SSE Certified Design Response Spectra: per Reg. Guide 1.60 with modification to accommodate high frequency exceedances of central and eastern USA or site specific SSE Ground Motion Response Spectra. • SSE Time History: Envelope SSE Design Response Spectra • OBE ≤ SSE/3 • Potential for surface tectonic deformation at site: None within the exclusion area boundary

Table 2-3. NGNP RB Design Feature – Issue Matrix and Recommended T&FRs

Item #	RB Design Feature	Issue	Impact to T&FR	References
1	Safety Classification	Based on the RB's safety significance it should be classified as safety-related. Alternate classification of RB interfaces, or portions of the RB structure, may be supported by determination of safety significance. However, current safety classification requirements for the RB should be expected to remain essentially unchanged following the adoption of an advanced reactor regulatory framework. E.g., see NUREG-1860, Appendices H and J.	RB shall be designed, fabricated, erected, tested, maintained and operated to quality standards commensurate with its safety functions. RB is subject to quality assurance and quality control program requirements whose implementation shall ensure satisfactory performance of its safety functions.	GDC 1 10CFR50 Appendix B SRP 3.2.2 (systems) SRP 3.8 (structures) NUREG-1860
2	Regulatory Treatment of Non-Safety Systems (RTNSS)	A design that includes passive safety systems should define the active systems relied on for defense-in-depth as needed to meet passive plant safety goals and investment protection goals.	Non-safety related SSCs that support RB functions (e.g., HVAC systems, HTS control systems) should be subject to risk-based determination of safety significance, evaluated for adverse systems interactions, reliability/availability missions and regulatory oversight to ensure acceptable performance.	GDC 1 RG 1.206 C.IV.9 SECY 94-084 URD Vol. 3
3	RTNSS – VLPC function	RTNSS includes SSCs' functions needed to meet the containment performance goal, including	RB design should consider the use of non-safety SSCs, e.g., normal building HVAC,	RG 1.206 C.IV.9 SECY 93-087

Item #	RB Design Feature	Issue	Impact to T&FR	References
		containment bypass, during severe accidents This criterion for assessing containment performance is the degree to which the design comports with the Commission's probabilistic containment performance goal of 0.1 conditional containment failure probability (CCFP) when no credit is provided for the performance of the nonsafety-related, defense-in-depth systems for which there will be no regulatory oversight. The CCFP is a containment performance measure that provides perspectives on the degree to which the design has achieved a balance between core damage prevention and core damage mitigation.	that support VLPC containment goals, and apply RTNSS provisions to ensure adequate reliability and availability.	
4	Seismic Design	Based on RB functions of protecting safety-related equipment and containing releases of radionuclides, RB should be classified as a Seismic Category I structure. The seismic free-field ground motions can be developed based on Regulatory Guides 1.60, 1.165, or 1.208, and are used to develop the ground motion response	RB shall be designed to withstand, without loss of function, the effects of SSE (Seismic Category I). Seismic classification of interfacing SSCs should be based on safety significance and specific seismic criteria for the application (e.g., RG 1.189 for fire protection). Systems, other than	GDC 2 10CFR50 Appendix S 10CFR100.23 10CFR100 Appendix A SRP 3.2.1 SRP 3.7 SRP 3.8 RG 1.29 RG 1.60 RG 1.165 RG 1.189 RG 1.208

Item #	RB Design Feature	Issue	Impact to T&FR	References
		spectra (GMRS), and OBE, SSE ground motion for the design of SSCs.	radioactive waste management systems, that contain, or may contain, radioactive material and whose postulated failure would result in potential offsite whole body (or equivalent) doses that are more than 0.005 Sv (0.5 rem), should also be classified as Seismic Category I. [SRP 3.2.1]	
5	Seismic Design - OBE	When subjected to the effects of the OBE ground motion in combination with normal operating loads, all SSCs of the nuclear power plant necessary for continued operation without undue risk to the health and safety of the public must remain functional and within applicable stress, strain, and deformation limits. If OBE is set at one-third or less of the SSE, an explicit analysis or design is not required (SRP 3.7).	RB seismic design should specify OBE = SSE/3 Structural elements of the RB such as penetrations and bellows should be evaluated for fatigue resulting from OBE stress cycles.	10CFR50, Appendix S SRP 3.7 SECY 93-087
6	Seismic Design – fluid system boundaries	Boundary between Seismic Category I and non-Seismic Category I portions of fluid systems in the RB needs to be defined.	For fluid systems that are partially Seismic Category I, the Category I portion of the system should extend to the first seismic restraint beyond the isolation valves that	GDC 2 SRP 3.2.1

Item #	RB Design Feature	Issue	Impact to T&FR	References
			isolate the part that is Seismic Category I from the non-seismic portion of the system. At the interface between Seismic and non-Seismic Category I piping systems, the Seismic Category I dynamic analysis will be extended to either the first anchor point in the non-seismic system or to a sufficient distance in the non-seismic system so as not to degrade the validity of the Seismic Category I analysis.	
7	Seismic Design – Bounding Site Parameters	See parameters below	RB design should be based on SSE that is expected to be conservative with respect to potential commercial site locations (see Table 2-2)	GDC 2 SRP 2.5
7a	• Site soil parameters - Shear wave velocity: ≥ 305 meters per second	SRM Fig. 1 shear wave velocity ≥ 305 meters per second	See Table 2-2	SRM Figure 1
7b	• Site soil parameters - Bearing capacity	SRM Fig. 1 Bearing capacity ≥ 73,000 Kg per square meter	See Table 2-2	SRM Figure 1
7c	• Site Safe shutdown earthquake	SRM Fig. 1 SSE: 0.30 g	See Table 2-2	SRM Figure 1
7d	• Site earthquake Shutdown evaluation level	SRM Fig. 1 OBE: 0.10 g	See Table 2-2	SRM Figure 1
8	External Hazards –	RB design needs to consider	Design measures to address	GDC 2

Item #	RB Design Feature	Issue	Impact to T&FR	References
	Flooding ▪ Ground water ▪ Probable Maximum Flood ▪ Probable Maximum Precipitation (runoff)	ground water and other hydrodynamic effects, including pipe breaks outside the RB. Dewatering systems or may be required to support plant construction and operation. Passive design measures to address flooding effects should be sufficient to maintain RB integrity without reliance on active dewatering systems.	external flooding should include: Collection and drainage of rainwater runoff for the RB and surrounding structures. Site grading to divert precipitation away from the RB. High ground water level and maximum probable flood should be below plant grade (See Table 2-2) See Section 3 for considerations to maintain the embedded RB watertight.	SRP 2.4.12 SRP 3.4.2 RG 1.59 RG 1.102
9	External Hazards - Man-related External Hazards	RB design should consider the nature and proximity of man related hazards (e.g. airports, dams, transportation routes, military and chemical facilities) Design-basis events on site or in the vicinity of the nuclear plant are defined as accidents with a probability of occurrence on an order of magnitude of 10E-7 per year, with potential consequences sufficiently serious to affect the safety of the plant to the extent that 10CFR Part 100 guidelines could be exceeded. If	RB should be designed to maintain low risk of man related hazards for the anticipated range of potential site locations and plant applications, including onsite hydrogen production.	10CFR100.3 10CFR100.20(b) 10CFR50.34(a)(1) SRP 2.2 RG 1.91 EPRI NP-5283-SR-A

Item #	RB Design Feature	Issue	Impact to T&FR	References
		unfavorable physical characteristics exist, the proposed site may be found acceptable if the facility design includes appropriate and adequate engineering safeguards to compensate for the observed deficiencies. Regulatory Guide 1.91 provides guidance for evaluating postulated explosions on transportation routes near nuclear plants. Approaches similar to those in NRC-endorsed EPRI NP-5283-SR-A, "Guidelines for Permanent BWR Hydrogen Water Chemistry Installations," may be an appropriate alternative to RG 1.91 for evaluating hydrogen hazards at NGNP Data are often not available to enable the accurate calculation of probabilities because of the low probabilities associated with the events under consideration. Accordingly, the expected rate of occurrence of potential exposures in excess of the 10CFR50.34 (a)(1) requirements as they relate to the requirements of 10CFR Part 100 guidelines by an order of magnitude of 10E-6 per year is acceptable if, when		

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		combined with reasonable qualitative arguments, the realistic probability can be shown to be lower.		
10	External Hazards - Onsite Hydrogen Plant	Section 4.4 of [PCDSR 2007] specifies hydrogen plant conditions: 90 m separation distance to RB Plant is outdoors, not enclosed H2 underground storage tank, 100 kg @ 450 bar Truck loading area These parameters do not result in a significant impact to the RB, but are not necessarily representative of a commercial application. .	RB design should consider H2 hazard consistent with commercial application H2 hazard evaluation should include underground blast of hydrogen storage and/or piping and blast impulse transmission from soil to the RB. Consider chemical hazards e.g., SI, and oxygen	29CFR1910.103 NUREG/CR-6944 Vol 6
11	External Hazards – Design Basis Threat	10CFR73 requires protection against design basis threats, including radiological sabotage. NRC policy on regulation of advanced nuclear reactors is being revised to explicitly address security measures as part of the design phase (e.g., 73 FR 26349).	RB should include security features inherent in the design, e.g., Resistance to impact and blast pressure loads; Access controls including barriers, personnel accountability;	10CFR50.54hh (when issued) 10CFR73.1 SECY-08-0099 73 FR 26349

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			Visual monitoring by security force; Intrusion detection and alarm	
12	External Hazards – Design Basis Threat (aircraft impact)	Mitigative strategies and response procedures for potential or actual aircraft attacks should be expected to be a license condition for NGNP, based on ongoing rulemaking activities. Figure 48 of [PBMR 2006] shows results of a representative aircraft impact considering 2.7 ton impact without penetration and B777 impact with penetration of outside shell and no compromise to nuclear safety.	Aircraft impact hazards should be addressed by NGNP consistent with 10CFR50.54hh (when issued following ongoing rulemaking actions) and considered in the RB design.	10CFR50.54hh (when issued) SECY-08-0099
13	Internal Hazards Protection	SSCs important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing and postulated accidents.	RB design considerations to address internal hazards may include a combination of 1. Physical separation or system enclosure 2. Mitigation of postulated breaks using redundant design features and	GDC 2 (non-seismic piping failures) GDC 4 SRP 3.4 SRP 3.5 SRP 3.6

Item #	RB Design Feature	Issue	Impact to T&FR	References
			postulation of a single active failure in any required system 3. Failure Modes and Effects analysis	
14	Internal Hazards – applicability of combustible gas hazard	Applicability of combustible gas hazards inside the RB (separate from external hazards associated with onsite hydrogen production). NGNP core and fuel design are not susceptible to hydrogen generation phenomena applicable to LWRs. Graphite oxidation is a HTGR consideration for compliance with 10CFR50.44.	An evaluation shall be performed to determine the degree to which accidents involving hydrogen or other combustible gases inside the RB are technically relevant to NGNP design.	10CFR50.44 NUREG-1860
15	Internal Hazards – High- and Moderate-Energy line breaks	SSCs important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with postulated pipe rupture.	SSCs important to safety in the RB shall be designed to accommodate the effects of postulated high and moderate energy breaks. Design considerations may include a combination of physical separation or system enclosure, mitigation of postulated breaks using redundant design features and postulation of a single active failure in any required system, and Failure Modes and Effects Analysis (FMEA). RB should include provisions	GDC 4 SRP 3.4.1 (internal flooding) SRP 3.6.1 SRP 3.6.2

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			to protect against leakage from nonmechanistic cracks in the structure, penetrations and access openings located above internal flood level.	
16	Internal Hazards – Flooding ▪ Pipe ruptures ▪ Tank Failures ▪ Fire Suppression System Operation	The effects of potential flooding from safety-related and non-safety related piping/equipment failures should be evaluated. In addition to the postulated failures of seismic piping per SRP 3.6, circumferential failure of any non-seismic piping should be postulated (SRP 3.4.1). Effects of fire protection system operation is also a flooding consideration (SRP 3.4.1).	RB flooding studies should include transport via drainage pathways, localized effects such as water spray and impingement, as well as accumulations in the RB. Flooding studies should demonstrate no reliance on active mitigation systems is required to achieve and maintain safe shutdown. If mitigation is required then only seismically qualified SSCs should be credited.	GDC 2 SRP 3.4.1 SRP 3.6 (guidance for pipe break postulation)
17	Internal Hazards – reliance on Leak-Before-Break	LBB evaluation methods may be used to eliminate the dynamic effects of postulated pipe ruptures for ASME III piping systems. Application of LBB requires fracture mechanics analyses, consideration of creep and creep-fatigue, leak detection, inservice inspection. URD requires maximal use of LBB as an alternative to ANSI/ANS 58.2 for break postulation.	RB design is affected by determination of whether LBB can be successfully applied, e.g., pipe rupture loads on supports and RB internal structures, and other dynamic effects, may be eliminated from consideration using LBB.	GDC 4 SRP 3.6.3 RG 1.45 (LWR leak detection) URD Vol 3 Ch 1, 4.6.1 URD Vol 3 Ch 1 4.5.5.2 URD Vol III Ch 5, 8.2.2
18	Internal Hazards – Missile Protection	SSCs important to safety shall be protected against internally	The RB and SSCs therein that are important to safety	GDC 4 SRP 3.5.2

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		generated missiles.	shall be protected from internally-generated missiles to ensure compliance with GDC-4 requirements, including consideration of internally-generated missiles from (1) component overspeed failures; (2) missiles that could originate from high-energy fluid system failures, including missiles from pressurized components and systems for generating missiles such as valve bonnets and hardware-retaining bolts, relief valve parts, instrument wells and reactor vessel seal rings; and (3) missiles caused by or as a consequence of gravitational effects. If the combined probability of a missile occurrence, impact to a significant target and damage to the target is greater than 10E-7, then the identified missile is considered to be significant and requires protective measures.	

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			SSCs important to safety will be afforded protection by locating them in individual missile-proof structures, physically separating redundant systems or system components, or providing special protective shields or barriers.	
19	RB Environmental Conditions for SSCs	RB environmental conditions associated with normal operation, maintenance, testing and postulated accidents establish design conditions for SSCs contained therein, including Environmental Qualification (EQ) of electrical, mechanical and I&C equipment important to safety, and RB internal structures.	All equipment in the licensing analysis classified as safety significant needs to be qualified for the service conditions under which they are assumed to operate. These service conditions need to include those of normal operation, accident conditions and conditions associated with security related events. Acceptable qualification techniques include testing under actual service conditions or demonstrated service history under similar conditions. (Draft Example Requirement from NUREG-1860, Table J-5)	GDC 4 10CFR50.49 SRP 3.11 NUREG-1860

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20	Protection from Turbine Missiles	Assuming NGNP uses an indirect PCS with the turbine outside the RB, consideration of turbine missile impact to the RB can be avoided by favorable orientation.	The steam turbine used to generate electric power should be favorably oriented with respect to missile generation (i.e., the RB are excluded from the low-trajectory hazard zone described in RG 1.115).	GDC 4 SRP 3.5.1.3 RG 1.115
21	RB Concrete	NRC endorses the use of ASME III Div. 2 for concrete containments to meet GDC 50 containment design basis criteria. NUREG-1860, Appendices H and J, offers an alternative to GDC 50 but would require demonstration of radiological functional capability.	ASME III Division 2 and related standards should be considered for applicability to the RB. RB loading combinations encompassing construction, testing, normal and abnormal operation, design basis events (including internal and external hazards) and consideration of severe accidents, are input to the RB design.	GDC 1 GDC 2 GDC 4 GDC 50 NUREG-1860 SRP 3.8 RG 1.136 ASME III Div. 2
22	RB liner and penetrations	NRC endorses the use of ASME III Subsection NE for containment penetrations to meet GDC 50 containment design basis criteria. NUREG-1860, Appendices H and J, offers an alternative to GDC 50 but would require demonstration of radiological functional capability.	ASME III Subsection NE and related standards should be considered for applicability to major RB penetrations, or portions thereof (e.g., penetrations that are intended to resist pressure but are not backed by structural concrete, sleeved and unsleeved piping penetrations, mechanical and electrical penetrations, hatches and personnel locks).	GDC 1 GDC 2 GDC 4 GDC 50 NUREG-1860 SRP 3.8.2 RG 1.57

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23	VLPC boundary definition	Typical LWR reactor containment pressure boundary consists of those ferritic steel parts of the reactor containment system which sustain loading and provide a pressure boundary in the performance of the containment function under the operating, maintenance, testing and postulated accident conditions cited by GDC 51.	VLPC boundary definition and loading conditions should be derived for NGNP based on a design-specific risk assessment.	GDC 50 GDC 51 SRP 6.2 NUREG-1860 Table J-5
24	Fracture prevention of VLPC boundary	Per GDC 51, LWR containment boundaries are designed with sufficient margin to assure that under operating, maintenance, testing, and postulated accident conditions (1) its ferritic materials behave in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the containment boundary material during operation, maintenance, testing, and postulated accident conditions, and the uncertainties in determining (1) material properties, (2) residual, steady state, and transient stresses, and (3) size of flaws.	The VLPC boundary should be designed with sufficient margin to assure that under operating maintenance, testing, and frequent and infrequent events (1) its ferritic materials behave in a non-brittle manner, and (2) the probability of rapidly propagating fracture is minimized. NUREG-1860 suggests a risk based application of GDC 51 is appropriate for VHTRs.	GDC 51 SRP 6.2.7. NUREG-1860

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25	Containment isolation – fluid systems	LWR regulations and guidance require redundant, safety related isolation capability of fluid systems penetrating containment. RB isolation features for NGNP should be developed using design specific functional requirements with consideration of LWR requirements.	RB isolation provisions for fluid systems should be designed to meet RB isolation functional requirements e.g., as defined in Section 4.2 of [GA 2008a]	GDC 54 GDC 55 GDC 56 GDC 57 SRP 6.2.4 NUREG-1860
26	Containment isolation – instrument lines	Regulatory Guide (RG) 1.11 describes acceptable containment isolation provisions for instrument lines. In addition, instrument lines closed both inside and outside containment are designed to withstand pressure and temperature conditions following a loss-of-coolant accident (LOCA) and dynamic effects are acceptable without isolation valves.	Instrument lines penetrating the RB should be closed both inside and outside containment, i.e., designed to withstand pressure and temperature conditions following events requiring RB isolation.	SRP 6.2.4 RG 1.11
27	Containment isolation – Type C local leak rate testing	Isolation valve leakage is included in determining overall leakage against VLPC performance requirements. NUREG-1860 considers 10CFR50 Appendix J containment leak test programs to be LWR-specific, but applies leak test requirements to the advanced reactor framework.	All containment isolation valves requiring a leak rate test should be included in a test program. Test pressures for isolation valve Type C tests should be included in the test program and technical specifications.	10CFR50 App J SRP 6.2.6 NUREG-1860
28	Containment isolation – Bypass leakage	Containment bypass can result from intersystem leakage from the primary HTS. LWR experience shows that containment bypass can be an important risk	Licensing basis events and severe accidents should be evaluated for their potential risk due to containment bypass. Bypass leakage	SRP 6.2.6 SRP 19.1.4 SRP 19.2 URD Vol III Ch 1 App A, 4.4

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		contributor	should be quantified and assessed against containment performance requirements.	(SRP 6.2.3 and BTP 6-3 address bypass specifically for dual containments, that would likely not apply to NGNP)
29	Containment Isolation – penetration leakage	RB penetrations are potential leakage pathways and are included in determining overall leakage. LWR experience has shown that penetrations (including hatches) are important contributors to overall containment leakage.	RB penetrations shall be accounted for in the evaluation of VLPC leakage paths and should be included in a leak test program. PRA may be useful in eliminating specific penetrations from further consideration.	10CFR50 App J SRP 6.2.6 URD Vol III Ch 1 App A, 4.3 NUREG-1860
30	Containment isolation – electrical penetrations	RB electrical penetration assemblies must maintain the mechanical integrity of the VLPC.	RB electric penetration assemblies shall be designed to withstand events requiring RB isolation, and withstand the full range of fault current (minimum to maximum) available at the penetration. Electrical penetrations should be designed to be leak tight and accessible for maintenance.	GDC 50 SRP 8.3.1 RG 1.63 IEEE-242 IEEE-317 IEEE-741 URD Vol 3 T 1.8-3
31	Mechanistic containment leakage	The typical fixed value of long term containment leakage may be replaced by more realistic values based on pressure.	Consider justifying credit for variation in VLPC leakage as a function of pressure if this is supported by appropriate test data.	10CFR50 App J NUREG-1860

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32	Containment hatches, access plates, airlocks	The potential for releases to occur due to failure of some penetrations to be isolated or properly sealed has been found to be important in previous studies. In particular, large leakage paths may be available, e.g., via equipment hatches.	Attachments to the VLPC boundary including hatches, plates and airlocks, should be designed and controlled to allow access for personnel and equipment, including major component removal, while maintaining leakage within design limits	SRP 6.2 URD Vol 3 Ch 1, 4.3
33	Penetration seals	RB penetrations should be sealed to maintain leakage integrity during design basis events.	Seals provided at the penetrations must be designed to maintain containment integrity for design-basis accident conditions, including pressure, temperature, and radiation.	SRP 3.8.1 SRP 6.2
34	Containment integrity – additional measures to improve integrity	LWR experience has identified potential means of improving assurance of containment integrity and low dose, some of which may be applied to NGNP, e.g., Passive containment heat removal; Minimize penetrations, especially open those that are normally open; Increased design pressure of systems which interface with the RCS; Gross, periodic, on-line containment integrity check; Significantly reduced leakage through leak paths directly from primary containment to the environment; Secondary building design to hold up fission product leakage from	Consider additional measures, to minimize challenges to VLPC integrity and assure low dose releases.	URD Vol III Ch. 1 App B, 2.1.3.2.4.2 URD Vol III, Ch. 5

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		the primary containment.		
35	Containment penetrations - embedment depth	Elevation of RB penetrations with respect to plant grade is a consideration for RB embedment depth.	RB penetration areas should be accessible (inside and outside the RB) for inspection, testing and maintenance of SSCs.	GDC 53 10CFR50.65
36	RB impact on Dose at EAB and EPZ	EAB is defined such that dose limits would not be exceeded in the event of a postulated fission product release as identified in 10CFR50.34(a)(1) as it relates to site evaluation factors identified in 10CFR100	RB design should support NGNP design criterion of EAB and EPZ boundaries at 425 meters from reactor. As an alternative, the EAB and EPZ sizes may be increased (requiring SRM revision).	SRM Fig. 1 10CFR50.34(a)(1) 10CFR100 NUREG-1860 (Table J-5 proposes Draft Example alternative to 10CFR50.34(a)(1)(ii)D
37	Dose – RB design shall support limits on normal operational dose to public	During normal operation, offsite radiation doses to the public shall be < limits specified in Appendix I of 10CFR50 and 40CFR190	See "RB impact on Dose at EAB and EPZ" above.	SRM Table 2 10CFR50 Appendix I 40CFR190
38	Dose – RB design shall support limits on occupational dose	Occupational radiation exposures ≤10% of the limits specified in 10CFR20. PLT 3.1.11.7 - The NGNP shall be designed to demonstrate plant personnel exposure of < 70 person-rem/GWe-year.	See "RB impact on Dose at EAB and EPZ" above.	SRM Table 2 SRM Figure 1
39	Dose – RB design shall support limits on offsite dose at EAB	During DBAs, offsite doses at the site EAB shall be less than those specified in the Manual of Protective Action Guides and Protective Actions for Nuclear Incidents (EPA-520/1-75-001) for sheltering and evacuation	See "RB impact on Dose at EAB and EPZ" above.	10CFR50.34(a)(1)(ii)(d) SRM Table 2 EPA-520/1-75-001 RIS-2005-08 EPA-400-R-92-001 NUREG-0654

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40	Normal operation filtration capabilities	Containment exhaust filter systems are not expected to be required for the VLPC. Primary system cleanup systems are provided. Design basis leakage of primary coolant activity to containment is expected to be limited in amount and contain only limited radioactivity.	RB design should confirm acceptable radiological area controls and gaseous effluent releases without reliance on a normal atmosphere cleanup system.	10CFR20 10CFR50.34a 10CFR50.36a 10CFR50 Appendix I RG 1.140
41	Post-accident filtration capability	Determine need for post-accident filtration capability. See Section 6.2 of this report	Determine appropriate period(s) for atmospheric cleanup following a DBA via dose analysis Specify filtration media Specify filtration system design criteria including seismic classification. Confirm filtration system operation does not adversely affect other SSCs Specify inspection and testing criteria	GDC 41 GDC 42 GDC 43 RG 1.52 SRP 19 (severe accident)
42	RB leakage and ventilation system design:	Design release rate (e.g., one building volume/day) will be a VLPC performance criterion and is therefore likely to require periodic surveillance testing. LWR demonstration of primary containment leakage integrity includes extensive 10CFR50	Confirm appropriate RB leak rate via licensing basis analysis. Design RB to maintain leakage within limit Provide means of	SRM § 4.24 GDC 50 GDC 52 GDC 53 10CFR50App J NUREG-1860

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		Appendix J leak test programs. An alternative approach for NGNP may be to maintain negative pressure with an upper limit on monitored ventilation exhaust flow (i.e., that would communicate with the VLPC), but such a system is more complex than the closed system using chillers described in [GA 2007].	periodically demonstrating leakage is within limits Provide means of monitoring and controlling routine releases.	
43	Reactor cavity concrete temperature	Reactor cavity concrete temperature must be maintained acceptably low to prevent degradation.	The RCCS shall be designed to limit the reactor cavity concrete temperature to a nominal value no greater than 66oC under all normally expected reactor operating conditions. (SRM § 4.7)	GDC 4 SRM § 4.7
44	RCCS structural design – pressure load	RB Subcompartment analyses need to consider the limiting primary coolant pressure boundaries' effect on the RCCS, which is beyond typical scope of subcompartment analyses	The RCCS design basis loading conditions shall include the effect of peak pressure differential that would be produced by a design basis failure of the primary coolant pressure boundary.	GDC 4 SRP 3.8 SRM § 4.7
45	RB design to preclude air ingress event	RB should be designed to prevent consequential loss of building integrity during a loss of primary system integrity, such that convective air flow between the environment and reactor vessel is avoided.	Dynamic effects of primary system piping failures shall be shown to have no adverse effect on RB integrity. Capability to isolate VLPC atmosphere from the outside environs shall be	GDC 4 SRP §3.6 NUREG/CR-6944 v2

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			maintained. Consider remote-manual capability to isolate VLPC e.g., pressure relief dampers.	
46	RCCS - radionuclide control	The RCCS shall be designed to assure that the top-level radionuclide control requirements of PLT 3.1.9 are satisfied.(SRM § 4.7)	The RCCS – VLPC boundary should be designed for zero leakage and shall support RB performance requirements.	SRM § 4.7
47	Safety Margin Basis	EPRI URD endorses development of a safety margin beyond the minimum required by 10CFR, to provide additional assurance of safety, investment protection and severe accident protection.	RB design should define a Safety Margin Basis.	URD Vol 1 § 2.5
48	RAMI	The NGNP shall be designed for high reliability, availability, maintainability, and inspectability (RAMI). Innovative designs to maximize RAMI and minimize human error shall be considered, including techniques for remote maintenance and easy replacement or repair of components	RB design and equipment layout should be analyzed for RAMI with consideration of LWR operating experience, e.g. URD Table 1.8-3 identifies problem areas based on LWR operating experience. Maintainability of equipment should be a consideration in building and equipment layout design, including the environmental conditions in which maintenance will be performed.	10CFR50.65 URD Vol 1§2.9 GDC 18 GDC 21 GDC 32
48a	– lifting and rigging capability	LWR experience has shown that the lack of adequate hoisting provisions is a major source of lost time and difficulties in maintenance.	RB shall provide means for lifting and rigging items greater than 50 pounds.	URD VOL iii ch 6, 2.4.2.5.1
48b	– Robotic capability	Robotics used for maintenance	RB design should	URD Vol 3 Table 1.8-4

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		and inspection activities can improve ALARA.	accommodate robotic activities; URD § 1.8 candidate activities should be considered.	
48c	–snubbers	[GA 2007] describes 2 large hydraulic snubbers for the direct cycle PCS. If snubbers are required for the PCS configuration of choice, they pose a significant challenge to RAMI and ALARA.	Accessibility to snubbers, if applicable, shall be considered in RB and subcompartment design, to enable ISI, maintenance and removal.	
48d	– Primary HTS maintenance		The IHX and the helium circulator shall be removable from the Vessel System as necessary to perform maintenance, repair, or replacement.	SRM § 4.4
48e	–SCS maintenance		The SCS shall consist of a heat exchanger, a motor-driven circulator, and a shut-off valve. This equipment shall be designed as a self contained unit and shall be installed in the bottom head of the reactor vessel. Service equipment shall be designed to remove and replace the shutdown circulator and the shutdown heat exchanger, including the shut-off valve, through the bottom head of the reactor vessel.	SRM § 4.5
48f	–monitoring internal structures	The maintenance rule (10CFR50.65) requires monitoring the effectiveness of maintenance	RB design shall consider provisions for condition monitoring of the building	10CFR50.65 RG 1.160 NUMARC 93-01

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		of structures. includes condition monitoring of internal safety-related structures that may pose a challenge to NGNP due to accessibility issues.	structure and internals.	
48g	– penetrations	Welds in piping that run through penetrations create problems with accessibility for inspection in addition to requiring consideration for break postulation.	Pipe welds in RB penetrations should be avoided.	SRP 3.6 SRP 6.2.4 URD Vol III, 12.8.1
48h	– FHSS		The plant shall be designed to include facilities and features by which the operational readiness of the in-core fuel handling equipment can be assured. This includes periodic inspections, maintenance, testing, and demonstrations of integrated equipment operation. Such inspection, maintenance, testing, and demonstration shall not interfere with core refueling operations nor shall they cause any adverse effects on plant operation.	SRM § 4.9
48i	-Steam Generator	SG RAMI considerations should be factored into the RB design	-Steam generator accessibility for ISI and maintenance, including tube repair, is required. -SG platform design is dictated by manway and	

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			nozzle/piping location. Temporary scaffolding may be required, e.g., for piping. -Davit arms are often required to support opening and closing manway covers, with space required to swing the manway cover away from the SG using the davit arm. -Adequate clearances are required for SG removal and installation, with consideration of appurtenances and rigging.	
49	Constructability – design completion	Lessons learned form past nuclear construction highlights the need to complete design well in advance of starting construction	(NGNP general goal) – 90% of design drawings must be 100% complete prior to start of nuclear construction.	URD Vol 1 § 2.9
50	Constructability - Modular	Section 5.6 of [HTGR-88512] identifies criteria for modularization of NGNP subassemblies, and notes that preassembly of reinforcing with structural elements was not developed at the time (1990). RG 1.142 currently includes a position (C.13) on modular construction of safety-related concrete structures (i.e., the same rules used in computing the strength of regular reinforced concrete should apply to composite members used in modular construction).	RB design should consider modularization of subassemblies using criteria of [HTGR-88512] with insights from current guidance and operating experience as it develops.	SRM Figure 1 HTGR-88512 RG 1.142 ACI 349-97 URD Vol 1 § 2.9
51	Constructability - schedule	36 month schedule is defined as	RB design to support 36	SRM Figure 1

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		top level NGNP requirement in [SRM, 2007]	month schedule form start of site work to commercial operation.	
52	Fire Protection - suppression capability	Operation of the fire protection systems should not compromise the integrity of the RB or other systems important to safety. Fire protection activities in the RB areas should function in conjunction with overall design requirements such as ventilation and control of contaminated liquid and gaseous release.	RB fire suppression system capability should be defined based on fire hazards analysis (FHA), with consideration of impact to total building function.	GDC 3 RG 1.189
53	Fire Protection- detection capability	Fire detection capability in the RB should be capable of detecting identified fire hazards.	Fire detector type and location should be specified to address each fire hazard identified in the FHA. General fire detection capability should also be provided as a backup (e.g., smoke or heat detection).	GDC 3 RG 1.189
54	Security – mitigation and response to potential or actual aircraft hazards	NRC rulemaking activities and current practice call for including security considerations in plant design.	RB design shall consider onsite actions necessary to enhance the capability of the NGNP to mitigate the consequence of an aircraft impact. Consider measures to reduce visual discrimination of the site relative to its surroundings. Consider guidance and strategies intended to maintain or restore core cooling, containment and spent fuel cooling following	10CFR50.54(hh) (as proposed in SECY-2008-0099)

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			losses of large areas of the plant due to explosion. Include I) fire fighting; ii) operations to minimize fuel damage; iii) actions to minimize radiological release.	
55	Security	Inherent sabotage resistance was achieved for ALWRs at little or no additional cost by developing ALWR plant layouts which take advantage of other design requirements (e.g., hardening for missiles, separation of safety system divisions, backup capabilities to accomplish safety functions, etc.).	Provisions for site security shall be provided for the protection of the reactor, reactor fuel, spent fuel, electrical power, and hydrogen. Plant security shall be a consideration in developing plant equipment layouts to protect against design basis threats.	10CFR73.1 10CFR73.2 10CFR73.55 SRM Table 3 SRM Figure 1 URD Vol 1, 2.13
56	Security – vital equipment access	Access to vital equipment shall require passage through at least two physical barriers (i.e., PA and VA boundary).	Access to the RB shall be designed and controlled such that it constitutes a vital area boundary and physical barrier as defined in 10CFR73.2	10CFR73.2 URD Vol 3 Ch 9
57	Passive safety	RB design features such as RV support, and RCCS interface need to be designed to support top level NGNP passive safety criteria. Passive reactor designs are not required to meet the requirements of GDCs 33, 34, 35, 38, 41, and 44 for 72 hours .(SRP 8.2).	RB design shall support passive safety functions. Passive safety systems should have sufficient capability to maintain safe shutdown conditions for 72 hours after design basis events, without operator actions, following a loss of both onsite and offsite power sources. Reliance on safety-related DC batteries	SRM PLT 3.1.3 10CFR50.63 SRP 6.3 SRP 8.2 SRP 8.3 URD Vol 1, 3.1.3

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			and their inverters is acceptable	
58	RB normal operating temperature and pressure limits – safety analysis input	Maximum normal operating containment temperature and pressure establish initial conditions for safety analyses.	Specify limiting maximum normal operating temperature and pressure of containment and provide capability for monitoring building temperature during normal operations.	GDC 4 GDC 50 SRP 6.2
59	Environmental Qualification conditions, e.g. RB temperature profiles and total integrated dose	The equipment required to be in an EQ program needs to be defined and evaluated	RB environmental conditions for normal operation, including anticipated operational occurrences, design basis accidents and , external events shall be developed as input to an EQ program	10CFR50.49 SRP 3.11
60	RB – structural integrity test capability	Containment buildings require structural integrity testing at pressures above design pressure.	RB shall include provisions for performance of a structural integrity test.	GDC 16 GDC 50 10CFR50 App J
61	RB – leakage testing		RB shall be designed to include provisions for verifying leakage within design limits	10CFR50 App J ASME III
62	60-year design life – RB integrity		RB design should incorporate features to support a 60 year design life, including provisions for inspection and condition monitoring.	
63	60-year design life – accessibility for maintenance of SSCs		RB design should consider life cycle maintenance and replacement needs of SSCs to support 60 year design	

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			life.	
64	PRA	License application shall be supported by a full-scope probabilistic risk assessment analysis for internal and external events. Probability of exposure exceeding the Protective Action Guides at or beyond the EAB shall be < 5 x 10-7 per plant year.	RB and interfacing systems' should be modeled in a full scope PRA of internal and external events.	SRM Table 2 SRM Figure 1 RG 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities." Rev 2 will endorse the level 1and large early release frequency PRA Standard (ASME/ANS RA-S-2007) NUREG-1860
65	Human Factors	Human Factors Engineering includes consideration of risk-important actions outside the control room.	Risk important actions in the RB, if any, should be identified and addressed as part of the Human Factors Engineering (HFE program)	10CFR50.34(f) SRP § 18
66	Minimized Contamination	Design and operational features must consider minimization of contamination and radwaste for the plant life cycle	RB design and procedures for operation should minimize, contamination of the facility and the environment, facilitate eventual decommissioning, and minimize, to the extent practicable, the generation of radioactive waste.	10CFR20.1406 RG 4.21[2]

[2] HTS loop alternatives study [GA 2008b] contains specific recommendations based on draft RG 4.21, Minimization of Contamination and Radioactive Waste Generation – Life Cycle Planning, which has since been issued in June, 2008. The recommendations in [GA 2008b] remain applicable.

Item #	RB Design Feature	Issue	Impact to T&FR	References
67	Safeguards		Measures shall be incorporated as necessary to prevent unauthorized access to nuclear material, theft, diversion, other malevolent acts, including sabotage intended to release radioactivity or disrupt operations.	SRM Table 3
68	Pipe routing and embedment depth	Pipe routing from SG to RB penetration and from IHX to H2 production is affected by RB embedment depth and relative location of components.	RB embedment effects on pipe routing, penetration elevation and pipe tunnel design and inspection should be addressed in the design.	See Section 3
69	Refueling capability – RB design supports 30 day (or less) refueling outages.	URD top-level requirement for refueling outage free from major problems to be conducted in 17 days or less (breaker to breaker) assuming 24-hour productive days, is more restrictive than 30-day SRM requirement..	RB design shall optimize refueling outage performance, e.g. via fuel handling and storage, ALARA, equipment accessibility, laydown provisions. – provisions for core service tools – RB supports fuel handing machine – RB supports auxiliary service cask used to remotely remove and install neutron control assemblies - accessibility to permanent power and service air should be provided in the RB at	SRM Figure 1 URD Vol 1, 3.2.3

Item #	RB Design Feature	Issue	Impact to T&FR	References
			critical locations for outage activities, e.g., SG cavity	
70	Special test provisions	The NGNP design shall include sufficient flexibility to allow for the future investigation of safety and operational performance margins in the plant responses to anticipated operating occurrences and risk-important events, as advancements and modifications for future designs are considered.	RB design for the NGNP shall consider access to SSCs and other provisions to accommodate special testing provisions (e.g., installation of temporary measurement and test equipment).	SRM PLT 3.5.1
71	Special test provisions	SRM requirement for experimental determination of fission product release during a primary coolant boundary rupture.	For demonstration of commercial plant radiological source terms, the NGNP shall be designed to experimentally determine the fission product activity that could potentially be released should there be a rupture in the primary coolant boundary.	SRM PLT 3.5.8
72	Construction cranes and foundation for heavy loads –	Construction crane(s) will likely be required to support major equipment maintenance and removal after initial startup. For a deeply embedded RB, vertical lifts through the RB roof are likely the only practical means of moving large components from the RB. Large component lifts would exceed the RB crane capacity and would require large construction cranes.	The RB and surrounding foundation should be designed to accommodate heavy load lifts.	SRM PLT 3.0.8 SRM PLT 3.0.9 SRP 9.1.5 NUREG-0612 [RIS 2005-021]
73	Safe Load path definition	Lifts of heavy loads through the	Safe load paths should be	SRM PLT 3.0.8

Item #	RB Design Feature	Issue	Impact to T&FR	References
	for heavy loads	RB and transport in proximity to the RB need to address the risk to nuclear safety	defined for heavy load transport and lifts.	SRM PLT 3.0.9 SRP 9.1.5 NUREG-0612 [RIS 2005-021]
74	Fuel handling and storage accident	The referenced regulations and guidance require evaluation of fuel handling accident preventive measures and radiological consequences. [PSID, 1992] identifies no AOOs or DBEs for fuel handling and storage.	RB design shall consider measures to prevent and limit the consequences of postulated fuel handling accidents.	GDC 61 GDC 62 SRP 9.1.5 SRP 15.0 SRP 15.7.4 NUREG-0612
75	Radwaste AOOs and DBEs	The referenced regulations and guidance require consideration of AOOs and DBEs related to radioactive waste. [PSID, 1992] identifies no AOOs or DBEs for radwaste-related events.	Radwaste system-related AOOs and DBEs shall be defined for NGNP and considered in the design of the RB and interfacing systems.	GDC 60 GDC 61 SRP 11.2 SRP 11.3 SRP 11.4
76	Inspections, tests, Analyses and Acceptance Criteria (ITAAC)	Under the 10CFR52 combined license process, ITAAC are required to provide reasonable assurance that the plant, including the RB, is constructed as designed and will operate in accordance with the design certification.	ITAAC should be developed for the RB using the guidance of SRP 14.3 and RG 1.206.	10CFR52.47(b)(1) SRP 14.3.11 RG 1.206 C.II.1

3. IMPACTS OF RB EMBEDMENT DEPTH ON NGNP DESIGN AND CONSTRUCTION

The effects of embedment depth on design of the NGNP at the INL site are presented herein by comparing design, construction, functional and licensing considerations for three alternative embedment concepts as depicted below:

Alternative 1: Cylindrical portion of RB embedded below plant ground surface.

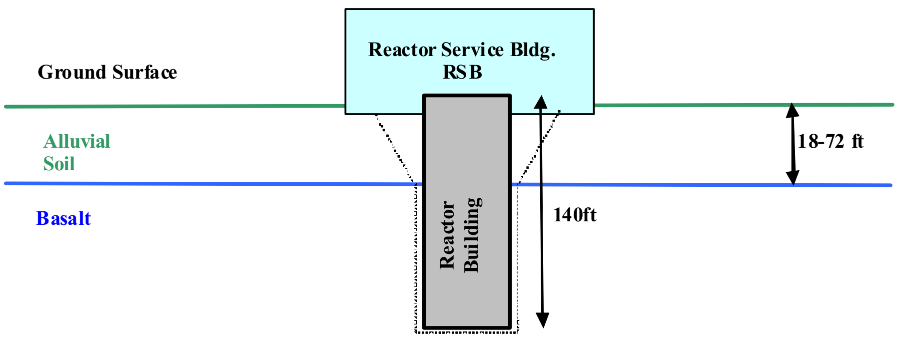

Figure 3-1. Fully Embedded RB

Alternative 2: Cylindrical portion of RB partially embedded below plant ground surface. This alternative would require functional rearrangements in the rectangular portion of the RB.

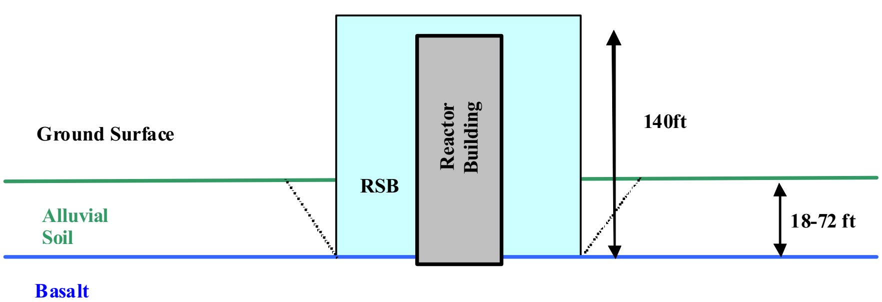

Figure 3-2. Partially Embedded RB

Alternative 3: Cylindrical portion of RB partially embedded below plant ground surface and partially backfilled.

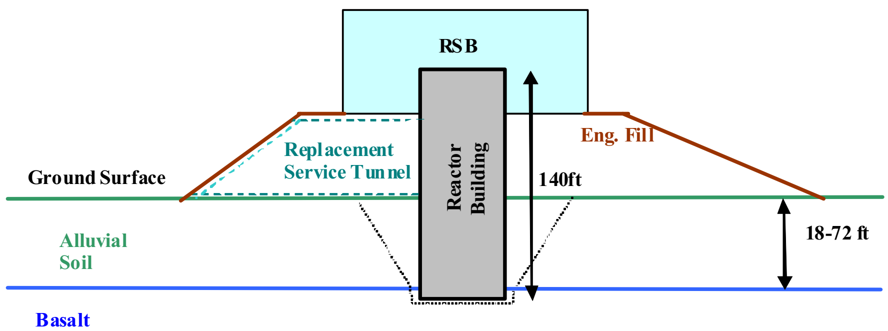

Figure 3-3. Partially Embedded RB with Backfill

3.1 Design Considerations

A greater embedment depth below ground surface will reduce the seismic response of the building. This can have an important effect on the magnitude of seismic loads used for design of equipment (mechanical, electrical and commodities such as piping, conduit, cable tray and HVAC ductwork) and their supports as reflected by amplified peaks in the in-structure response spectra (ISRS). For the portion of the structure embedded below ground surface, the 5% damping ISRS, for most of the frequencies, most probably will be below the Performance Category PC 4 (10,000 year) spectra) developed for the INL site as presented in Fig. 3-4. For the portion of the structure above grade, the 5% damping ISRS most probably will be above the PC 4 spectra in Fig. 3-4 for most frequencies. Therefore, with respect to seismic design of equipment and their supports, Alternative 1 is the most favorable. The seismic response at the top of the engineered fill in Alternative 3 will be higher than the response at the ground surface at site grade, which makes this alternative the least favorable from a seismic design perspective.

Deeper embedment reduces self-inertia seismic loads on building structural components. For the portion of the structure embedded below ground surface at site grade, the horizontal seismic coefficient (S_a) is expected to be lower than the PC 4 Peak Ground Acceleration (PGA) of 0.363 g. For the portion of the structure above grade, S_a is expected to be amplified to as much as 1.15 g. With respect to this design consideration, Alternative 1 is the most favorable and Alternative 3 is the least favorable.

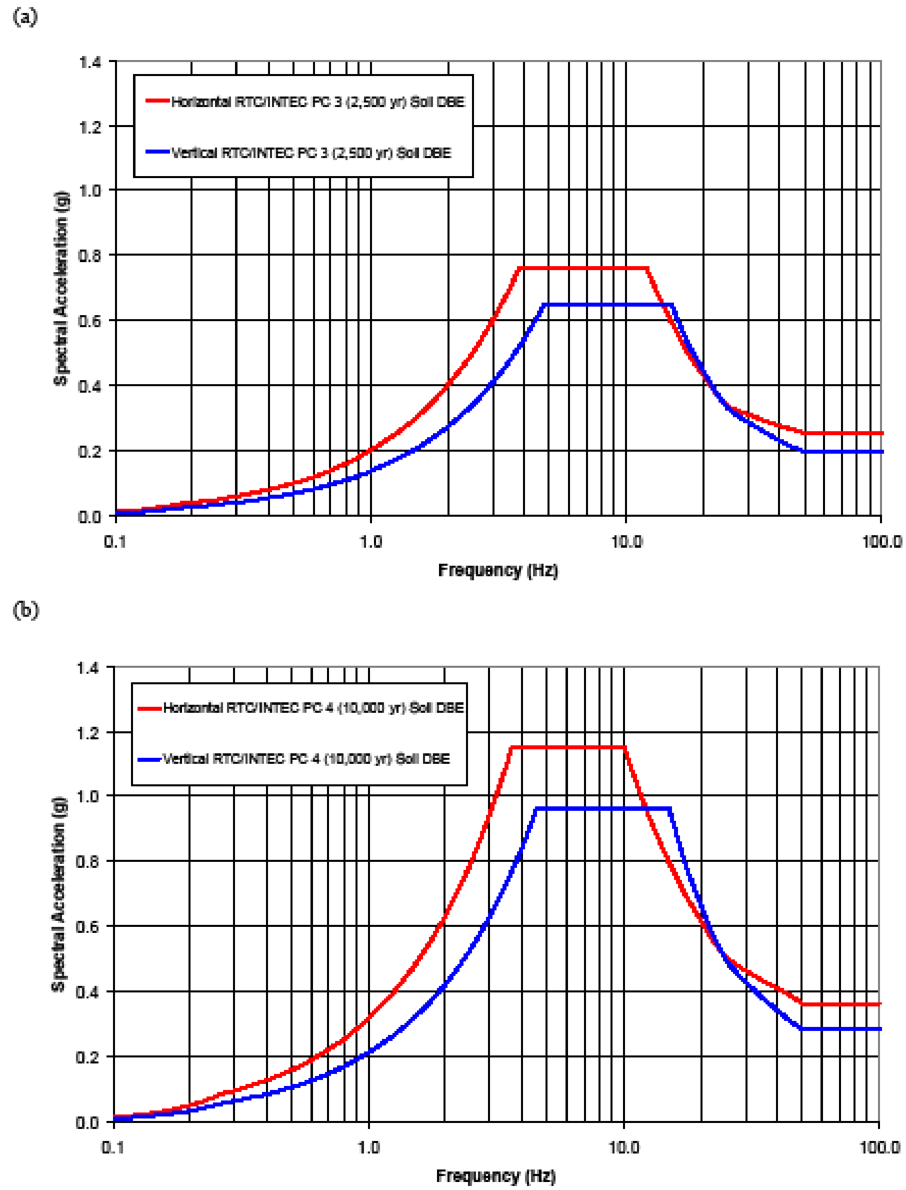

Figure 46. The horizontal and vertical RTC/INTEC Soil DBE 5% damped spectra for (a) PC 3 (2,500 yr) and (b) PC 4 (10,000 yr).

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Figure 3-4. INL Earthquake Spectra (Figure 46 from [Payne 2006])

Soil surrounding the building exerts soil pressures on the exterior walls of the embedded portion of the RB. The magnitude of the soil pressure loads is proportional to soil unit weight γ, soil coefficient at rest (K_O) and the square of the depth of embedment. Balanced soil pressures will result in compressive hoop stresses on the circular perimeter, reinforced concrete wall cross section that encases the RB. Because of the inherent capacity of the circular encasement configuration of the RB, this is expected to have less significant impact on the reinforcement design for Alternatives 1 and 3. For Alternative 2, at-rest soil pressure will have some effect on the design of the reinforcement of the basement walls of the RB.

Ground water pressures act on the portion of the building below the elevation of the water table. The magnitude of ground water pressure load is proportional to the square of the depth of building below the water table elevation. Ground water pressure results in compressive hoop stresses on the circular perimeter of the reinforced concrete wall cross section that encases the RB. Because of the inherent capacity of the circular encasement configuration of the RB, this is again expected to have less significant impact on the reinforcement design for Alternatives 1 and 3. For Alternative 2, the water pressures will have some effect on design of the basement walls reinforcement structures. The design of the reactor well in Alternative 1 would have to address measures to prevent water leakage into the RB, e.g., by a dewatering system or other sealing design.

Soil surrounding the building also exerts seismic dynamic soil pressures on the RB. The magnitude of dynamic soil load is proportional to soil unit weight γ, earthquake magnitude, footprint dimensions, and square of the depth of embedment. (Refer to Fig. 3.5-2 of [ASCE-4-98]). The seismic soil pressures will affect the design of the underground exterior walls for all three alternatives. With respect to seismic earth pressures, Alternative 2 is the favorable solution.

Deeper embedment reduces wind loads, tornado loads, and blast loads on the building. These loads are a function of height of the building above grade and foot print dimensions of the building. The magnitude of wind and tornado loads is comparable with the magnitude of dynamic soil pressure loads (Eq. 3 in [ASCE 7-88]).

It is proposed to locate the nuclear plant and hydrogen production facility as close to each other as possible (within 100 m or less) in order to minimize the distance over which high-temperature heat is transferred. INL has performed an engineering evaluation for these separation requirements and has concluded separation distances in the range of 60 m to 120 m should be adequate in terms of safety [INL 2005]. Other recommendations from the INL study include a 100 kg on-site limit for hydrogen storage, use of double-walled pipes for hydrogen transport, and location of the nuclear plant control room outside of the dispersion zone for chemical release. The below-grade installation of the MHR module, combined with an earthen berm for defense-in-depth, provide additional safety margin for co-location of the two facilities. Detailed safety assessments should be performed in follow-on design phases to better define the risk envelope associated with co-location of the nuclear plant and hydrogen production facility.

Japan Atomic Energy Agency (JAEA) has performed computational fluid dynamics simulations of transport and detonation of a hydrogen cloud resulting from an accident in the hydrogen production plant. JAEA has also concluded that relatively short separation distances between the nuclear reactor and hydrogen production plant should not compromise overall plant safety, especially if an earthen berm or other barrier is placed between the nuclear reactor and hydrogen production plant [Nishihara 2005].

Definition of design basis threats, including commercial aircraft strike scenarios, are defined using safeguards information. Qualitatively, the fully embedded RB offers greater protection from man-related hazards including aircraft strikes. The effect of such hazards on the exposed (above ground) portion of the building should be considered for potential adverse interactions on the safety significant equipment located below grade, e.g., effects of overhead crane failure as a consequence of an impact to the RB. Alternative 2 is the most adversely affected by these external loads.

Deeper embedment poses challenges in pipe tunnel design and pipe routing, particularly for main steam and feedwater lines. It is preferable to have the SG steam outlet and turbine inlet lines at relatively close elevations, to allow gradually sloping steam lines. A pipe tunnel would assure protection and accessibility of the penetration area and piping outside the RB. The depth of a pipe tunnel is dictated by the elevation of the penetrations. Greater embedment of the pipe tunnel increases the overburden pressures. The elevations of the SG steam and feedwater nozzles are affected by embedment depth and several primary HTS and vessel system design constraints, including:

- The horizontal cross vessel fixes the relative elevation of the RV nozzles equal to the SG nozzles
- The thermal center of the RV is higher than the thermal center of the SG

The heights of the SG, and to a lesser extent the RV, determine the overall building depth. Assuming the steam turbine is located at approximately grade elevation, Alternative 1 is the least favorable option and Alternative 2 is the most favorable option for pipe routing.

<u>Waterproofing Considerations for Embedded RB</u>

Design measures to protect the embedded portion of the RB against external flooding may include providing a water barrier on all exterior concrete members subjected to ground water via a combination of installing a waterstop and applying waterproofing chemicals to the concrete.

The waterstop could be constructed from polyvinyl chloride (PVC) and would be installed at all below grade cold joints in the perimeter walls and slabs exposed to outside environment. The waterstop would be designed to withstand the maximum hydrostatic pressure on the building, e.g., a waterstop designed to withstand 150 feet of head should be sufficient for a RB buried 140 feet in the ground. Stainless steel waterstops can resist higher hydrostatic pressures, but at greater cost compared to PVC waterstops.

The concrete waterproofing compound may be applied as an admixture at the time of batching (e.g., for relatively thin concrete members) or as a membrane waterproofing in the form of a cementitious coat (typically used for thicker members). The waterproofing chemical is a dry powder compound consisting of portland cement, very fine silica sand and various active proprietary chemicals. The chemical compound has an affinity with water and when mixed with water, by the process of diffusion, a nonsoluble crystalline formation of dendritic fibers develops. The fibers penetrate the pores and capillary tracts of concrete thereby sealing the concrete against the infiltration of water.

A fully embedded RB may warrant additional protective measures to avoid the risk of leakage and poses a potential challenge for condition monitoring of the foundation, as described below.

3.2 Construction Considerations

The depth of embedment will increase the construction costs for excavation, excavation dewatering, backfilling, and transportation and disposal of excavated material. The costs of excavating material are a function of both the depth of the excavation and the elevation of bedrock. Blasting of rock can be 5-10 times more expensive than the excavation of alluvial soil. This has a major impact on the cost of constructing the RB in Alternative 1 below the bedrock elevation. The costs associated with the transportation, disposal and compaction of excessive excavated material have an effect on Alternatives 1 and 2.

The required amount of the engineered fill, its location, and material properties have impacts on the cost for excavation, transportation, and placing and compaction of fill material. These construction costs are most significant for Alternative 2 which has greater engineered fill. It is anticipated that the excavated alluvial soil can be used for backfilling.

The portion of the RB below the ground level will be affected by elevation of the ground water table that can increase the costs for dewatering and shoring of the excavation. The cost for excavation dewatering can be significant for Alternative 1 if the depth of the excavation at a given site is significantly below the water table.

Expansion joints have to be constructed if needed to accommodate excessive differential settlements. The different subgrade conditions below the cylindrical and rectangular portions of the RB for Alternatives 1 and 3 would require construction measures to accommodate the excessive differential settlements and seismic displacements.

The Reactor Cavity Cooling System (RCCS) utilizes a natural draft vent that requires a height of approximately 95 ft above the top of the reactor to provide sufficient natural convection airflow

[GA 2007]. The embedded RB would facilitate construction of the RCCS vent stack by allowing a lower height with respect to grade elevation compared to a partially embedded RB.

A vent stack may be installed to improve atmospheric dispersion of radiological releases, and its height may be governed by the height of the RB. Per RG 1.194, in order to take credit for the additional dispersion provided by an elevated release, the stack height is required to be 2.5 times the height (above grade) of adjacent structures. A fully embedded RB would reduce the above-grade height of adjacent structures and hence would reduce the required stack height. Vent stack design options are discussed in Section 6.3.

If a deeply embedded pipe tunnel is dictated by the RB embedment and equipment layout as described above, then the potentially significant excavation, dewatering and construction costs associated with the pipe tunnel design should be factored into the final selection of RB embedment depth.

3.3 Functional Considerations

The design solution for the depth of excavation affects the construction and operation costs related to replacement of major (heavy and bulky) mechanical components. Alternative 1, where the whole height of the reactor cavity is embedded below the ground surface, requires installation of major components during initial construction (and possibly later if equipment replacement is required) through the roof of the building by a heavy crane temporally erected outside the building. Consideration should be given to including a crane foundation as part of the initial design effort to allow for the possibility of future equipment replacement. Alternatives 2 and 3, where part of the RB is above the ground elevation, could allow mechanical components in the RB to be tilted and taken out in horizontal positions through the walls of the building provided the diameter of the building is increased as necessary to accommodate the tilting. Alternative 3 would also require a service tunnel to be constructed for equipment replacement through the building wall. Another alternative for major equipment replacement is to increase the height of the rectangular portion of the RB to accommodate tilting of the equipment for horizontal removal through the RB walls.

The soil around the RB in Alternatives 1 and 3 provides additional protections from design basis threats and other external hazards (such as explosions in the hydrogen production plant). The soil around the RB also allows for conduction cooling from the RB that may serve to back up the RCCS ultimate heat sink function for beyond design basis events.

3.4 Licensing Considerations

<u>Adequacy of Supporting Media and Soil Structure Interaction (SSI)</u>

Adequacy of Supporting Media and Soil Structure Interaction (SSI) NRC acceptance of Seismic Category I structures, such as the NGNP RB, is based in part on the adequacy of the structures' supporting media, with consideration of foundation embedment depth, depth of soil over bedrock and soil layering characteristics (SRP § 3.7.1). As given in SRP § 3.7.2, specific areas of uncertainty that must be accounted for in modeling SSI include:

A. The random nature of the soil and rock configuration and material characteristics.

B. Uncertainty in soil constitutive modeling (soil stiffness, damping, etc.).

C. Nonlinear soil behavior.

D. Coupling between the structures and soil.

E. Lack of uniformity in the soil profile, which is usually assumed to be uniformly layered in all horizontal directions.

F. Effects of the flexibility of soil/rock.

G. Effects of the flexibility of basemat.

H. The effect of pore water on structural responses, including the effects of variability of ground-water level with time.

I. Effects of partial separation or loss of contact between the structure (embedded portion of the structure and foundation mat) and the soil during the earthquake.

A fully embedded RB potentially adds complexity to soil-structure interaction (SSI) considerations (Items A, B, C E and H). Licensability of the RB design with respect to these aspects is a function of the adequacy of the site geotechnical characteristics. Appendix B provides a summary of INL site-specific geotechnical considerations.

<u>Post-Construction and Inservice Surveillance of Foundation</u>

Category I building foundations should include provisions to accommodate inspection of critical areas. NRC determines the acceptability of post-construction and inservice surveillance programs for foundations on a case-by-case basis [SRP § 3.8.5], with consideration of factors such as:

- Periodic condition monitoring of inaccessible areas
- Ground water chemistry
- Monitoring of settlements and differential displacements

A fully embedded RB poses potential challenges to inspection and condition monitoring of the foundation that require further evaluation.

Section 3.8.6 of [PSID 1992] states:

“There are no requirements for post-construction testing or inspection of any Standard MHTGR structure.”

The viability of this approach for NGNP requires further investigation in light of the RB equipment protection and radionuclide control functions and the expected applicability of GDC 1 to the RB design, as discussed in section 2.5.

Penetrations and Piping

Compliance with GDC 53 requires that the reactor containment be designed to permit (a) periodic inspection of penetrations, (b) an appropriate surveillance program, and (c) periodic testing of the leak tightness of penetrations with resilient seals and expansion bellows. [SRP 6.2.6]. An alternative to GDC 53 for advanced reactors is suggested in Draft Example Requirement 16 in Appendix J to NUREG-1860. The alternative to GDC 53 would require radiological containment performance criteria to be met and periodically tested. A fully embedded RB coupled with the NGNP vessel and piping system layout would require mechanical penetrations at low elevations, particularly for main steam piping. The ability to inspect and test the penetrations and piping would be subject to licensing review to assure verification of containment performance criteria.

Protective features for penetrations located below flood level are verified via Inspections, Tests, Analyses and Acceptance Criteria for Combined License applicants. [SRP § 14.3]

4. RADIONUCLIDE SOURCE TERM SUMMARY

4.1 Dominant Events for Offsite Consequences and Risk

A spectrum of possible licensing basis events (LBEs) for use in determining the site suitability source terms were identified for the 350 MWt steam-cycle MHTGR through the application of a rigorous and structured analytical process as described in the MHTGR Preliminary Safety Information Document (PSID) [PSID 1992], the Probabilistic Risk Assessment for the Standard Modular High Temperature Gas-Cooled Reactor [PRA 1988], and LBEs for the Modular HTGR [GA 1987]. It is expected that the same set of events will also govern the evaluation of offsite radionuclide releases for a 600 MWt steam-cycle plant.

LBEs cover the full spectrum of events from anticipated operational occurrences to beyond design basis events with frequencies as low as 5×10^{-7} per plant year. Design Basis Events (DBEs) are not expected to occur within the plant lifetime, but may occur in the lifetime of a population of plants for which the plant is designed and conservatively assessed against 10CFR100. Safety Related Design Conditions (SRDCs) are derived from the DBEs by assuming that only safety related equipment is available to mitigate the consequences. As discussed in [Dilling 1993], two SRDCs have been identified as the risk dominant events for the steam-cycle MHTGR. These two events are summarized below and described in more detail in [PSID 1992] and [GA 1991].

The overall VHTR source term is a function of as-manufactured fuel quality, fuel performance during normal operation, the extent of fuel heatup during loss of forced cooling, and the extent of chemical attack during water or air ingress events. There are generally two distinct components to the VHTR source term: (1) a prompt source term which can be released immediately and (2) a delayed source term whose timing is determined by the slow heatup of the core. The design of the RB, particularly in terms of design pressure and design leak rate, is determined in part by the characteristics of the prompt and delayed source terms, which are discussed in more detail below.

The prompt source term is comprised of both circulating and plateout radioactivity. Circulating radioactivity is comprised of mostly noble gases that can be released during a primary coolant depressurization event. Plateout radioactivity is comprised of mostly condensable radionuclides (e.g., I-131) that plateout on the cooler wetted surfaces of the primary coolant circuit. Plateout activity can be released as the result of surface shear forces during a rapid depressurization event (large break of the primary coolant pressure boundary) and by wash-off or steam-induced vaporization during a water ingress event that causes the pressure relief valves to actuate. The time scale for the prompt source term ranges from seconds to minutes.

For the VHTR, the delayed source term is typically much larger than the prompt source term. The combination of graphite with high heat capacity and a core with low power density results in limited fuel temperature transients that occur slowly over time periods of several days during loss of flow or loss of coolant accidents. The delayed source term develops over the course of the heatup portion of the transient and consists primarily of radioactivity released from heavy metal contamination, defective fuel particles that fail during normal operation, and the very small fraction of non-defective particles that fail during normal operation and during the heatup.[3] During a water-ingress event, hydrolysis of the exposed heavy metal increases the release rate of radioactivity, but the hydrolysis reaction typically occurs over several hours. The time constant for the delayed source term ranges from several hours to days.

As discussed in [Dilling 1993], the delayed source term dominates the radiological source term and offsite doses for the more severe accidents. For these types of accidents, the radiological consequences are reduced by allowing the RB to vent at a low differential pressure, resulting in a slow, low-concentration release from the RB. If the RB is designed to vent at a higher differential pressure, the delayed source term will build up within the RB until the relief setpoint is exceeded, resulting in a more rapid, higher-concentration release, which typically results in higher offsite doses.

4.1.1 Small Helium Leak with LPCC (SRDC-11)

This event was referred to as SRDC-11 for previous steam-cycle MHTGR concepts. This event is initiated by failure of one of the small instrument or service system lines that penetrate the reactor pressure vessel, resulting in a slow depressurization. The reactor trips automatically on low primary coolant pressure and only the safety-related RCCS is assumed to be available to remove decay heat. The reactor then undergoes a Low Pressure Conduction Cooldown (LPCC). Figure 4-1 shows the peak and average fuel temperature responses during this event for the 450 MWt MHTGR. In terms of impact on the source term, the significance of the slow depressurization event is that the helium coolant is still exhausting from the RPV while core temperatures are rising. During the heatup period, radionuclides are released from exposed heavy metal and the flow of helium transports some of this radioactivity into the RB and increases the source term available for release to the environment. For a larger break with rapid depressurization (e.g., failure of a primary coolant pressure relief line), there is no significant outflow of helium from the RPV during the core heatup period. The rapid depressurization events result in a greater release of plateout radioactivity because of the higher shear forces on the wetted surfaces, but previous safety assessments have shown the slow depressurization events result in a greater overall release of radioactivity to the RB and environment.[4] Figure 4-2 shows the predicted release of I-131 from the core, vessel, and RB during this event for the 450 MWt MHTGR.

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[3] In the highest temperature regions of the core, some radioactivity (typically noble metals, e.g., Ag-110m) can be released by diffusion through intact coatings.

[4] This result applies to prismatic block concepts with very low levels of radioactivity associated with circulating dust. For pebble bed concepts, the concentrations of dust (and circulating activity) in the primary circuit are expected to be significantly higher, and the prompt source term from a rapid depressurization may have more of an impact on containment design considerations.

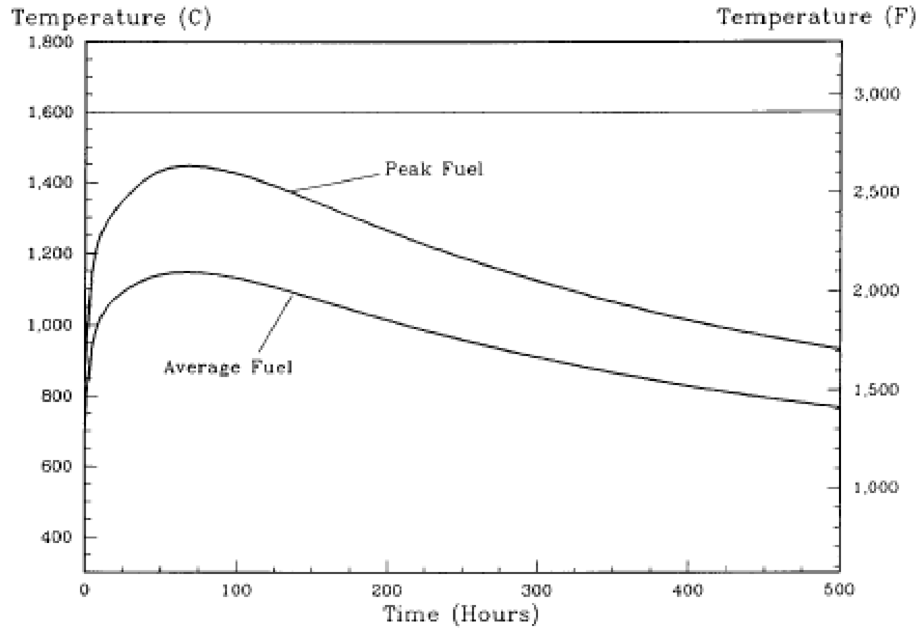


Figure 4-1. Fuel Temperature Response during SRDC-11 for 450 MWt MHTGR

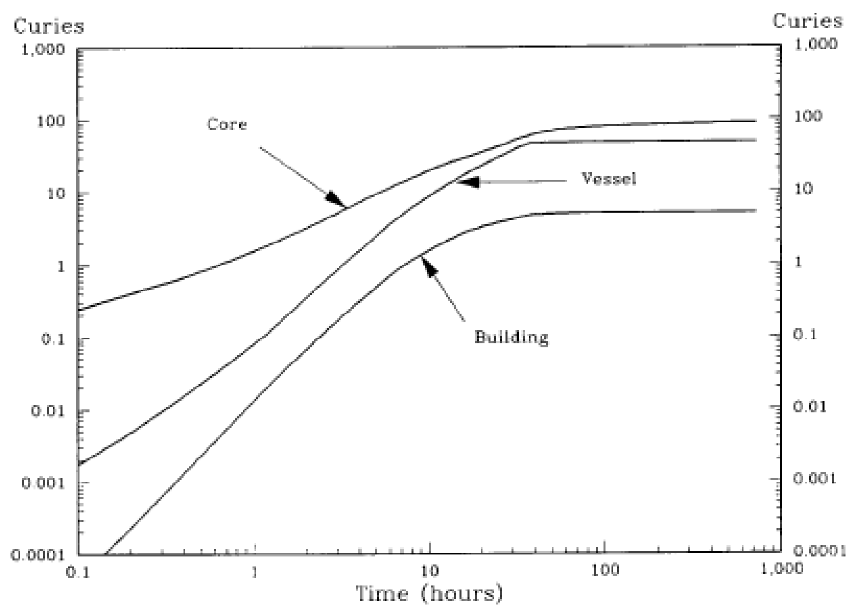


Figure 4-2. Cumulative I-131 Release during SRDC-11 for 450 MWt MHTGR

4.1.2 Steam Generator Tube Failure with Delayed LPCC (SRDC-6)

This event was referred to as SRDC-6 for previous steam-cycle MHTGR concepts. This event is initiated with an offset rupture of a steam generator tube and results in moderate water ingress into the primary loop. The steam mixes with the helium, which causes a significant increase in the primary coolant moisture concentration. Both the moisture monitors and the neutron flux controller are non-safety related equipment and are assumed to have failed. The moisture ingress causes an increase in core reactivity, which causes the reactor trip setpoint on high core power-to-flow ratio of 1.5 to be exceeded within a few seconds. Following reactor trip, the feedwater pumps are ramped down to 15% of total flow, causing a similar reduction in primary coolant flow. Continued moisture ingress causes the high primary coolant pressure setpoint to be exceeded, which results in insertion of reserve shutdown material, shutdown of the main circulator, and isolation of the steam generator.[5] However, the steam generator is not dumped since the dump system is not safety related. The shutdown cooling circulator is also non-safety related and fails to start. These events initiate a high pressure conduction cooldown (HPCC) to the RCCS.

The steam reacts endothermically with graphite to produce hydrogen and carbon monoxide.[6] Steam also reacts with the heavy metal in the small fraction of fuel particles with failed coatings and the small fraction of heavy metal in the form of contamination outside intact fuel particles. In terms of impact on the source term, hydrolysis of the exposed heavy metal is a key contributor to radionuclide release. For this event, the internal pressure within the reactor pressure vessel increases because of the steam ingress, hydrogen and carbon monoxide generation, and increasing temperatures. As shown in Fig. 4-1, an analysis of this event for the 450 MWt MHTGR [GA 1991] showed the pressure relief valve cycling 3 times during the course of the accident, after which cooling and diminished moisture ingress prevented additional openings of the relief valve (see Fig. 4-3). To assess the maximum potential for radionuclide release for this accident, the relief valve was assumed to fail open after the third cycle, approximately 21 hours after initiation of the accident. During this time period, a significant fraction of the exposed heavy metal is predicted to undergo hydrolysis (see Fig. 4-4). After the final pressure relief, the system depressurizes in about 13 minutes, after which the reactor undergoes a LPCC, with fuel temperature response similar to that shown in Fig. 4-1. Figure 4-5 shows the predicted release of I-131 from the core, vessel, and RB during this event for the 450 MWt MHTGR.

[5] Both the insertion of control rods and reserve shutdown material occur by gravity after a trip signal from the Reactor Protection System.

[6] Previous safety assessments for this event [GA 1991] have accounted for generation of hydrogen and carbon monoxide using available data and correlations for oxidation of H-451 graphite by steam [GA 1984]. The oxidation rates are relatively slow, and the concentrations of hydrogen and carbon monoxide in the VLPC following activation of the primary coolant pressure relief valves are well below established flammability limits [PRA 1988]. However, the potential impacts of flammable gas generation should be re-evaluated for an NGNP operating with higher core outlet temperatures in the range 900°C to 950°C.

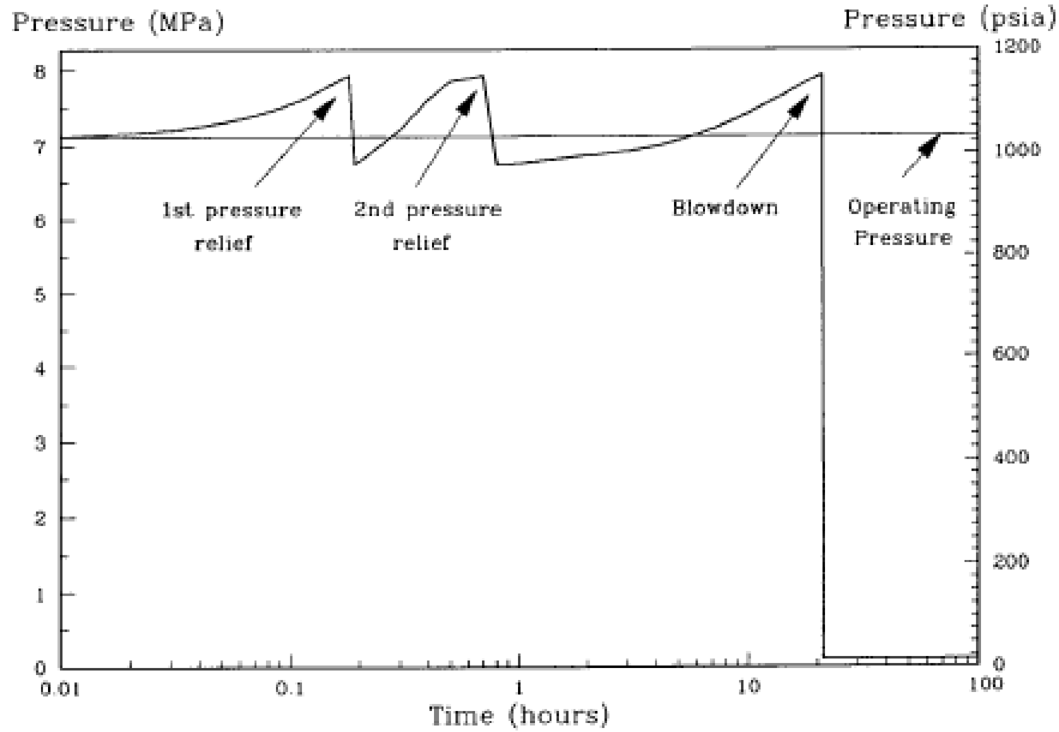


Figure 4-3. Primary Coolant Pressure Response during SRDC-6 for 450 MWt MHTGR

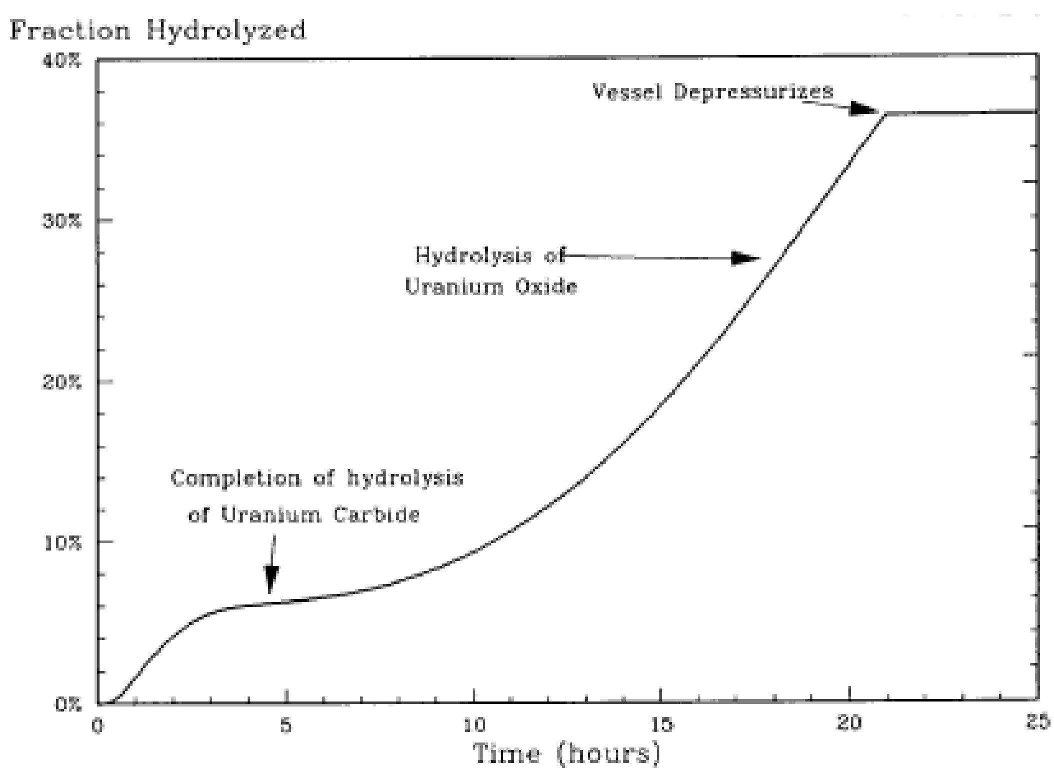


Figure 4-4. Exposed Heavy Metal Hydrolyzed during SRDC-6 for 450 MWt MHTGR

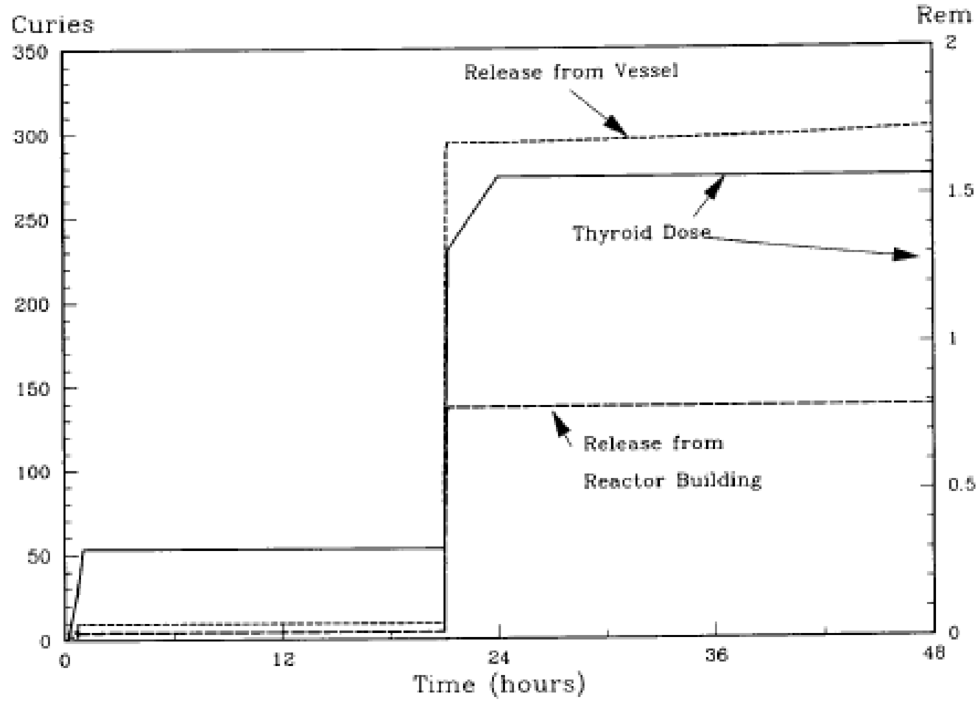


Figure 4-5. Cumulative I-131 Release during SRDC-6 for 450 MWt MHTGR

4.2 Radionuclide Inventories

As indicated in Table 4-1 and Fig. 4-6, a total of 18 radionuclides contribute to more than 95% of the whole-body gamma dose for both the SRDC-6 and SRDC-11 events. As shown in Fig. 4-7, the isotopes I-131, I-132, I-133, I-134 and I-135 are the dominant contributors to the thyroid dose. The radionuclide inventories for the 450 MWt module were obtained from scaling radionuclide design criteria developed for the 350 MWt MHTGR [GA 1987] and were used for the assessments described in [Dilling 1993]. The inventories for the 600 MWt module were obtained from the NGNP Contamination Control Study [Hanson 2008]. The inventories for the 600 MWt module were estimated using assumptions essentially identical to those used previously for the 350 MWt module,[7] and the inventories scale nearly linearly with thermal power level.

[7] In some cases, the fission yields were slightly different to reflect more recent data.

Table 4-1. Inventories of Radionuclides that Dominate Radiological Consequences

		SRDC-6 Dose Contribution (%)		SRDC-11 Dose Contribution (%)		Inventories (Ci)	
Nuclide	Half-Life	Whole-Body Gamma	Thyroid	Whole-Body Gamma	Thyroid	450 MWt	600 MWt
Kr-87	76.0 m	1.39	0	0	0	9.17E+06	9.90E+06
Kr-88	2.8 h	10.77	0	8.26	0	1.28E+07	1.38E+07
Rb-88	17.7 m	3.76	0	2.59	0	1.31E+07	1.41E+07
Ag-110m	252.0 d	4.53	0	0	0	1.77E+04	2.81E+04
Te-131m	30.0 m	1.39	0	0	0	2.13E+06	3.07E+06
I-131	8.041 d	2.42	63.97	7.91	70.76	1.20E+07	1.65E+07
Te-132	78.0 h	3.18	0	1.49	0	1.74E+07	2.37E+07
I-132	2.285 h	19.06	2.97	32.22	1.81	1.76E+07	2.40E+07
Te-133m	55.4 m	1.84	0	0	0	1.44E+07	1.90E+07
I-133	20.8 h	6.26	26.32	15.8	23.89	2.61E+07	3.52E+07
Xe-133	5.29 d	1.13	0	2.12	0	2.61E+07	3.53E+07
I-134	52.6 m	10.52	0.69	4.71	0	2.93E+07	3.92E+07
Cs-134	2.06 y	4.5	0	0	0	1.36E+06	1.90E+06
I-135	6.585 h	11.32	5.75	16.97	3.36	2.43E+07	3.34E+07
Xe-135m	15.3 m	2.03	0	1.66	0	4.77E+06	6.66E+06
Xe-135	9.17 h	1.75	0	2.97	0	3.19E+06	4.40E+06
Ba-137m	2.25 m	8.28	0	0	0	1.05E+06	1.51E+06
Cs-138	32.2 m	1.29	0	0	0	2.52E+07	3.33E+07
	Total	95.42	99.7	96.7	99.82	2.40E+08	3.15E+08

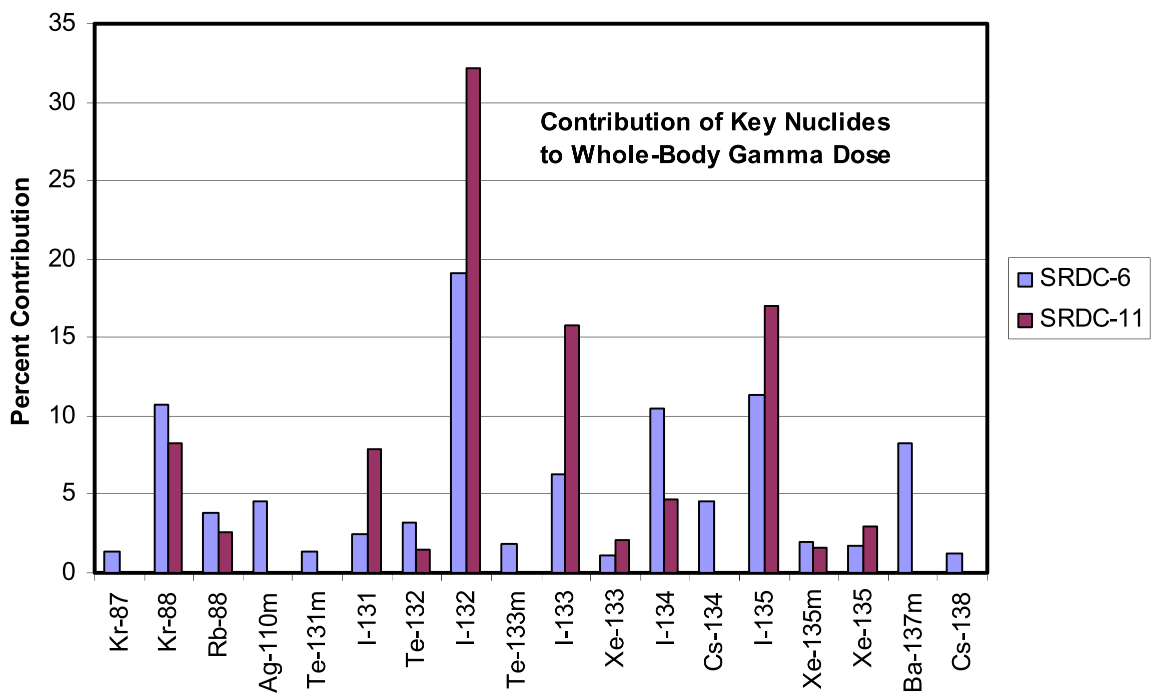


Figure 4-6. Contribution of Key Nuclides to Whole-Body Gamma Dose

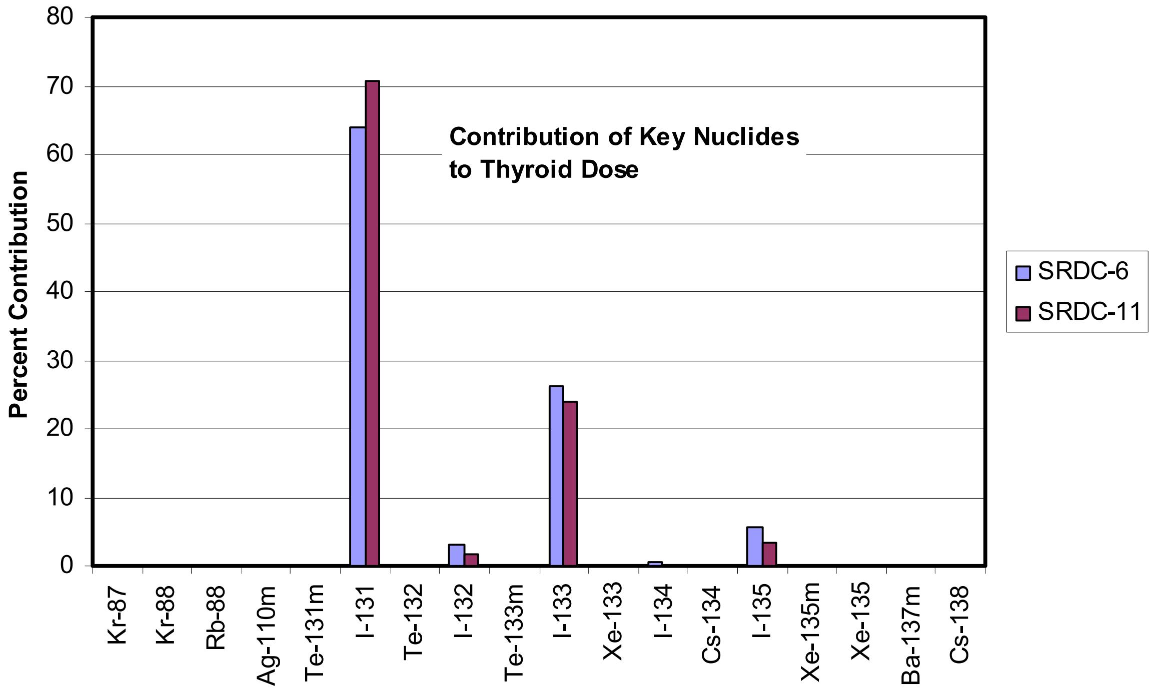


Figure 4-7. Contribution of Key Nuclides to Thyroid Dose

5. RESPONSE OF THE VLPC TO KEY EVENTS

5.1 Main Steam Line Break

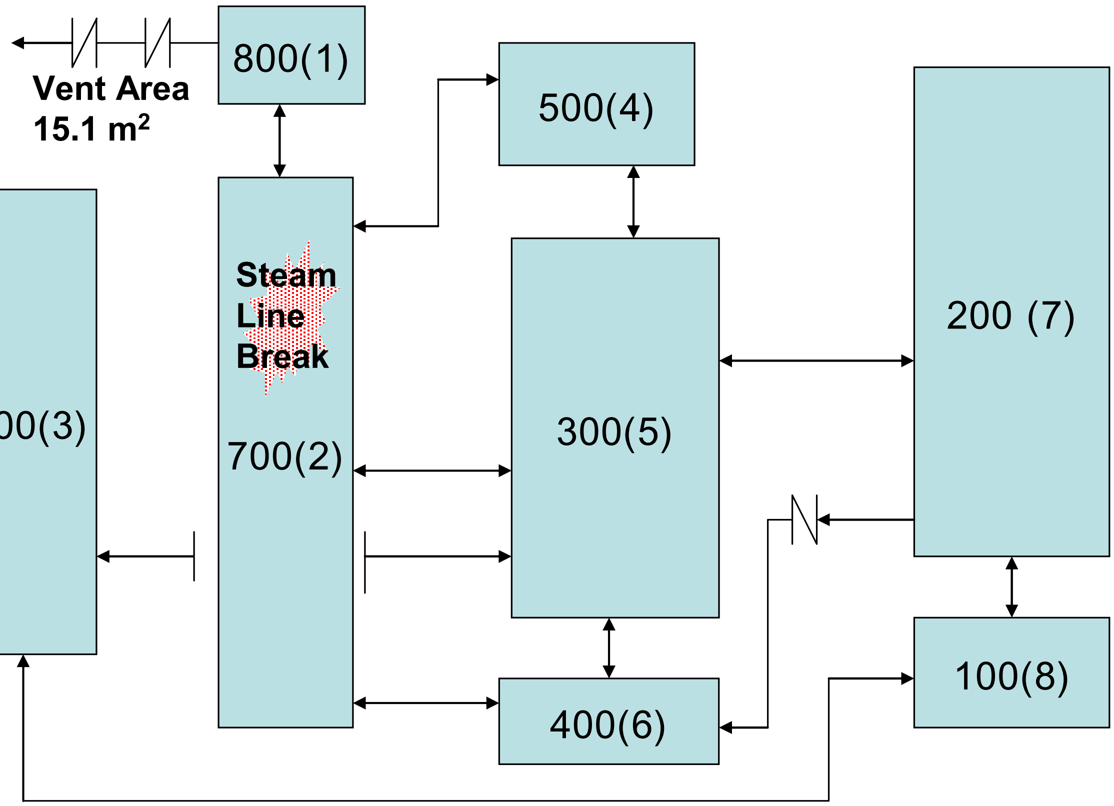
Based on previous analyses, the most severe RB pressure transient occurs following a complete failure (doubled-ended guillotine break) of the main steam line.[8] This event has been modeled for previous MHTGR concepts, including the 450 MWt steam cycle plant [Dilling 1993]. For analysis of this event, the building is modeled as a set of volumes or nodes connected by flow pathways. Blowdown mass and energy per unit time is input into the volume used to represent the break location and the pressure transient is calculated by solving the conservation equations for mass, momentum, and energy. For the present study, KAERI has analyzed this event using the MELCOR code, using the same nodal arrangement that was used to assess this event for the 450 MWt design [Dilling 1993]. The MELCOR model was successfully verified using the input data for the 450 MWt design and comparing with the pressure transient results given in [Dilling 1993]. KAERI then used their GAMMA code to estimate the transient blowdown mass and energy release for a main steam line break for a 600 MWt plant. The MELCOR code was initially run using the same node volumes and vent area used for the 450 MWt plant. With these parameters, the peak pressure during the transient was predicted to be approximately 13 psig, which would exceed the 10 psig design limit for the air-cooled RCCS. The MELCOR model was then revised using the node volumes and vent area shown in Fig. 5-1. The revised model includes a 30% increase for control volumes 300, 600, and 700 and a 45% increase in the vent area to mitigate the more energetic release for a 600 MWt plant. As shown in Fig. 5-2, the peak pressure during this event is predicted to be slightly above 9 psig using the revised model. The KAERI analyses are described in more detail in Appendix C.

5.2 Safety Related Design Conditions

5.2.1 Dose Assessment Methodology

As an approximation for this study, the source terms and radiological consequences for SRDCs are derived from previous safety assessments for the 450 MWt MHTGR [GA 1991] by scaling according to the radionuclide inventories given in Table 4-1 for the 450 MWt and 600 MWt modules. However, the previous safety assessments were performed using dose conversion factors (DCFs) from RG 1.109 and/or RG 1.4. Per RG 1.183 and 1.195, design basis accident analyses should use ICRP-30 or Federal Guidance Reports [FGR-11] and [FGR-12] DCFs. These DCFs are lower and offset the impact of the higher radionuclide inventories for the 600 MWt module. For this study, the FGR-11 and FGR-12 DCFs were applied to the radionuclides in Table 4-1 to estimate radiological consequences.

[8] Because radiological contamination of the secondary steam is controlled to very low levels, the radiological consequences from this event are negligible.



Node	CV #	Description	Volume (m^3)
1	800	Vent path space above steam and and feedwater piping	1,246
2	700	Vent path space from -155 ft to -15 ft	1,031
3	600	Equipment shaft space	699
4	500	Space above main circulator	623
5	300	Steam generator cavity	1,032
6	400	Space below steam generator	297
7	200	Reactor cavity	1,529
8	100	Shutdown cooling system maintenance space	1,246

Figure 5-1. MELCOR Model of the VLPC

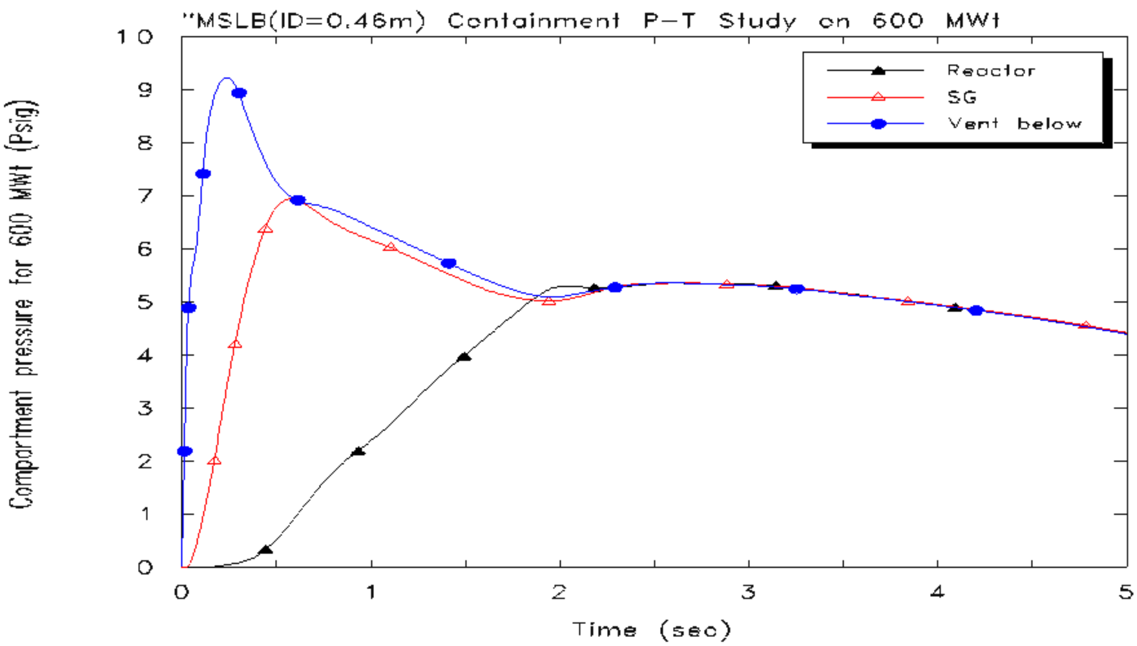


Figure 5-2. MELCOR Predictions of RB Pressure Transient

Doses to the skin from beta radiation and to the whole body from gamma radiation are determined as follows [PSID 1992]:

$$D_j = \chi/Q \times DCF_j \times A_j$$

where $D_j \equiv$ dose from nuclide j to the whole body (or skin) in rem, $\chi/Q \equiv$ atmospheric dispersion factor in s/m^3, $DCF_j \equiv$ whole body (or skin) dose conversion factor for isotope j in rem-m^3/Ci-s, and A_j = the activity released in Ci. Doses to the thyroid gland and other internal organs are incurred by a person breathing the radioactive cloud and the dose is determined as follows:

$$D_{ij} = \chi/Q \times BR \times DCF_{ij} \times A_j$$

where $D_{ij} \equiv$ dose from nuclide j to organ i in rem, $BR \equiv$ breathing rate in m^3/s, and $DCF_{ij} \equiv$ dose conversion factor for isotope j and organ i in rem/Ci.

The χ/Q values are functions of wind speed, wind direction, atmospheric stability conditions, and obstructions in the path of travel. Previous assessments performed for the 350 MWt and the 450 MWt MHTGRs used χ/Q values based on the methodology of RG 1.4. For the dose uncertainty analysis, 10% of the RG 1.4 χ/Q values were used for the 50 percentile value (per RG 4.2) and the RG 1.4 χ/Q values were used as the 95 percentile value with a lognormal distribution. This methodology was chosen since it results in typical values for a potential site and is expected to bound about 85% of all U.S. sites. As discussed in [PSID 1992], a building wake factor of 2.1 was applied to take credit for dilution caused by building wake effects. Only two time periods were considered for these previous assessments; 0 to 8 h and > 8 h. The corresponding 50 percentile χ/Q values were 1.20×10^{-4} s/m^3 and 2.70×10^{-5} s/m^3, respectively. This methodology was also used for the present study.

Future assessments should be performed using the currently recommended RG 1.145 methodology to determine accident-condition χ/Q values.[9] The INL site and a range of other sites should be evaluated using the RG 1.145 methodology in order to better define the design features of the VLPC required to satisfy offsite dose requirements, including any special requirements that may exist for the INL site. The resulting VHTR design should be capable of meeting the requirements for 85% of potential U.S. sites.

As indicated in Tables 5-1 and 5-2, the combination of revised DCFs and higher radionuclide inventories result in offsite doses that are slightly lower than those estimated previously for the 450 MWt MHTGR. The safety analysis codes and methods used previously for the 350 MWt and 450 MWt MHTGRs are described in detail in [Dunn 1987].

5.2.2 SRDC-11

As discussed in Section 4.1.1, SRDC-11 is a slow depressurization resulting from a small break in the primary coolant pressure boundary, leading to a LPCC event. This SRDC results in the largest radionuclide release from the RB for a depressurization event without moisture ingress. Table 5-3 shows the inventories and released activity for two key nuclides, Kr-88 and I-131, which are significant contributors to offsite doses. As indicated in Table 5-3, plateout of I-131 in the RB attenuates the release by approximately a factor of 10. The RB plateout model is described in [Vasquez 1987]. As indicated in Table 5-4, the predicted whole body and thyroid doses at the EAB are significantly below the PAG and 10CFR100 limits.

[9] This methodology requires using a more detailed characterization of the time dependence of the source term (0 h – 2 h, 2 h – 8 h, 8 h – 24 h, 1 d – 4 d, 4 d – 30 d) than used for previous safety assessments to determine the 30-d dose at the EAB. For the present study, the RG 1.4 methodology was used because there was insufficient data available to characterize the time-dependence of the source term required for RG 1.145.

Table 5-1. Adjustments to Whole Body Dose

Nuclide	RG 1.109 DCF	FGR 12 DCF	Ratio (RG 1.109/FGR)	SRDC-6 Whole Body Dose Contribution (%)	SRDC-11 Whole Body Dose Contribution (%)
Kr-87	1.98E-01	1.52E-01	1.30	1.07	0.00
Kr-88	4.89E-01	3.77E-01	1.29	8.32	6.38
Rb-88	1.59E-01	1.24E-01	1.28	2.94	2.02
Ag-110m	6.83E-01	5.03E-01	1.36	3.34	0.00
Te-131m	3.57E-01	2.59E-01	1.37	1.01	0.00
I-131	9.53E-02	6.73E-02	1.41	1.71	5.59
Te-132	5.77E-02	3.81E-02	1.51	2.10	0.98
I-132	5.73E-01	4.14E-01	1.38	13.79	23.31
Te-133m	5.55E-01	4.22E-01	1.31	1.40	0.00
I-133	1.52E-01	1.09E-01	1.39	4.49	11.33
Xe133	1.13E-02	5.77E-03	1.96	0.58	1.08
I-134	6.56E-01	4.81E-01	1.37	7.70	3.45
Cs-134	3.89E-01	2.80E-01	1.39	3.24	0.00
I-135	3.94E-01	2.95E-01	1.33	8.48	12.71
Xe-135m	1.08E-01	7.54E-02	1.43	1.42	1.16
Xe-135	6.20E-02	4.40E-02	1.41	1.24	2.11
Ba-137m	1.49E-01	1.07E-01	1.40	5.90	0.00
Cs-138	5.90E-01	4.48E-01	1.32	0.98	0.00
Effective fractional dose with DCF adjustment				0.70	0.70
Effective fractional dose with DCF and power adjustment				0.93	0.94

Table 5-2. Adjustments to Thyroid Dose

Nuclide	RG 1.109 DCF	FGR 11 DCF	Ratio (RG 1.109/FGR)	SRDC-6 Thyroid Dose Contribution (%)	SRDC-11 Thyroid Dose Contribution (%)
I-131	1.49E+06	1.08E+06	1.38	46.37	51.29
I-132	1.43E+04	6.43E+03	2.22	1.34	0.81
I-133	2.69E+05	1.80E+05	1.49	17.61	15.99
I-134	3.73E+03	1.06E+03	3.52	0.20	0.00
I-135	5.60E+04	3.13E+04	1.79	3.21	1.88
Effective fractional dose with DCF adjustment				0.69	0.70
Effective fractional dose with DCF and power adjustment				0.92	0.93

Table 5-3. Release of Key Nuclides for SRDC-11

	Kr-88	I-131
Inventory (Ci)		
Total	1.38E+07	1.65E+07
Available for Release	828	990
Activity Released (Ci)		
Circulating	5.55	0.04
Liftoff	0	7.1E-04
Heatup (30 d)	9.2	120
Release from Vessel	5.0	65.8
Release from Building	2.6	6.9

Table 5-4. SRDC-11 Dose Comparisons

	Whole Body (Rem)	Thyroid (Rem)
Dose at EAB (30 d)	*0.0008*	*0.154*
PAG Limit	*1.0*	*5.0*
10CFR100 Limit	*25.0*	*300.0*
Background (30 d)	*0.00125 – 0.0117*	*N/A*

There are large uncertainties associated with transport of iodine in the RB, and additional data for transport of iodine under prototypical VHTR VLPC conditions should be obtained to reduce the uncertainties and improve the modeling capabilities. For LWRs, the MELCOR code is typically used to model the response of the containment building under accident conditions. For this study, KAERI used the MELCOR code to model the SRDC-11 event using built-in MELCOR models for iodine transport that are based on LWR data. Calculations were performed assuming (1) iodine was in elemental form (I_2) and (2) iodine was in the form of CsI. Because elemental iodine has a high vapor pressure, results for the first case indicated the RB provided essentially no attenuation of iodine release. The second case is consistent with data from the accident that occurred at Unit 2 of the Three Mile Island plant and indicated the RB would attenuate iodine release by about a factor of 20. As discussed above, the RB iodine transport model used for previous safety assessments indicated the RB would attenuate iodine release by about a factor of 10, which falls between the two limiting cases analyzed with the MELCOR code. Figure 5-3 shows the cumulative release of I-131 assuming iodine transported as CsI and a building leak rate of 1 volume per day. Figure 5-4 shows the distribution of deposited I-131 activity among the MELCOR control volumes (see Fig. 5-1). Approximately two-thirds of the I-131 activity is predicted to deposit in the reactor cavity, where the small break in the primary coolant pressure boundary is assumed to occur.

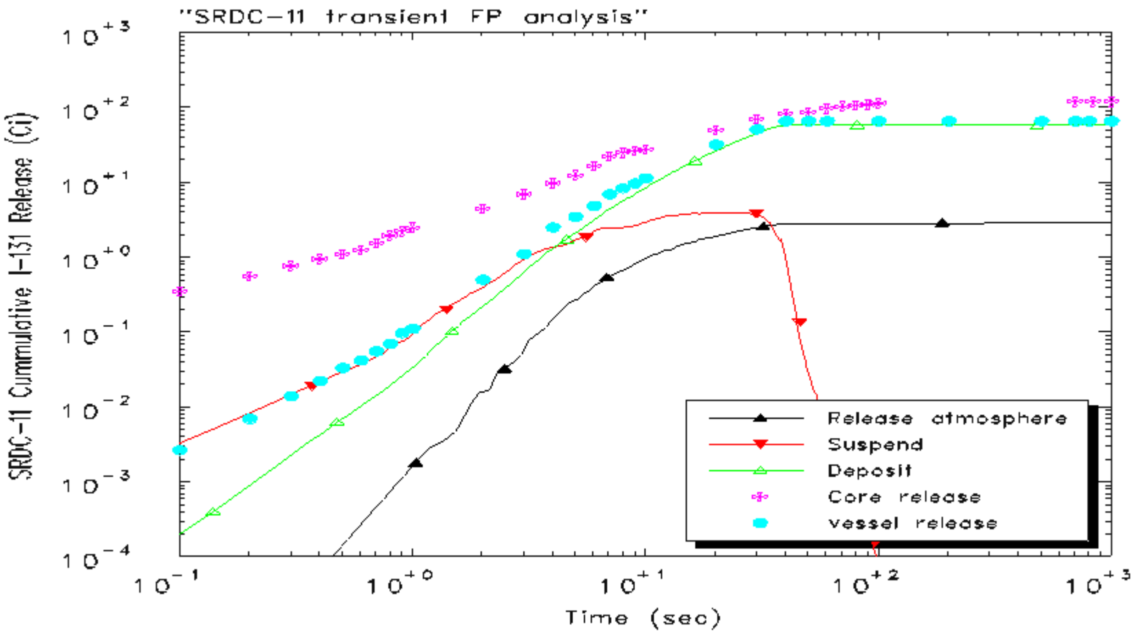


Figure 5-3. MELCOR Predictions for I-131 Transport (as CsI) in the RB During SRDC-11

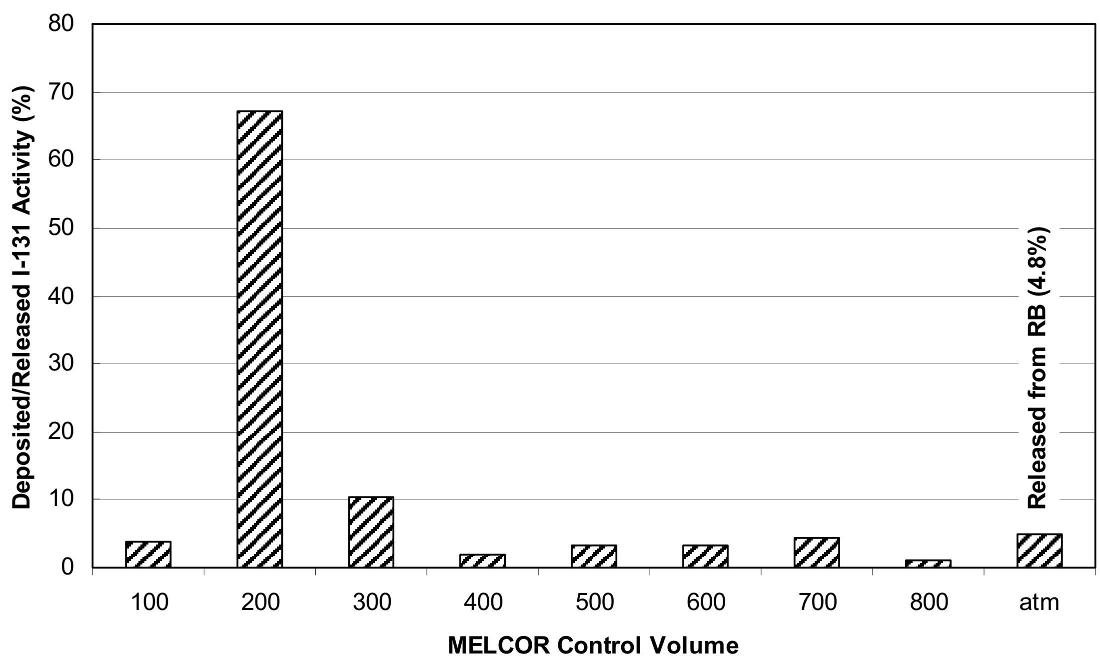


Figure 5-4. MELCOR Predictions for Distribution of I-131 Activity During SRDC-11

5.2.3 SRDC-6

As discussed in Section 4.1.2, SRDC-6 is a moisture ingress event caused by failure of a steam generator tube, followed by a sequence of depressurizations through the primary coolant relief valves and a delayed LPCC. This SRDC results in the largest radionuclide release from the RB. Table 5-5 shows the inventories and released activity for two key nuclides, Kr-88 and I-131, which are significant contributors to offsite doses. As indicated in Table 5-5, plateout of I-131 in the RB attenuates the release by approximately a factor of 2 for this event.[10] As indicated in Table 5-6, the predicted whole body dose at the EAB is significantly below the PAG and 10CFR100 limits. However, the predicted thyroid dose is only about a factor of 3 below the PAG limit, which emphasizes the need to accurately characterize the iodine source term and its transport behavior in all of the containment barriers, including the RB.

Table 5-5. Release of Key Nuclides for SRDC-6

	Kr-88	I-131
Inventory (Ci)		
Total	1.38E+07	1.65E+07
Available for Release	828	990
Activity Released (Ci)		
Circulating	5.55	0.04
Steam Induced Vaporization/Recirculation	0	21.1
Fuel Hydrolysis	252	301
Graphite Oxidation	neg.	neg.
Heatup (30 d)	9.2	98.9
Release from Vessel	73.4	371
Release from Building	69.1	188

Table 5-6. SRDC-6 Dose Comparisons

	Whole Body (Rem)	Thyroid (Rem)
Dose at EAB (30 d)	*0.015*	*1.5*
PAG Limit	*1.0*	*5.0*
10CFR100 Limit	*25.0*	*300.0*
Background (30 d)	*0.00125 – 0.0117*	*N/A*

5.3 Beyond Design Basis Events with Air Ingress

For previous MHTGR and GT-MHR designs, only beyond design basis events (frequencies below 5×10^{-7} per year) that result in massive failure of the reactor coolant pressure boundary

[10] As indicated in Fig. 5-3, the RB was predicted to attenuate I-131 release by about a factor of 10 for SRDC-11. The reduced attenuation by the RB for SRDC-6 is caused primarily by the different RB conditions for SRDC-6, including the presence of steam.

can result in significant levels of air ingress into the RPV, eventually leading to graphite oxidation. For the 350 MWt steam-cycle MHTGR, assessments of beyond design basis events are described in Appendix G of the MHTGR Probabilistic Risk Assessment (PRA) [PRA 1988]. One of these events is catastrophic failure of the cross vessel and hot duct, which opens both the hot and cold legs and provides a pathway for natural circulation of air through the core. Ruptures of this size fall outside of the design basis and are only possible if a large defect in excess of the critical size for crack propagation has gone undetected. The following sections provide an overview of the accident phenomena and describe independent assessments performed by FES and KAERI.

5.3.1 Overview of Air Ingress Phenomena

<u>Graphite Combustibility</u>

High-purity, nuclear-grade graphites react very slowly with oxygen and are noncombustible by conventional standards. Nevertheless, graphite combustibility has always been a subject of discussion for graphite-moderated reactors, primarily because of the accidents at Windscale in 1957 and Chernobyl in 1986 [Richards 1995]. An assessment of the Windscale accident concluded oxidation occurred primarily with the metallic uranium fuel. An analysis of the Chernobyl accident [Richards 1988] showed any convective airflow into the damaged core would provide a net cooling effect, i.e., heat removed by convection was predicted to be greater than heat generated by the exothermic reaction of graphite with oxygen, and the dominant heat source causing the “red glow” was nuclear decay heat.

One definition for graphite combustion is rapid oxidation at high temperatures. Burning may be defined as self-sustained combustion, such that high temperatures are maintained by the combustion process itself, i.e., the combustion heat-generation rate matches or exceeds the heat losses by conduction, convection, and radiation. For burning to occur, several conditions must exist simultaneously:

1. An appropriate geometry with high surface-to-volume ratio.
2. Sufficiently high temperatures.
3. An adequate oxygen supply.
4. A high intrinsic reaction rate.
5. A favorable heat balance with small heat losses.

These conditions are very difficult to achieve for high-purity, nuclear-grade graphites. The heat generation rates are low because of very low concentrations of volatiles and catalytic impurities and available reaction sites are reduced by graphitization. At higher temperatures, reaction rates are further limited by diffusion of oxygen across the boundary layer and into the graphite pores. Heat losses are relatively high because of the high thermal conductivity and emissivity of graphite and convective cooling provided by low-temperature, ambient air.

In contrast, charcoal and coal oxidize at much faster rates for the following reasons:

1. High levels of impurities catalyze the oxidation reaction.
2. High porosity provides large internal surface area, resulting in a more homogeneous reaction.
3. Volatile gases are generated (e.g., methane) which react exothermically to increase temperatures.
4. Porous ash is formed, which allows oxygen to pass through, but reduces heat losses by conduction and radiation.
5. The thermal conductivities and specific heats are lower than those for nuclear-grade graphites.

Nuclear-grade graphites exhibit none of the above properties. In fact, powdered graphite is used as a fire extinguishing material for highly reactive metals, including zirconium.

Representative tests for assessing combustibility of nuclear-grade graphite were performed at Los Alamos National Laboratory (LANL) as part of the New Production Reactor program [Richards 1995]. The test specimens consisted of annular H-451 graphite [11] tubes that simulated the geometry of a single coolant hole of a MHR prismatic fuel element. The tube dimensions were as follows: inner diameter = 1.6 cm, outer diameter = 3.4 cm, and length = 80 cm. The specimens were heated in a 3-zone furnace which was insulated to minimize heat losses. An oxygen/nitrogen mixture was preheated to near the graphite temperature (which greatly reduced convective heat losses) and introduced into the annular region of the test specimen. Thermocouples were used to measure graphite temperatures along the length of the specimen. The concentrations of O_2, CO, and CO_2 in the effluent gas were also measured. Two tests were completed for initial graphite temperatures of ~500°C and ~700°C, respectively. The mass-flow rate of 0.015 g/s was selected to be representative of a buoyancy-driven flow and the inlet O_2 mole fraction of 0.1 was judged to be representative of hypothetical accident scenarios, in which some oxygen depletion would occur before the airflow entered the reactor core.

For the test with 500°C initial temperature, very little oxidation occurred and the graphite temperature remained near the initial temperature (primarily because of the furnace heaters). Figure 5-5 shows a schematic of the test configuration and the graphite temperature response for the test with 700°C initial temperature. Despite the absence of any significant mechanisms for heat removal, the maximum temperature increase after 6.5 hours of oxidation was only

[11] Grade H-451 graphite was selected as the material for the GT-MHR fuel elements. The coke source for this graphite is no longer available, but the oxidation rates for highly purified, nuclear-grade graphites are similar, and the H-451 oxidation rate is expected to be about the same as those for alternative grades that are currently under development.

about 25°C and occurred at an axial location approximately 25 cm from the tube inlet. The total graphite burnoff was about 1.1%. Measurements of the surface burnoff indicated oxidation was occurring in the in-pore, diffusion-controlled regime. Measurements of the exit gas concentration showed nearly all of the O_2 was converted to CO_2. This result indicated CO deflagration occurred in the flow channel, since the dominant graphite/oxygen reaction product at 700°C should be CO. However, CO deflagration apparently had little effect on the graphite temperature response and the overall oxidation behavior.

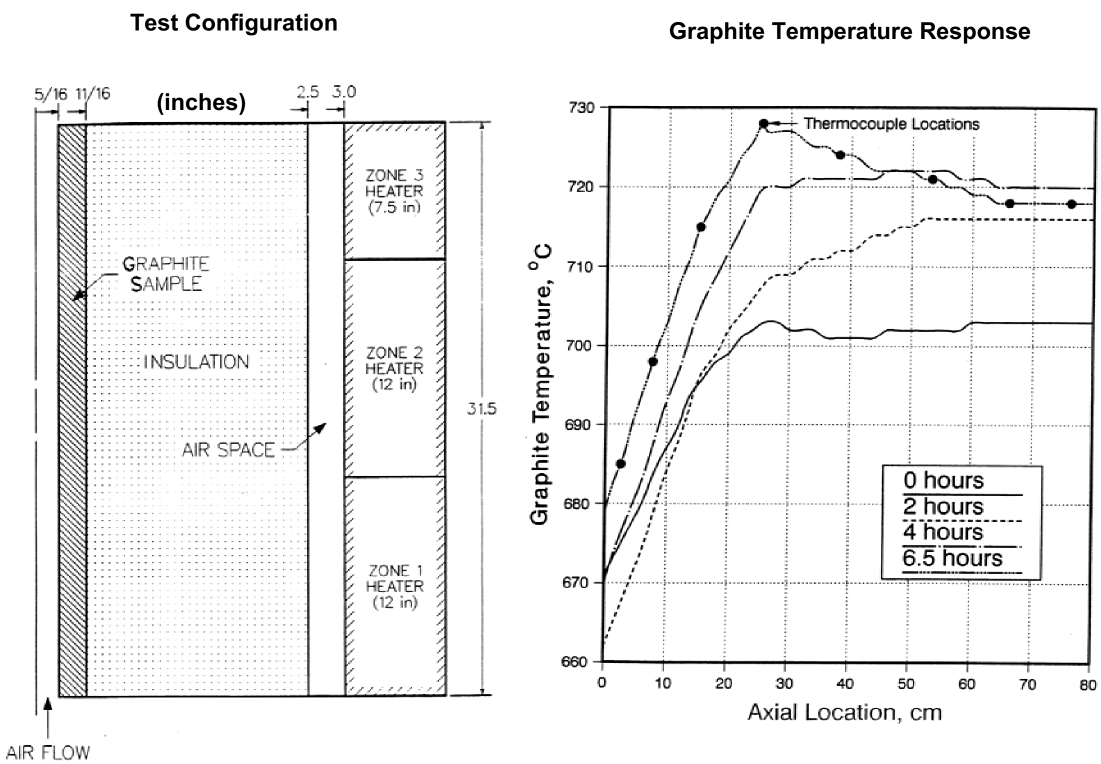


Figure 5-5. LANL H-451 Graphite Oxidation Test

In the mid 1980s, there was some interest in re-starting the graphite-moderated N-Reactor, located on the Hanford reservation in Washington State. However, because of the age of the N-Reactor, some of the nuclear-grade TSX graphite had started to expand.[12] Since continued expansion could compromise structural integrity, a study was performed to develop methods for removing graphite [Reich 1986]. One of the methods considered was to use oxyacetylene torches to burn off the excess graphite. Demonstration tests were performed on TSX graphite blocks with the following results:

[12] Graphite initially shrinks with neutron fluence, but starts to expand above a threshold fluence.

- After 57 minutes of exposure to two torches, the maximum surface temperature of the blocks was approximately 1650°C (measured using an optical pyrometer). The peak recorded thermocouple temperature was 1260°C.
- For several minutes, the acetylene to one torch was shut off and pure oxygen was blown directly onto the hot block. Because of its high velocity, the oxygen jet had a net cooling effect and temperatures near the region below the torch nozzle quickly dropped. Both torches were removed after 65 m of exposure.
- After the torches were removed, the graphite cooled off. Despite the very high graphite temperatures, no self-sustained combustion occurred. The total graphite burnoff was approximately 3%.

Throughout the test, the blocks were glowing from red at the lower-temperature locations to yellow-white at locations directly below the torches. The following conclusion was given in [Reich 1986]:

> "There is a common perception taken from our experiences with coal and charcoal that when a mass of these fuels achieves a glowing red condition a self-sustaining combustion is underway. Transferring this perception to graphite has led to repeated references to 'burning' graphite when in fact a self-sustaining reaction was not in progress. The test sequences described in these tests demonstrate how difficult it can be to achieve conditions for self-sustained combustion of graphite."

Based on these tests, it was concluded that thermal milling of N-Reactor graphite was not feasible.

<u>Cross Vessel Rupture Event</u>

The event described in Appendix G of the PRA [PRA 1988] assumes a near instantaneous rupture of the cross vessel and hot duct. The flow area for depressurization and eventual air ingress is limited to some extent because of constraints on the relative movements of the RPV and SG vessel. However, the vessel completely depressurizes in less than 10 seconds, and the peak pressure in the reactor cavity greatly exceeds design limits for the RCCS and RB (see Appendix C). The reactor trips on low primary coolant pressure and the ultimate heat sink is conduction to the ground surrounding the silo portion of the RB. The RB is assumed to be damaged to the extent that it is unable to provide any significant retention of radionuclides by settling or plateout mechanisms.

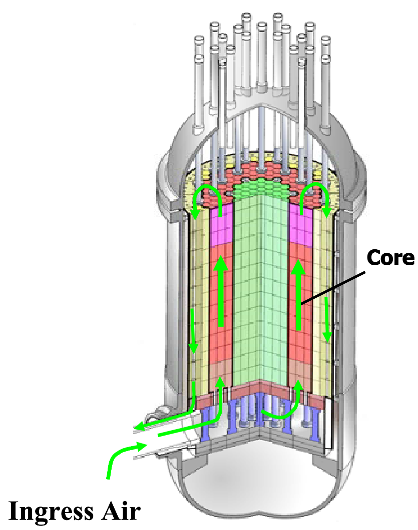
Following the rupture, the RPV resembles an inverted bottle containing light gas. Because counter diffusion of air is a relatively slow process, up to several days may be required for air to replace helium and flow via natural convection through the RPV. The supply of air is essentially unlimited, but the flow rate is limited by friction associated with the long, small-diameter channels within the fuel and reflector blocks. The flow rate is further limited as the core heats up because of the increase in air viscosity with temperature. Eventually the core cools down and the oxidation stops completely. The total graphite mass loss is typically a few percent of the total and most of the oxidation is predicted to occur in the lower plenum and reflectors. Oxidation that occurs in these lower regions limits the availability of oxygen to fueled regions of the core. The overall heat rate from oxidation remains well below the decay heat rate throughout the course of the accident, and graphite oxidation contributes very little to the radiological consequences of the accident.

5.3.2 Assessments Performed by Fuji Electric Systems

As part of this study, FES performed an assessment of the cross-vessel rupture event and prepared a report titled "Analytical conditions and results for graphite oxidation analysis" [FES 2008].[13] Key results from this report are summarized below.

<u>Analytical Model and Calculational Procedure</u>

Calculations were performed using both the commercial ANSYS code and the GRACE code. The GRACE code was developed by FES and was used during preparation of the safety review for the Japan Atomic Energy Agency (JAEA) High Temperature Engineering Test Reactor (HTTR). As shown in Fig. 5-6, this event progresses in 3 phases, starting with a rapid depressurization, followed by a period of several days or more for air to displace helium, followed by a period of natural circulation airflow during which graphite oxidation can occur.



<u>Path of air ingress due to natural circulation</u>

- 1st Phase – Rapid Depressurization
  - Pressure balances between inside and outside of the RPV.
- 2nd Phase – Molecular Diffusion
  - The air enters the RPV gradually by molecular diffusion. Graphite oxidation during this phase is negligible.
- 3rd Phase – Natural Circulation
  - Natural circulation eventually established after air concentration in the RPV becomes sufficiently large.

Figure 5-6. Air Ingress Event Sequence

[13] The distribution of this document is limited because of Japanese nuclear export control restrictions.

ANSYS was used for thermal calculations, including estimates of natural circulation flow rates. The ANSYS model is shown in Fig. 5-7 and is essentially the same as that used previously by FES to support the RPV and IHX Pressure Vessel Alternatives Study [Richards 2008].

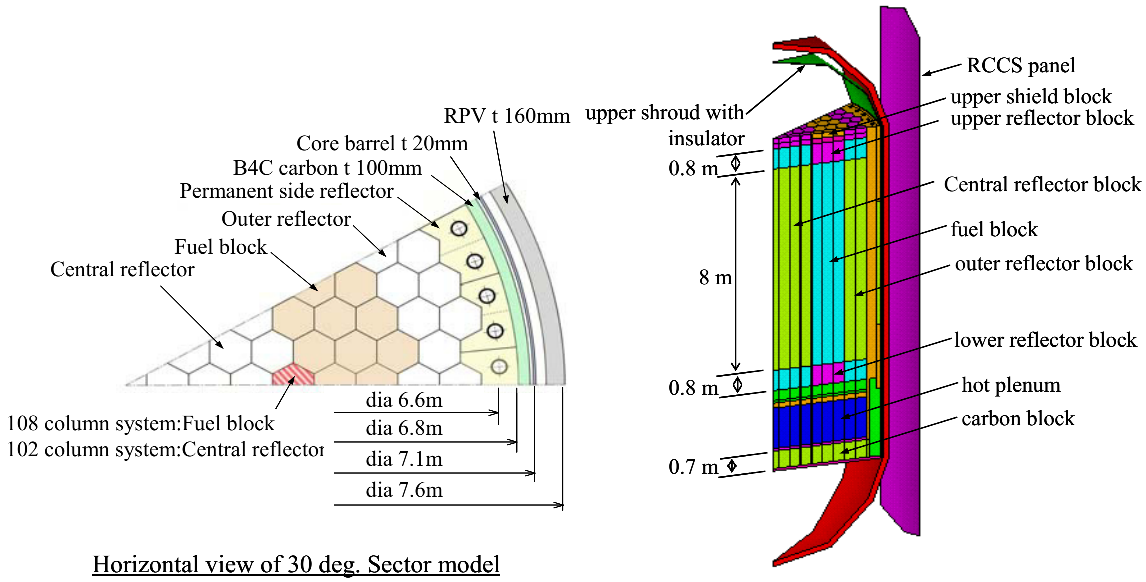


Figure 5-7. ANSYS Model Used to Simulate Air Ingress Event

GRACE was used to estimate graphite oxidation, including changes in gas composition as the air flows through the RPV. Figure 5-8 shows the GRACE model used for the analyses. Three fuel channels are used to represent the core, with Channel A representing the channel with highest graphite temperatures. Input data for graphite oxidation were based on H-451 graphite design data provided by GA (see Section 3.2 of Appendix D). A single volume mixing model was used to estimate the time-dependent gas composition in the RB assuming (1) the RB volume is 8,500 m^3, (2) initial gas composition in the RB is pure air, (3) outside air is supplied at a rate corresponding to the design RB leak rate of 1 volume per day, and (4) depletion of oxygen caused by graphite oxidation. Iterations were performed between ANSYS and GRACE until convergence of temperatures and graphite oxidation was obtained.

For this event, the pressure transient is expected to be sufficiently severe to damage the RCCS. For these studies, existing models were utilized that include an operational RCCS. However, previous assessments of beyond design basis LPCC events with RCCS failure have shown that conduction cooldown to the ground results in only somewhat elevated fuel and graphite temperatures, with the primary impacts being RPV and concrete temperatures that would exceed ASME code limits [PRA 1988].

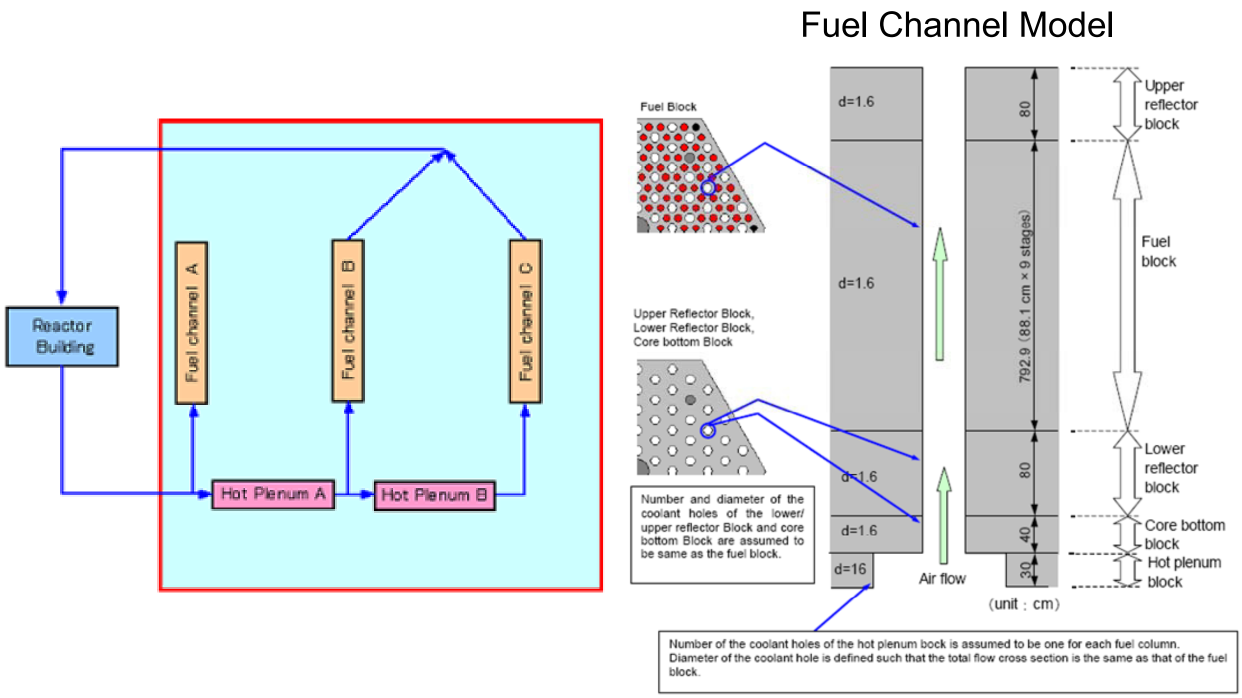


Figure 5-8. GRACE Model Used to Simulate Air Ingress Event

Summary of Results

For this study, the time for onset of natural circulation of air was treated as a parameter, but was considered to occur relatively early following rupture of the cross vessel. Five cases were analyzed:

Case 1: Natural circulation starts immediately after depressurization (0 d)

Case 2: Natural circulation starts 24 h after depressurization (1 d)

Case 3: Natural circulation starts 48 h after depressurization (2 d)

Case 4: Natural circulation starts 72 h after depressurization (3 d)

Case 5: Natural circulation starts 96 h after depressurization (4 d)

As shown in Fig. 5-9, tentative corrosion limits were based on the geometries of the fuel blocks and reflector/core bottom blocks. For the fuel block, the limit is set to the minimum web thickness between the coolant hole and fuel compact (0.4 cm). For the reflector/core bottom blocks, the limit is set to half of the web thickness between coolant holes (0.8 cm).

In general, results for all cases showed low levels of oxidation in the active core and very little impact of graphite oxidation on the temperature response of the core, even if natural circulation of air were sustained for time periods of 500 h or more. Because of oxygen depletion, oxidation in the active core is generally limited to the bottom-most layer of fuel blocks. Hence, the radiological consequences of this event associated with graphite oxidation are expected to be small compared with those associated with a LPCC without air ingress. Figure 5-10 shows the fuel temperature response at the core bottom and at the axial midplane where the peak fuel temperatures occur for Case 1 and for an LPCC event without air ingress.

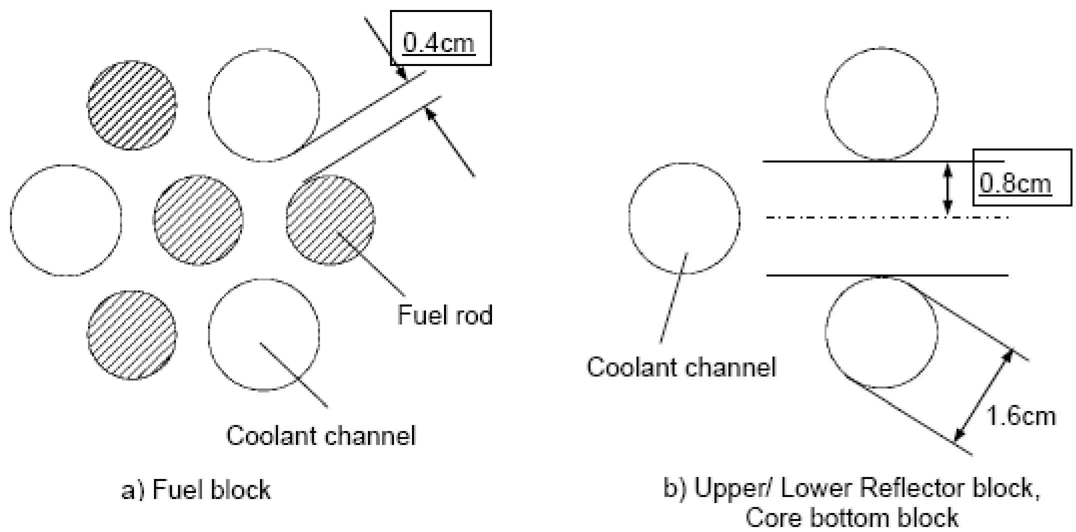


Figure 5-9. Tentative Limits for Graphite Corrosion Based on Block Geometries

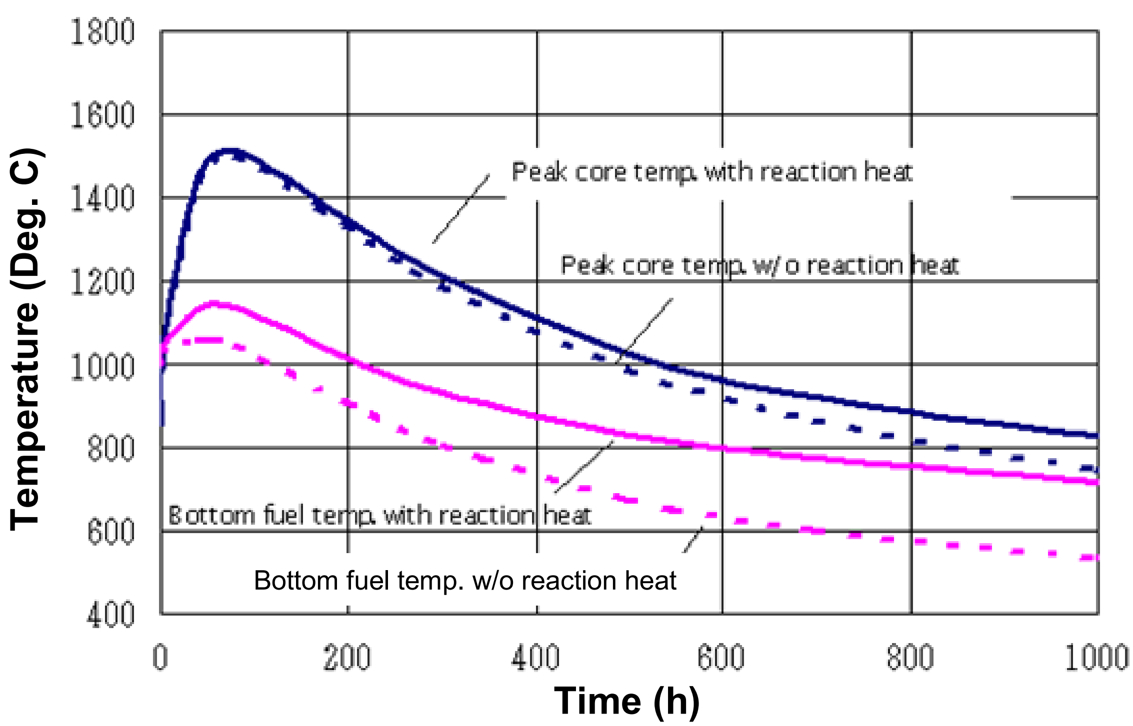


Figure 5-10. Fuel Temperature Response with and without Air Ingress

Figure 5-11 shows the corresponding axial temperature distribution in Channel A for Case 1. Figure 5-12 shows the total air mass flow rate as a function of time and Fig. 5-13 shows the time dependence of the oxygen concentration in the RB.

As shown in Fig. 5-14, significant levels of oxidation can occur in the lower graphite structures below the active core, and complete oxidation of the graphite web between the coolant holes can occur within about 100 hours after the onset of airflow. However, as shown in Fig. 5-15, approximately 700 hours (approximately 1 month) of continuous airflow is required to consume the graphite web between coolant holes and fuel blocks in the bottom-most layer of the active core. Figure 5-16 shows the axial distribution of the equivalent corrosion thickness for Case 1 and Case 5. Figure 5-17 shows the fraction of graphite oxidized in the active core as a function of time. Approximately 500 h (21 d) of continuous airflow is required to oxidize 2% of the active core.

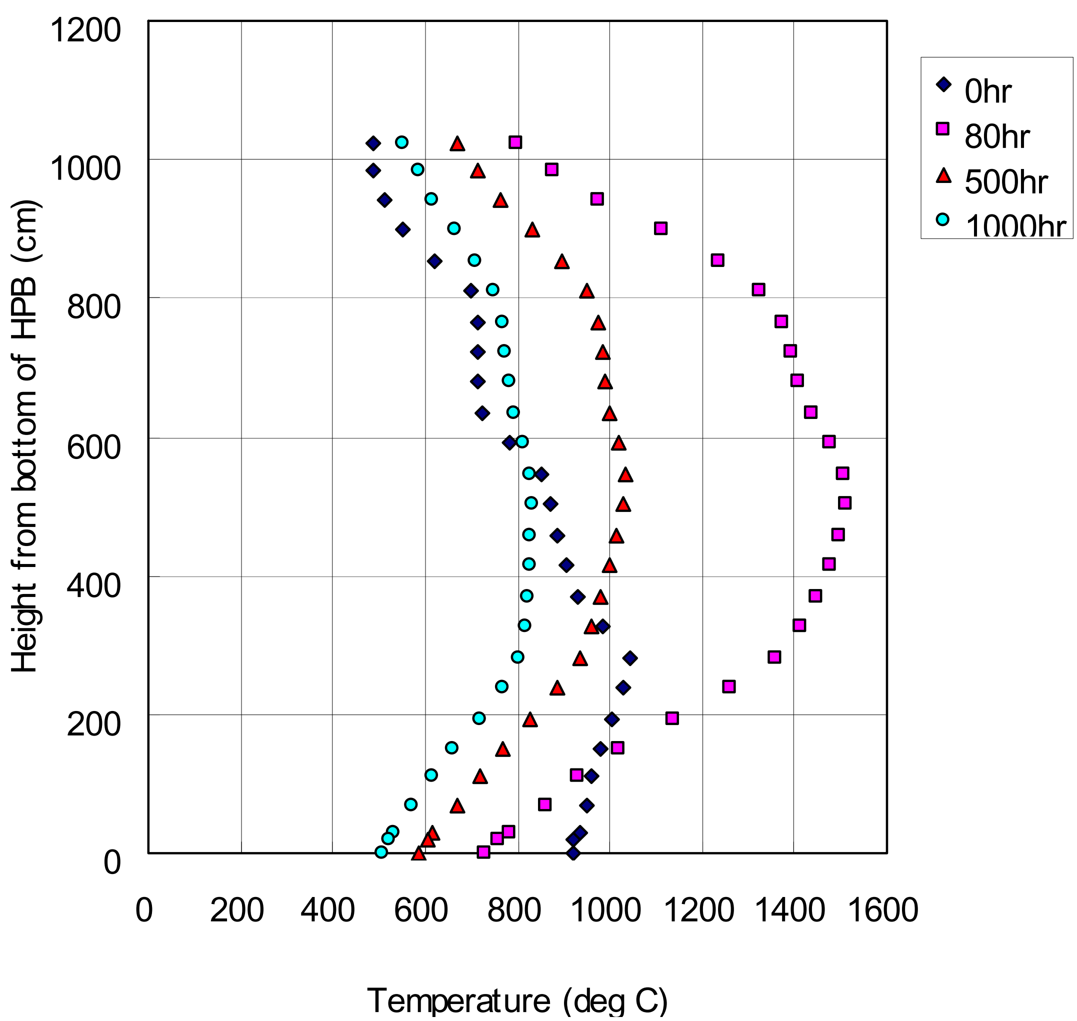


Figure 5-11. Axial Temperature Distribution for Fuel Channel A

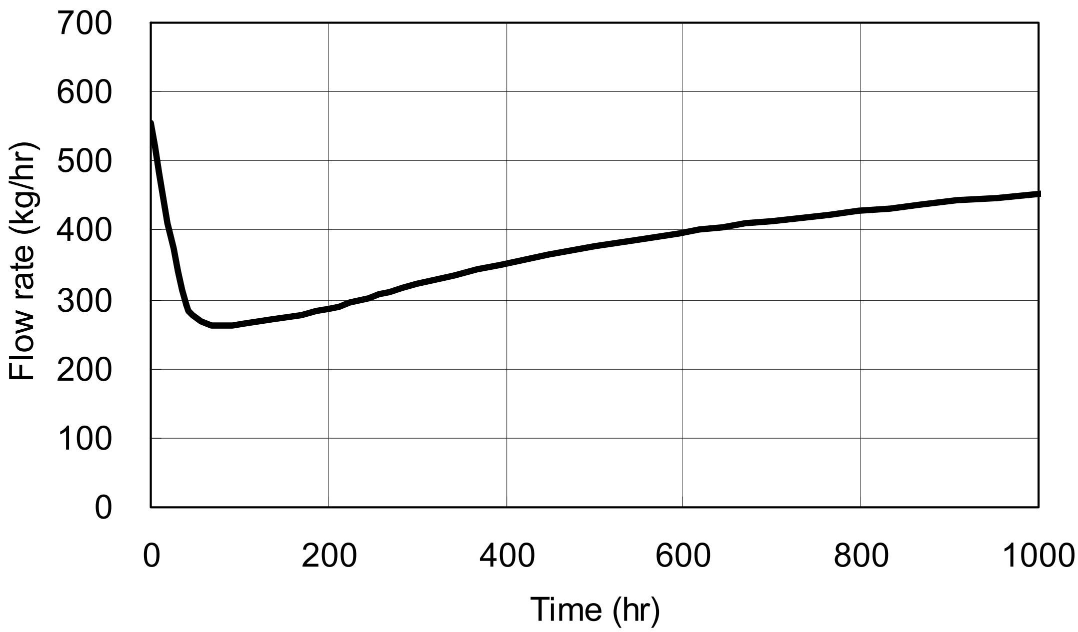


Figure 5-12. Total Air Mass Flow Rate during Air Ingress Event

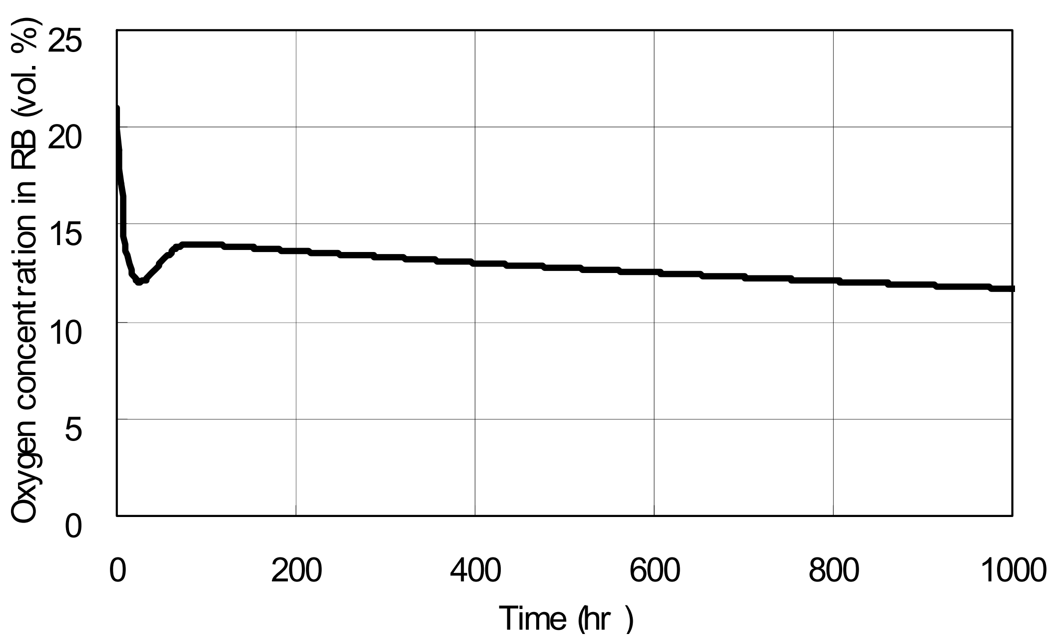


Figure 5-13. Oxygen Concentration in the RB during Air Ingress Event

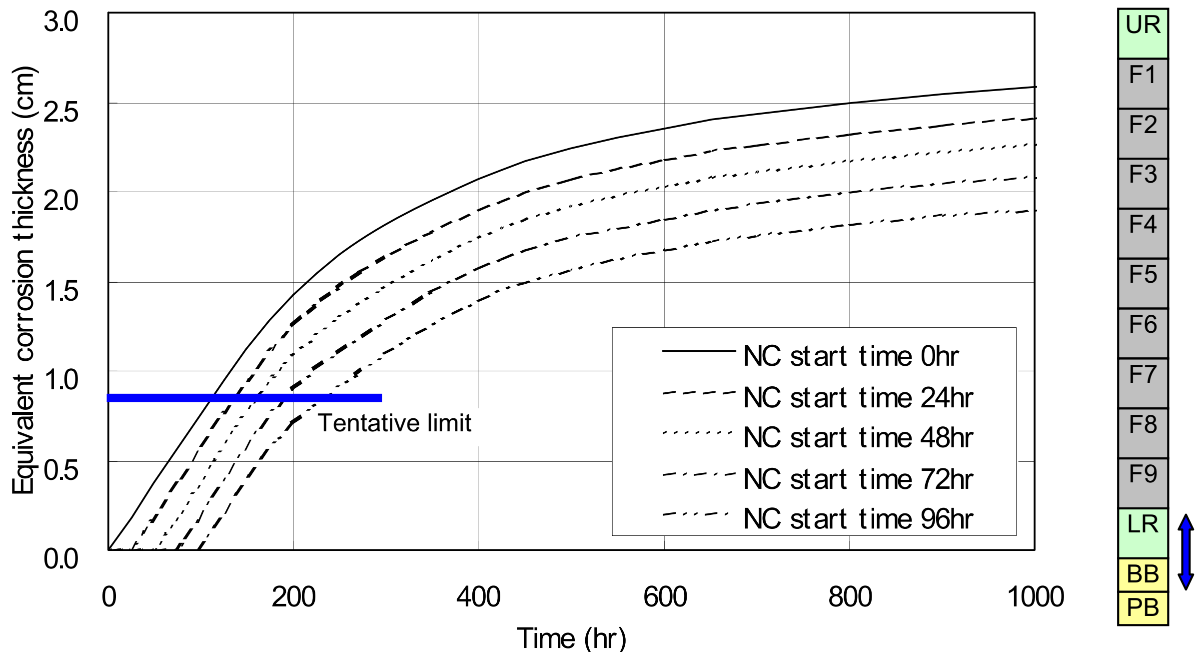


Figure 5-14. Equivalent Corrosion Thickness, Lower Graphite Structures, Channel A

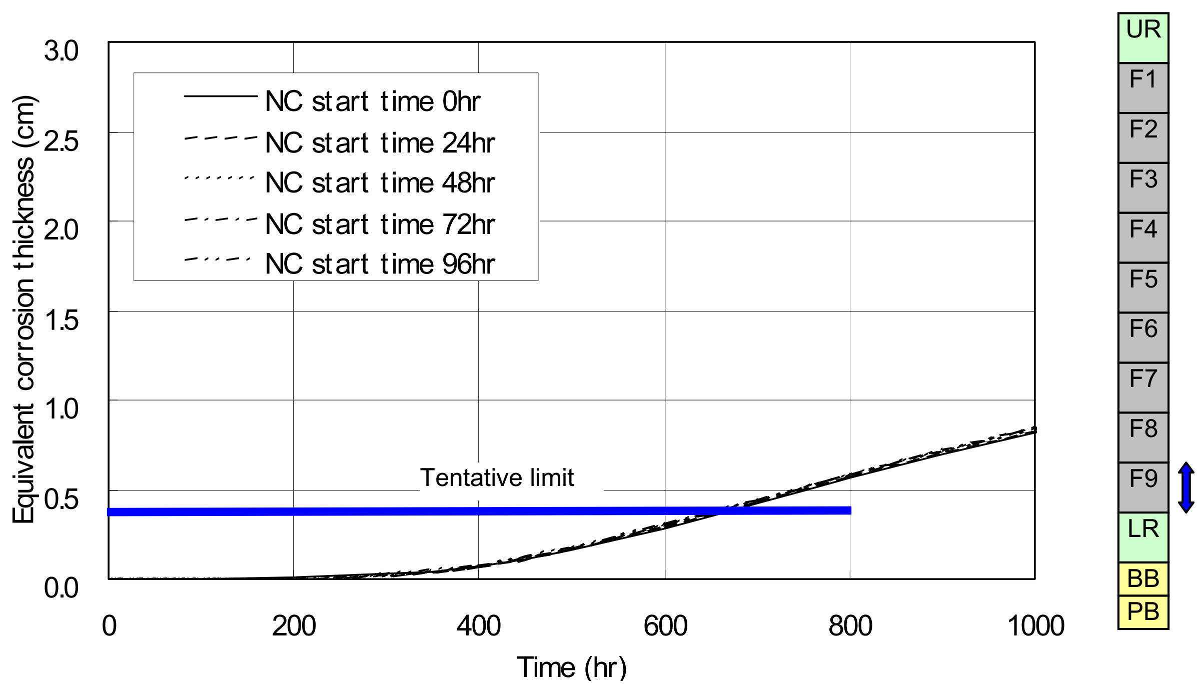


Figure 5-15. Equivalent Corrosion Thickness, Bottom-Most Fuel Element, Channel A

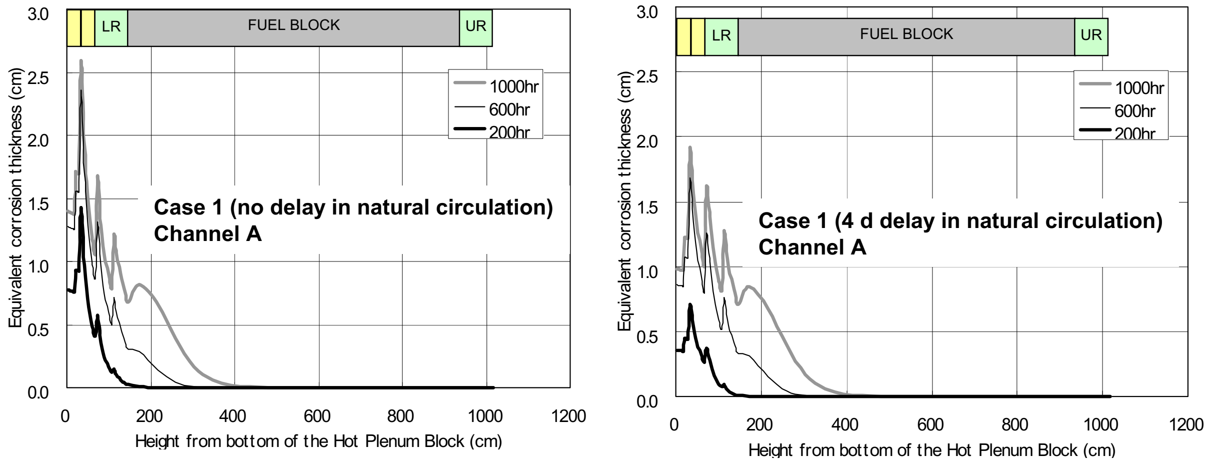


Figure 5-16. Axial Distribution of Equivalent Corrosion Thickness

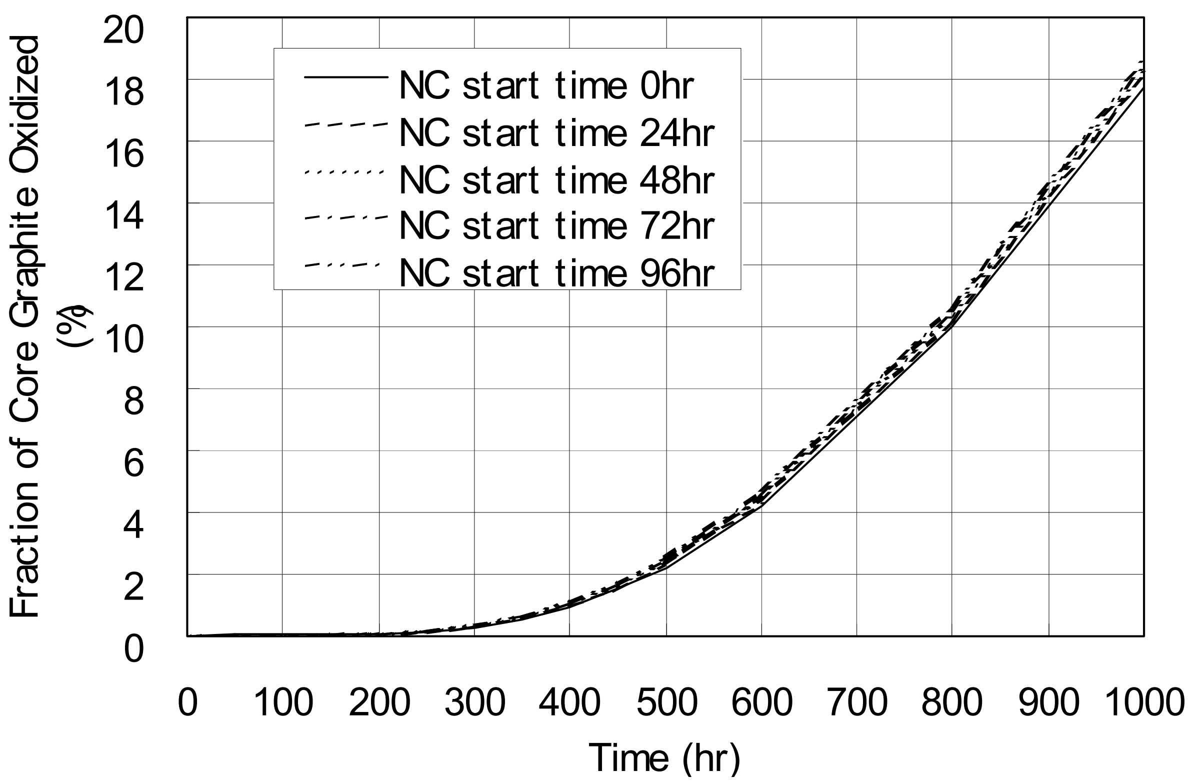


Figure 5-17. Fraction of Graphite Oxidized in the Active Core Region

An overall heat balance shows the heat rate from graphite oxidation during the air ingress event is much smaller than the decay heat rate and the RCCS heat removal rate (see Fig. 5-18).

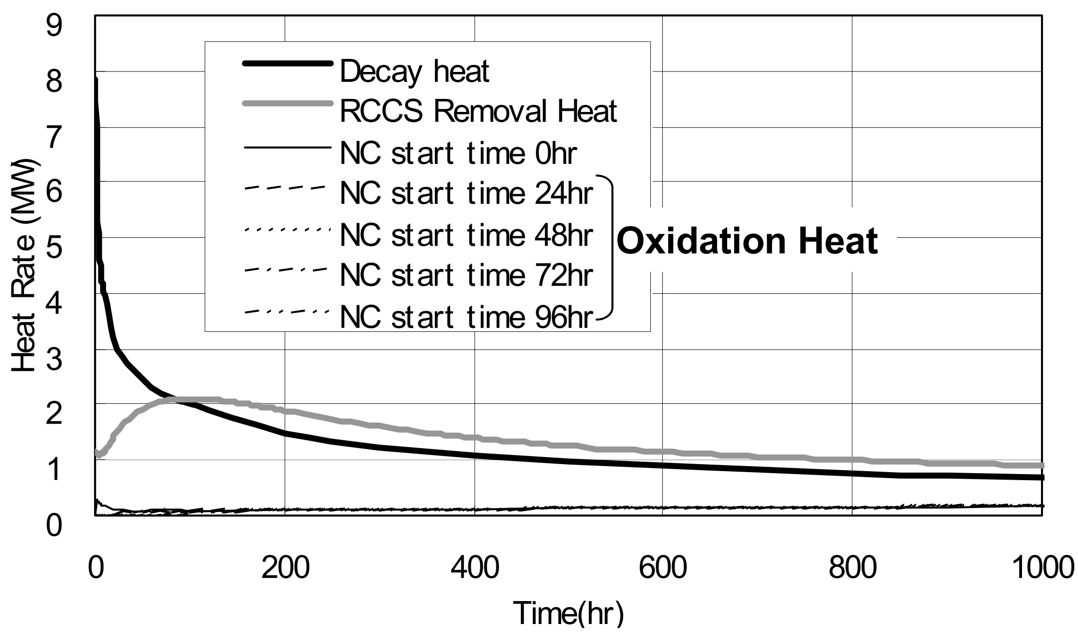


Figure 5-18. Heat Rates during Air Ingress Event

5.3.3 Assessments Performed by KAERI

As part of this study, KAERI performed an independent assessment of the cross-vessel rupture event and prepared a report titled "Air-Ingress Analysis for NGNP Reactor". This report is included as Appendix D. Key results from this report are summarized below.

<u>Analytical Model and Calculational Procedure</u>

Calculations were performed using the GAMMA+ code. The GAMMA+ model is shown in Fig. 5-19 and is based on the model used previously by KAERI to assess cooled-vessel design options [Richards 2008]. The system model consists of the reactor coolant system, the reactor cavity and the RCCS, the Shutdown Cooling System (SCS) heat exchanger, and the RB. Two- or three-dimensional heat transfer is modeled in all solid regions. Fluid regions are modeled using a combination of two- and one-dimensional flow networks. In particular the reactor cavity and the annulus between the core barrel and the RPV are modeled in two dimensions in order to simulate local flow circulation phenomena. Thermal radiation heat transfer is modeled in the top plenum, the annulus between the core barrel and the RPV, the reactor cavity containing the RCCS panels, and the annulus between the downcomer wall and the reactor cavity wall. The

RB is modeled as single fluid volume. A one-dimensional model is used for the air-cooled RCCS, with ambient air outside the RB assumed to be at 1 bar and 43°C.

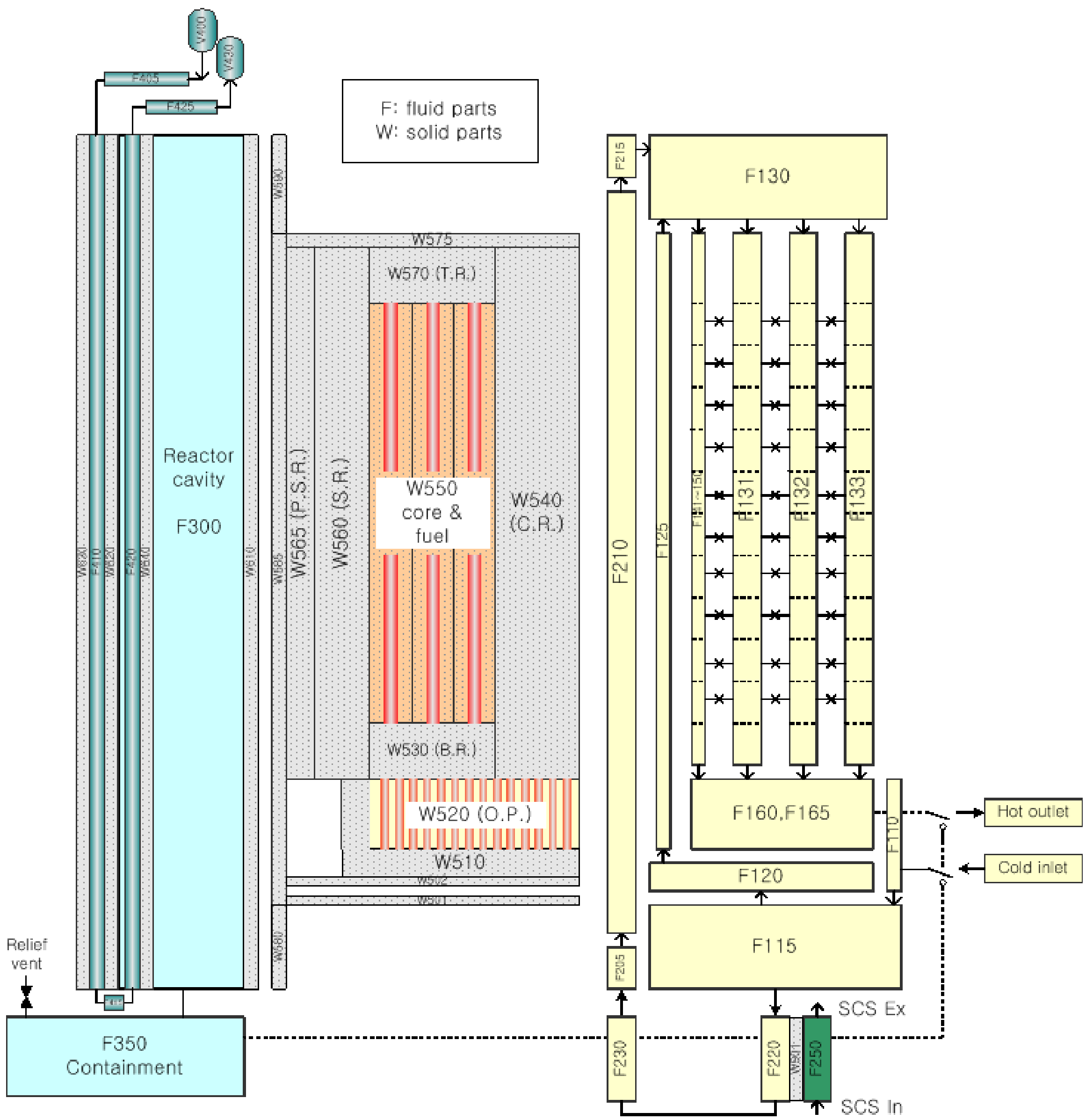


Figure 5-19. GAMMA+ Model used to Simulate Air Ingress Event

As part of their assessments, KAERI developed models to estimate the rate of air supply into the RB. These models are described according to the five cases below:

Case A: The guillotine rupture area is exposed to a continuous supply of air. Airflow through the RPV is determined from a balance of buoyancy and friction forces. The oxygen mass fraction in the RB is assumed to be the same as that in ambient air.

Case B: The RB atmosphere is refreshed by a pressure-gradient driven flow through a rupture area corresponding to breakage of a primary coolant pressure relief line (82 cm^2). The pressure gradient is assumed to be 1 psid, resulting in an air supply rate of 1.02 kg/s.

Case C: Same as Case B, except the leakage area is assumed to be 6.6 cm^2, resulting in an air supply rate of 0.082 kg/s.

Case D: The air supply rate to the RB is assumed to be controlled by diffusion/natural convection. The RB vent dampers are assumed to have failed in the open state, resulting in an area of approximately 10 m^2 exposed to the ambient atmosphere.

Case E: Same as Case D, except the area exposed to the ambient atmosphere is assumed to correspond to breakage of a primary coolant pressure relief line (82 cm^2).

KAERI used the GAMMA+ code to estimate the time for onset of natural convection airflow, assuming the reactor internals design shown in Fig. 5-20. This design incorporates a Vessel Cooling System (VCS) to maintain RPV temperatures below 350°C during normal operation. This VCS utilizes the SCS to provide a small flow of cold helium to a flow path between the core barrel and RPV. Natural circulation of air is assumed to begin after a sufficient quantity of air has replaced helium in the upper plenum to create a buoyancy force that is large enough to overcome the frictional and form losses associated with the flow paths. Because of the large volumes associated with the upper plenum, VCS annulus, and lower head and the slow nature of the diffusion process, the GAMMA+ model predicted a significant delay (~23 d) before the onset of natural circulation. This delay requires additional confirmation,[14] but provides a contrast to the assessments performed by FES (see previous section), which treated the time for onset of natural circulation as a parameter and assumed relatively short delay periods ranging from no delay to 4 d. A long delay time for the onset of natural circulation provides time for operator actions to mitigate air ingress. As shown in the previous section, short delay times result in much of the oxygen being consumed in lower graphite structures before the airflow reaches the active core. However, if the air ingress cannot be mitigated by operator actions during a long delay period, the lower graphite structures will undergo a significant cooldown during the delay, which will allow oxygen-rich air to reach a larger portion of the active core. The level of oxidation in the active core will depend on the time-temperature history of the core in relation to the delay before natural circulation begins. As discussed in the following section, if the delay period can be extended for very long times, even the active core will become sufficiently cool to limit graphite oxidation to very small levels.

[14] The time for onset of natural circulation should be confirmed using more detailed computational fluid dynamics (CFD) models and comparison of models with experimental data. Some recent air ingress benchmarking studies are described in [Kadak 2006].

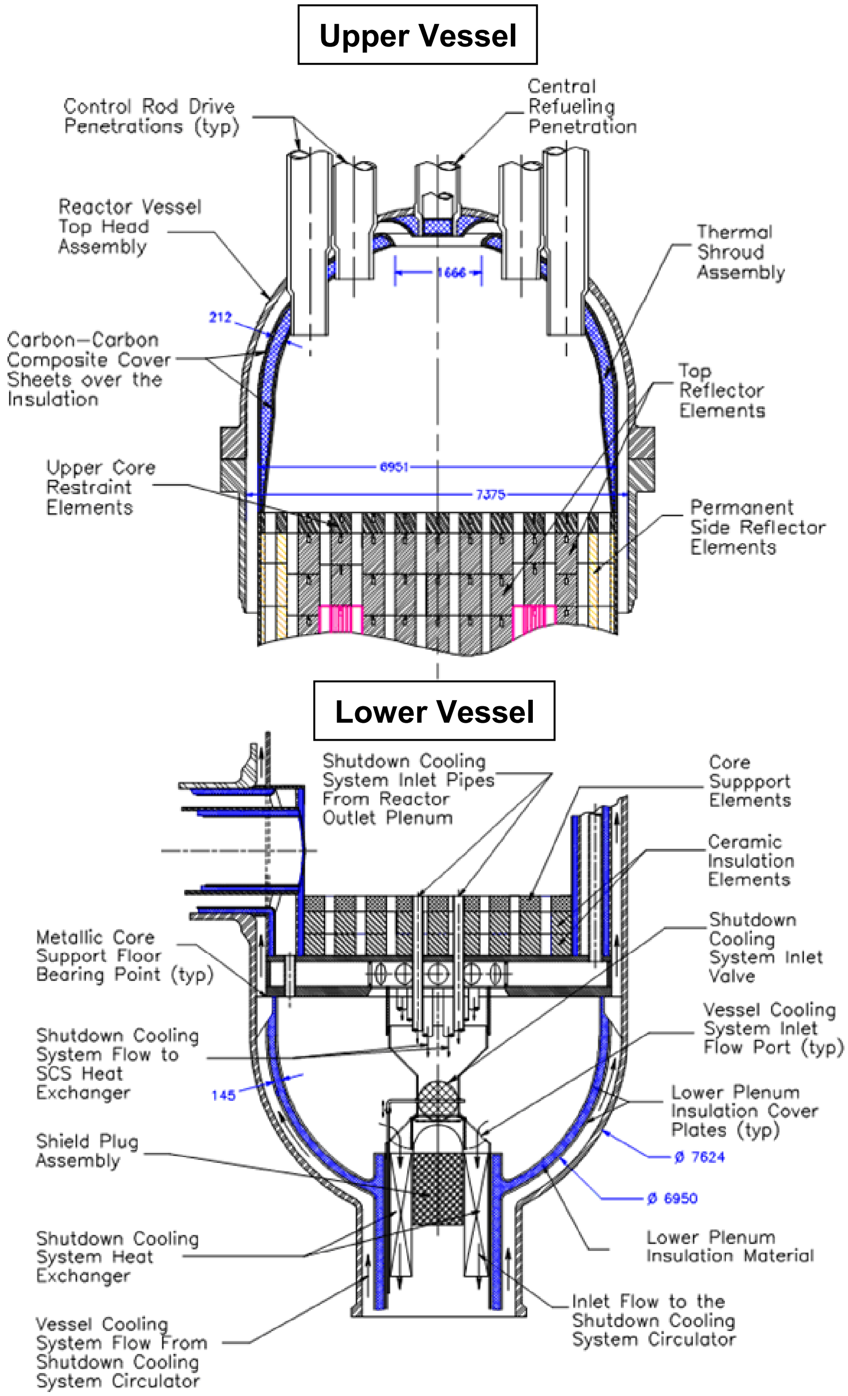


Figure 5-20. Reactor Internals Configuration Used for Air Ingress Simulation

<u>Summary of Results</u>

Figure 5-21 shows the predicted mass flow rate of air into the RPV for the five cases described above. For Cases A through D, the predicted flow rates are in the range 0.15 to 0.2 kg/s (540 to

720 kg/h). For Case E, the limited flow area assumed for the RB results in a significantly lower supply rate of air into the RB.[15] These flow rates are somewhat higher than those predicted by FES (see Fig. 5-12). However, because of the long delay time before onset of natural circulation, graphite temperatures are lower, which results in lower viscosity of the gas mixture and reduced frictional forces compared to the cases analyzed by FES. Hence, the higher flow rates predicted by KAERI are not unexpected. The predicted oxygen mass fractions in the RB for the five cases are shown in Fig. 5-22. The oxygen concentration in the RB decreases as the supply rate of air to the RB decreases.

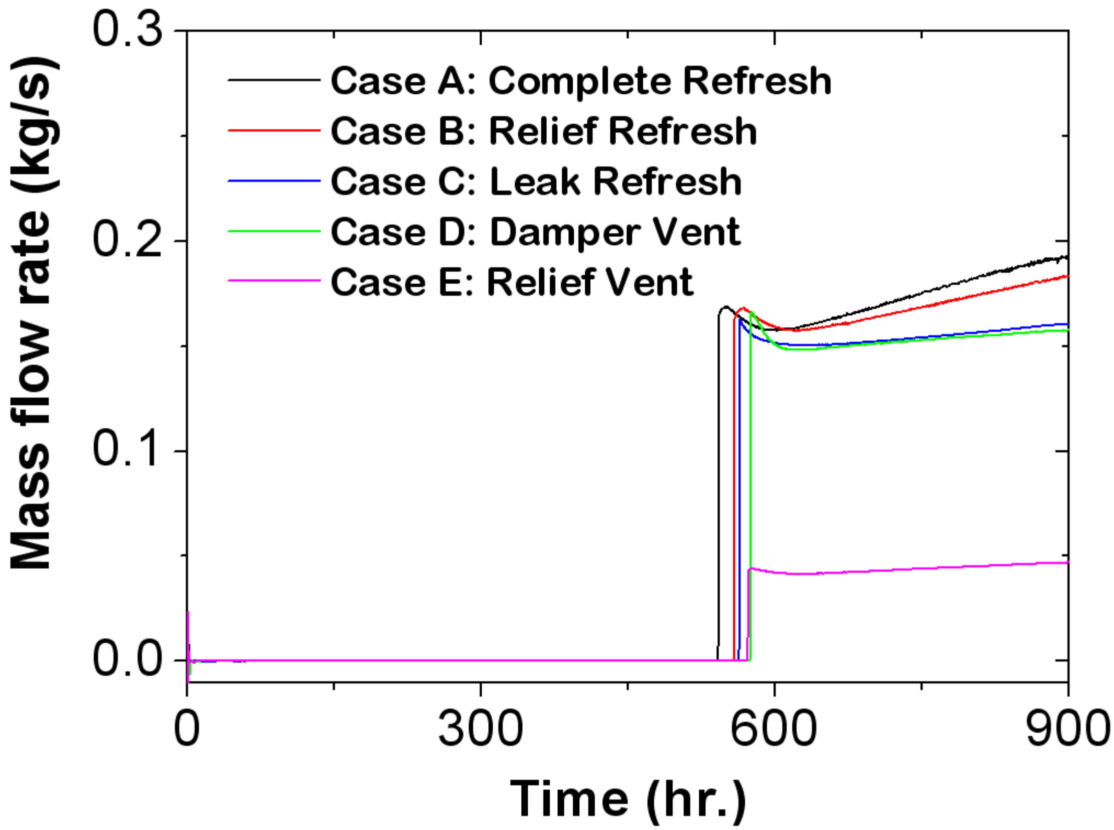


Figure 5-21. Predicted Air Mass Flow Rates into RPV during Air Ingress Event

[15] This case would correspond to an RB that remained very leak tight following a cross-vessel rupture event, which is probably not very likely.

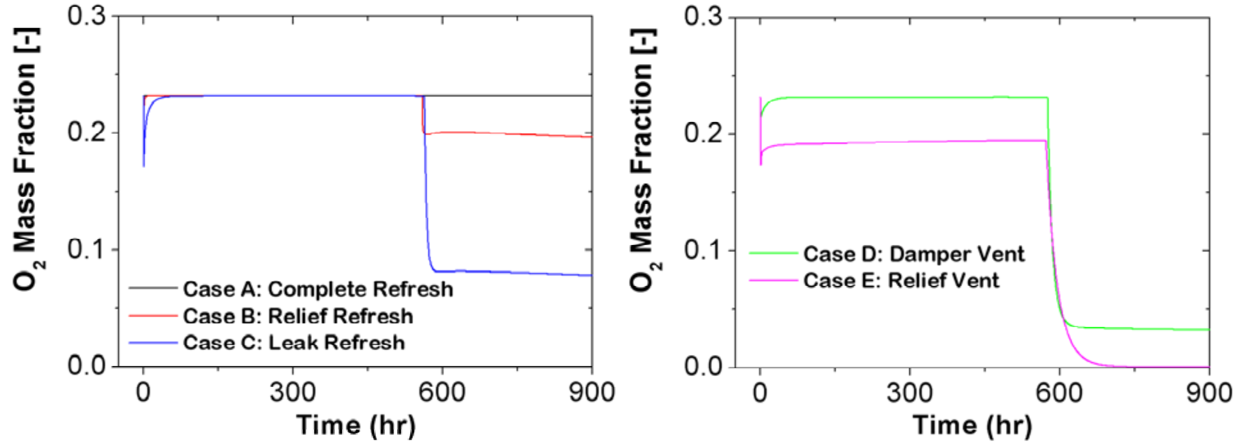


Figure 5-22. Predicted Oxygen Concentration in RB during Air Ingress Event

Figure 5-23 shows the transient temperature response for the five cases. Temperature excursions occur after the onset of airflow, but peak fuel temperatures remain well below 1600°C.

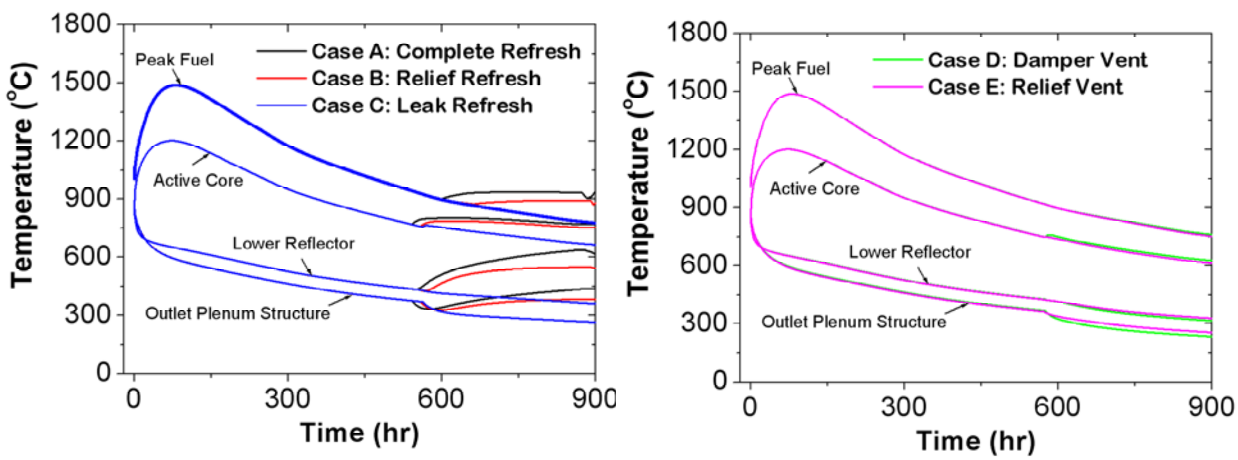


Figure 5-23. Transient Temperature Response during Air Ingress Event

Predictions for graphite oxidation after 900 hours of simulation time are given in Table 5-7. For the worst-case scenario (Case A), the total graphite oxidation is approximately 3%. Approximately 11% of the graphite in the active core is predicted to oxidize. As shown in Fig. 5-24, most of the graphite oxidation is predicted to occur in the lower graphite structures and bottom-most layer of the active core, which is consistent with the assessments performed by FES (see Section 5.3.2).

Table 5-7. Predicted Graphite Oxidation at 900 h

Region	Initial Graphite Volume (m^3)	Percent Oxidized				
		Case A	Case B	Case C	Case D	Case E
Active core	72.3	1.10E+01	1.06E+01	4.90E+00	2.20E+00	2.30E-01
Outlet plenum	20.4	7.61E-01	2.79E-01	6.27E-02	5.59E-02	3.12E-02
Bottom reflector	12.9	1.88E+01	6.90E+00	2.43E-01	9.12E-02	3.16E-02
Central reflector	75.2	2.29E-02	2.17E-02	1.10E-02	5.66E-03	7.38E-04
Side reflector	110.4	6.53E-02	5.34E-02	1.66E-02	6.77E-03	6.79E-04
Permanent side reflector	43.3	2.68E-03	2.77E-03	2.83E-03	2.79E-03	1.19E-03
Top reflector	14.5	8.47E-05	8.72E-05	6.64E-05	6.63E-05	4.15E-05
Total	348.9	3.04E+00	2.49E+00	1.04E+00	4.66E-01	5.11E-02

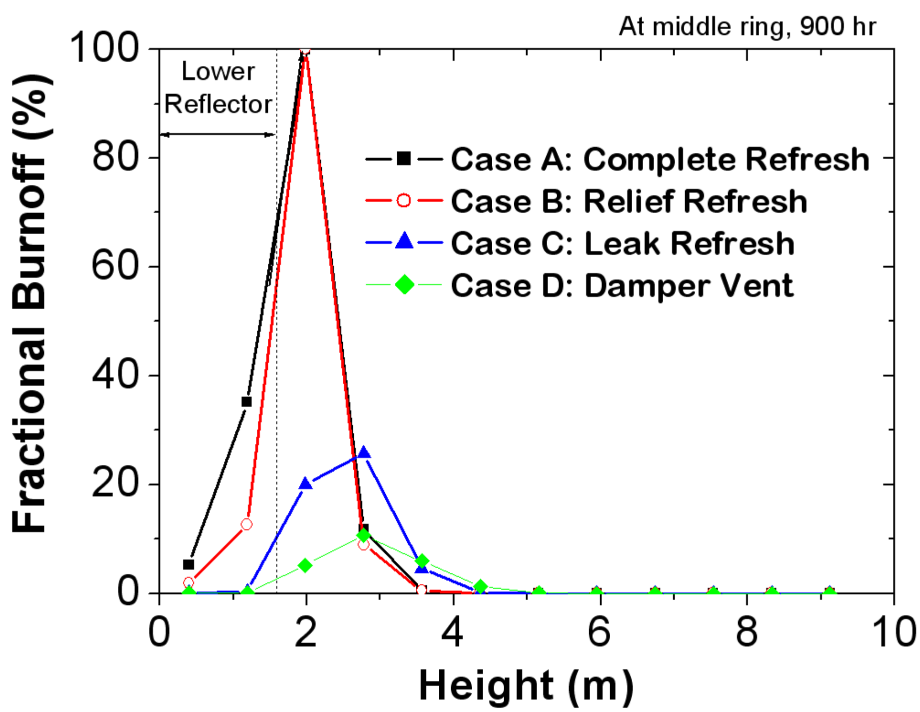


Figure 5-24. Axial Distribution of Local Graphite Burnoff at 900 h

5.3.4 Design Options to Mitigate Air Ingress

As discussed in the previous section, even if the onset of natural convection of air is delayed by approximately 500 h because of natural processes, temperatures remain sufficiently high such that non-negligible levels of graphite oxidation can occur in the active core and lower graphite structures. Although cross-vessel rupture and other events leading to massive air ingress are beyond design basis events, with graphite oxidation being a minor contributor to the radiological consequences and overall physical damage, it may be desirable to incorporate passive design measures into the VHTR design to mitigate air ingress and address potential regulatory concerns. As described in [Yan 2008], JAEA has developed a concept referred to as Sustained Counter Air Diffusion (SCAD) that can potentially delay the onset of natural circulation of air for very long time periods on the order of 2000 h. If the delay for onset of airflow can be extended to very long time periods, all of the graphite structures, including the active core, will cool to temperatures that result in negligible levels of graphite oxidation.

The SCAD concept is based on injecting a small flow of helium from a pressurized source that counters the diffusion of air. In practice, the pressurized He source could be a tank that is connected to the upper portion of the RPV with a small penetration. The tank would be maintained at normal reactor operating pressure. During a rapid depressurization event, a rupture disk would burst, causing He to flow from the tank into the RPV, with an orifice used to regulate the flow rate to a low level, but at a level sufficiently high to counter air diffusion. Of course, for the system to be effective, it would have to be designed to survive the initiating event. Figure 5-25 shows a CFD model for a 600 MWt VHTR with a SCAD port for He injection in the upper plenum region. As shown in Fig. 5-26, a constant SCAD He injection rate of 0.14 kg/h can effectively mitigate the onset of natural circulation of air. As shown in Fig. 5-27, only approximately 300 kg of injected He is required to prevent the onset of natural circulation of air for over 2000 h.

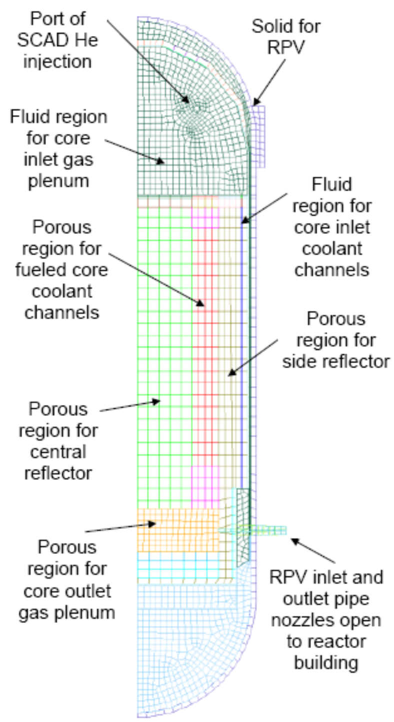


Figure 5-25. CFD Model for 600 MWt VHTR with SCAD Injection Port (figure courtesy of JAEA)

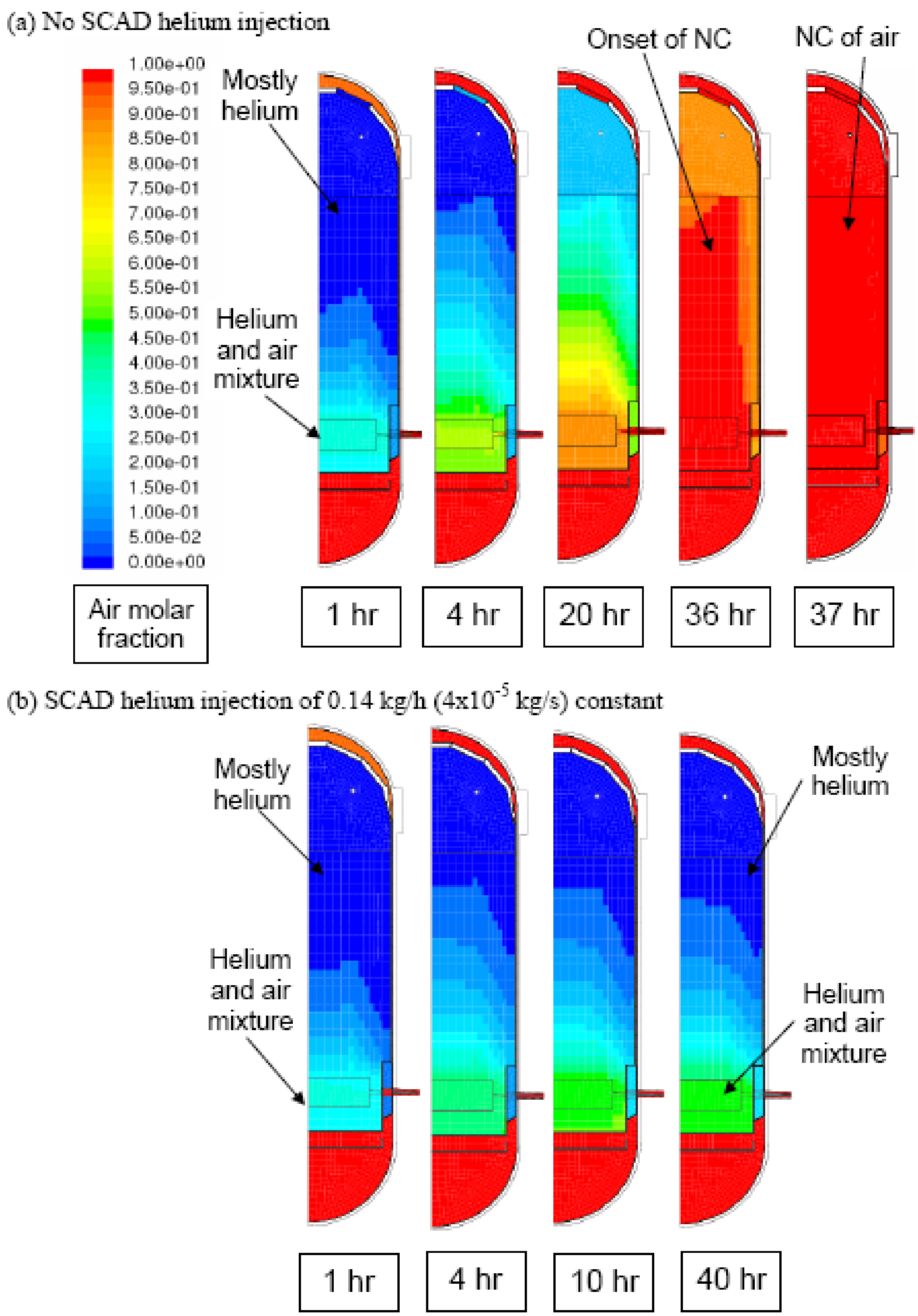


Figure 5-26. Effect of SCAD He Injection on Mitigating Air Ingress (figure courtesy of JAEA)

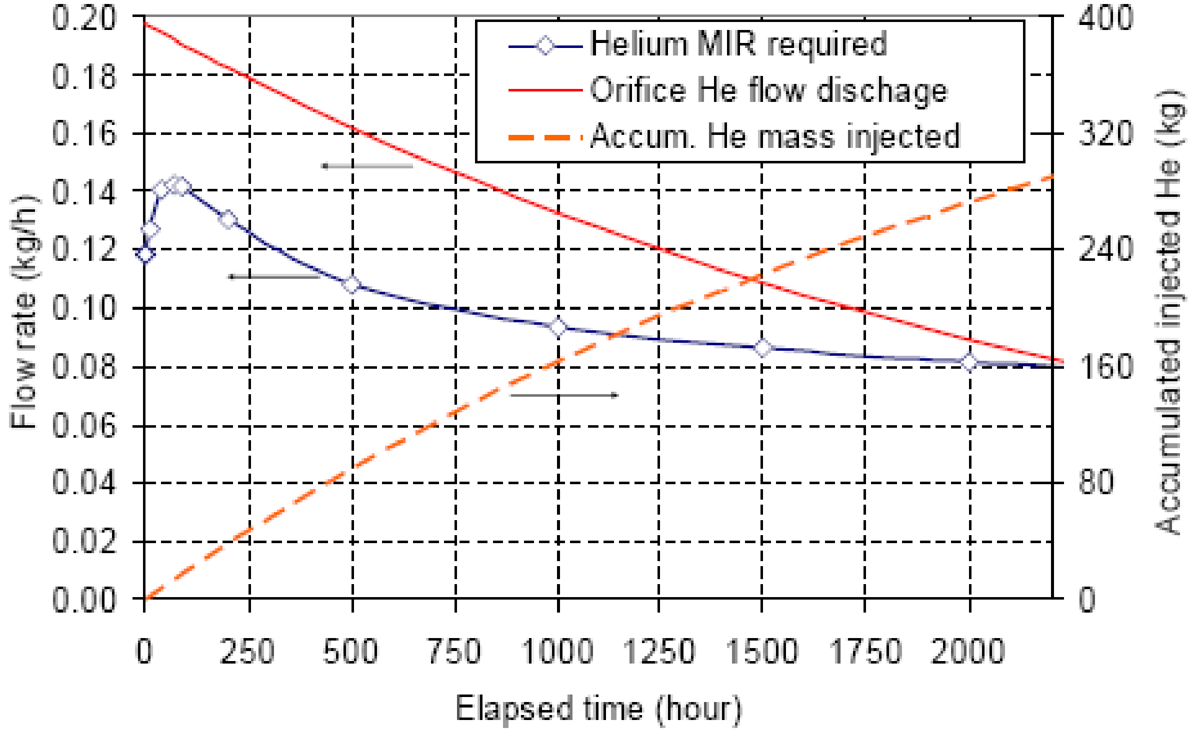


Figure 5-27. SCAD He Injection Requirement to Mitigate Air Ingress (figure courtesy of JAEA)

6. REACTOR BUILDING DESIGN ALTERNATIVES

6.1 Summary of Previous Studies for Steam-Cycle MHTGRs

6.1.1 MHTGR Containment Study

Containment alternatives were evaluated as part of a containment study for the MHTGR [Bechtel 1989]. This study considered the following alternatives:

1. Vented, 100% per day leak rate (reference design)
2. Vented, filtered, 100% per day leak rate
3. Vented, filtered, 5% per day leak rate
4. Unvented, moderate pressure, 5% per day leak rate
    a. Air-cooled RCCS
    b. Water-cooled RCCS
5. Unvented, high pressure
    a. 5% per day leak rate
    b. 1% per day leak rate

Advantages and disadvantages of these alternatives are summarized in Table 6-1. This study concluded that the reference design (Alternative 1) would meet all safety requirements with large margin and was the only option that was likely to meet economic goals.

6.1.2 MHTGR Cost Reduction Study

Containment alternatives were also evaluated as part of a cost reduction study report performed for the 450 MWt MHTGR [Bechtel 1990]. Results of this study are summarized below.

<u>Embedment Alternatives</u>

This study considered two different embedment depths: (1) a fully embedded reactor complex (operating floor at plant grade) and (2) a partially embedded RB that placed the operating floor at 28 ft elevation. The partially embedded option reduced costs by about $3 million; however the increased seismic loads, reduced flexibility on the maintenance floor and uncertainties of impacts on other building costs offset the small cost reduction.

Table 6-1. Summary of Advantages and Disadvantages for Alternative RB Concepts

RB Alternative	Advantages	Disadvantages	Estimated Cost Increase (1989 $M)
1	Provides acceptable level of public safety. Maximizes reliance on simple, passive features. Least complex Design.	Depends on successful completion of technology programs for fuel fabrication, fuel performance, fission product behavior, physics, graphite, thermal performance, metals, and major components.	Baseline
2	Provides acceptable level of public safety.	Increases cost, may not meet economic goal. Increases radionuclide containment system complexity. Depends on successful completion of essentially the same technology development program as for Alternative 1.	5.3
3	Provides acceptable level of public safety.	Adds substantial cost, not likely to meet economic goal. Increases radionuclide containment system complexity.	15.3
4	Provides acceptable level of public safety, including mitigation of hypothetical fuel failure accidents, in the event the passive safety design assumptions are not validated. Reduces reliance on technology development program.	Prohibitive cost, reduced plant availability, does not meet economic goal. Substantial increase in complexity of radionuclide containment system requires active systems. Decreases reliability of long term heat removal. Reduces module interdependence. Introduces enhanced radionuclide transport mechanisms for low frequency events for Alternative 4a.	38.8 (4A) 32.6 (4B)
5	Same as 4.	Prohibitive cost, reduced plant availability, does not meet economic goal. Substantial increase in complexity of radionuclide containment system requires active systems. Decreases reliability of long term heat removal. Introduces enhanced radionuclide transport mechanisms for low frequency events for Alternative 4a.	89.4

Reactor Complex Cost Drivers

The following design considerations were identified in Section 6.8.1 of [Bechtel 1990] as cost drivers for the Reactor Complex:

- Excavation and freeze wall
- Concrete below elevation (-)30 ft
- Special shielding
- Concrete above elevation (-)30 ft
- Construction of a non-safety grade RB
- Impact of increased reactor power

The report concluded that excavation and water control cost will vary depending on site specific conditions.

All concrete thickness below grade is dependent on shielding requirements. The shielding requirements have not been fully defined for routine access. An estimated savings of $3.5 million is reported in [Bechtel 1990] if shielding in certain areas is not required.

Special shielding materials (steel and polysiloxane) are used in certain areas of the plant. Based on the expected fuel quality, a decrease in shielding requirements is expected, resulting in about a $1.6 million cost reduction. Shielding design and cost reduction efforts should consider current shielding technologies (e.g., using boroflex). The concrete shielding around the RCCS is based in part on neutron transport calculations that were considered to be conservative in the previous MHTGR studies.

Based on a significantly reduced post-accident source term compared to LWRs, a cost reduction option identified in [Bechtel 1990] is to procure commodities and verify construction as a standard industrial concrete structure; e.g. ACI-318, and design the structure as a Seismic I structure per ACI-349. This would permit a reduction in inspections and material testing. As a result, a $16 million direct cost reduction was estimated.

This study also considered additional alternatives for radiological source term mitigation, including filtration, increasing the RB stack height, and extending the EAB. These alternatives were also considered as part of a follow-on study [Dilling 1993] and are discussed in the following sections.

6.1.3 450 MWt MHTGR Source Term and Containment Report

In response to a NRC request to evaluate alternative source terms for the MHTGR, containment alternatives were evaluated with respect to their impacts on radiological consequences during accidents [Dilling 1993]. Specifically, the NRC requested that source terms be evaluated that considered (1) lower quality fuel with higher defect fractions, (2) very rapid hydrolysis of defective fuel during water-ingress events, (3) “weak” fuel, i.e., fuel that performs as expected during normal operation but fails at higher rates than expected during accidents, and (4) higher than expected release of plateout activity during accidents. The RB alternatives considered as part of this study are described below:

VLPC Alternatives

A. Reference VLPC design. The vent path is designed to open at 1 psid and the RB can withstand a 10 psid internal pressure transient load. The source term from fission products that leak into the RB are reduced by plateout and deposition before release to the environment via the vent path. The RB has a design leak rate of 100%/day. Building leakage and the vent path discharge are considered ground level releases and the EAB is 425 m.

B. This alternative adds a system to collect pressure relief discharges and collect them onto a simple filter located outside the RB. It also includes a second building vent path which goes to another simple filter. This filtered pathway will relieve the pressure from slow depressurizations such that the large building vent does not open. The decontamination factors (DFs) for these filters are 10.

C. This alternative is the same as Alternative B, except that more efficient filters are used for both filtered pathways.

D. This alternative is the same as Alternative C, except the EAB is increased from 425 m to 805 m (1/2 mile).

E. This alternative adds a tall stack to Alternative D, such that releases from both filtered pathways can be considered as elevated releases. The stack is assumed to be 90 m above ground level, which is 3 times the height of the tallest building. Discharges from building leaks are considered to be ground level releases.

Low-Leakage, High-Pressure Containment Alternatives

F. This alternative is a conventional, high-pressure containment structure based on the design evaluated as part of the MHTGR Containment Study [Bechtel 1989]. The internal design pressure is on the order of 55 psig and alternative (possibly active) decay heat removal and containment cooling systems would be required. The EAB is assumed to be 425 m and the containment leak rate is assumed to be 5%/day.

G. This alternative is the same as Alternative F, except the containment leak rate is assumed to be 0.5%/day and the EAB is assumed to be 805 m.

These alternatives were evaluated with respect to their impacts on radiological consequences for the SRDC-6 (water-ingress) event, the SRDC-10 (rapid depressurization) event, and the SRDC-11 (slow depressurization) event. As discussed in section 5.2.3, SRDC-6 results in the largest radionuclide release from the RB, primarily because of the additional source term from fuel hydrolysis. Table 6-2 shows a comparison of these alternatives and their relative impacts on 30-d thyroid and whole-body doses at the EAB for the SRDC-6 event.

Table 6-2. Comparison of RB Alternatives

	Alternative						
Design Parameter	**A**	**B**	**C**	**D**	**E**	**F**	**G**
Filter on pressure relief line	No	Yes	Yes	Yes	Yes	No	No
DF for Iodine	1	10	100	100	100	N/A	N/A
Increase EAB from 425 m to 805 m	No	No	No	Yes	Yes	No	Yes
Elevate release stack to 90 m	No	No	No	No	Yes	No	No
RB leak rate (%/d)	100	100	100	100	100	5	0.5
Estimated incremental cost (1993 $M) (Baseline cost is ~$1,400M)	Base	8	16	28	50	132	144
Impact on 30-d Doses at EAB for SRDC-6 Event							
Thyroid dose reduction factor	1	0.20	2.0×10^{-2}	1.5×10^{-2}	9.6×10^{-4}	1.2×10^{-3}	6.5×10^{-5}
Whole-body dose reduction factor	1	0.32	0.19	0.14	9.1×10^{-3}	7.0×10^{-3}	2.6×10^{-4}

As indicated in Table 6-2, inclusion of a filter on the pressure relief line can significantly reduce the thyroid dose and inclusion of an elevated release stack can significantly reduce both thyroid and whole-body dose. For the conditions evaluated as part of this study, increasing the EAB from 425 m to 805 m provided only a modest reduction in thyroid and whole-body dose.

6.2 VLPC Filtration Options

As discussed in the previous section, options for VLPC filtration include a filter on the vent pathway from the RB and on the primary helium pressure relief line. These design options are shown schematically in Fig. 6-1. Additional capital costs for filtration systems are expected to range from $5 million to $50 million, depending on the design approaches and requirements. These cost estimates are based on historical experience with installations of vented filtered containment features at government and commercial facilities.

6.2.1 Filtered Release Pathway from the RB

For this option, the filter capacity, in terms of the size of blowdown to be accommodated, is a parameter that can be selected, and a set of constraints on the filter and other building aspects (vent pressure setpoint and leak rate) can then be determined. The head loss through the filter must be less than the building vent pressure setpoint.[16] The head loss is a function of the flow, and as the blowdown proceeds, the flow rate and head loss through the filter will gradually decrease until the building returns to atmospheric pressure. The physical and chemical properties of the blowdown gases, particular the temperature, place limitations on the type of filter medium that can be used. Sand or gravel and packed glass fibers both meet the requirement to withstand expected temperatures (~425°C) and to be suitably inert over a long standby period.

[16] For a filter on the RB vent path to be effective, the RB design pressure will likely have to be increased from 1 psid to approximately 5 psid to prevent bypass of the filter during a blowdown.

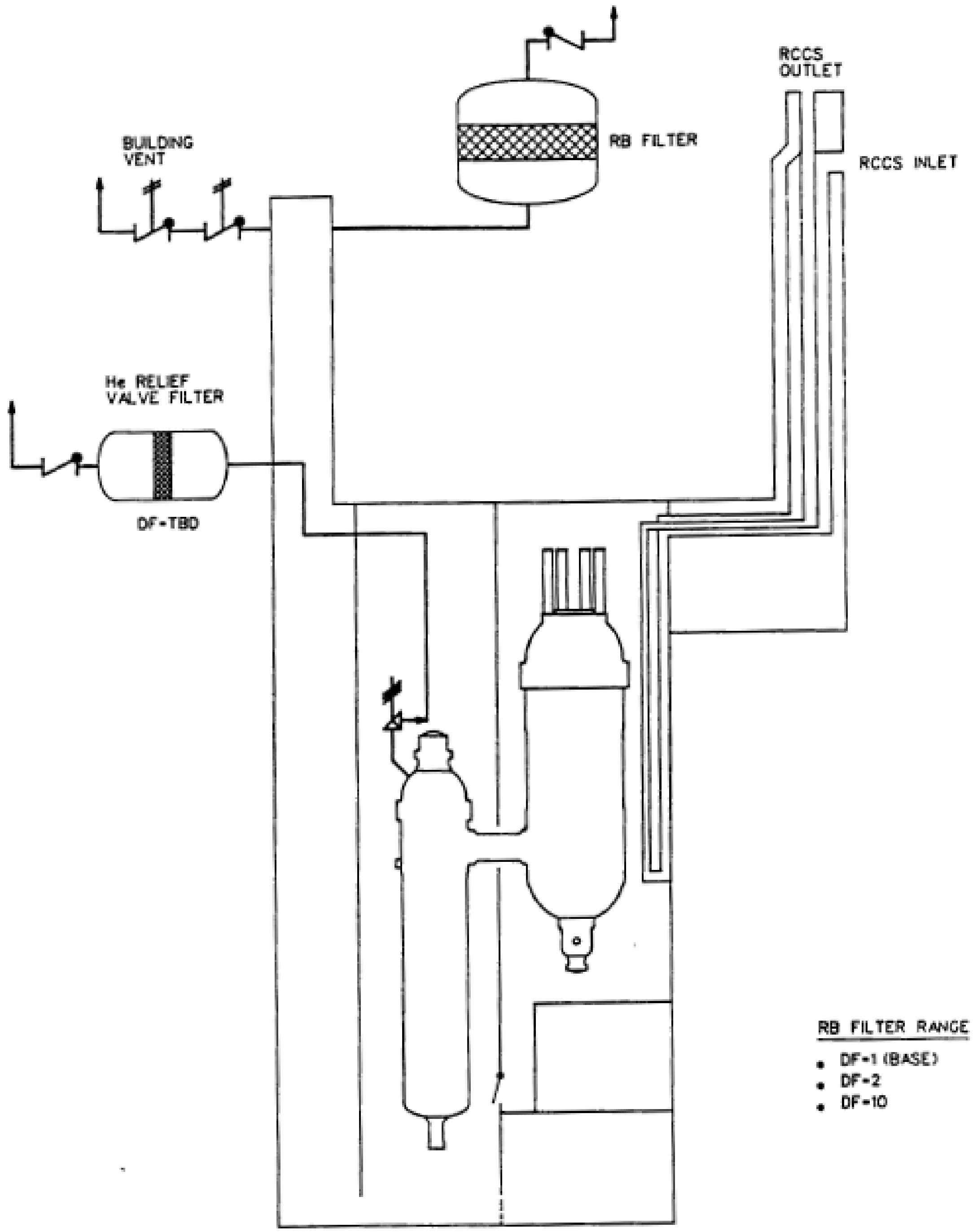


Figure 6-1. Reactor Building Filter Train Design Options

A candidate filter is specified by a limited set of parameters, specifically the filter medium, bed depth, and frontal area. The performance of the filter will vary with the flow rate, which is determined by the break size. The DF of the filter increases as the bed depth increases, and is inversely proportional to the flow velocity through the bed. The pressure loss through the filter is proportional to flow velocity. As discussed in [Dilling 1993], a bed consisting of packed glass fibers is usually more efficient in terms of head loss with depth, but sand may be superior in terms of DF per unit pressure loss. The physical location of the VLPC filter can be within the RB (at or below grade) or outdoors. Each VLPC would have its own filter arrangement. A low pressure rupture disk at the filter outlet can be used as a means of excluding moisture and foreign matter during standby conditions. Candidate filter designs are evaluated in [Dilling 1993].

During normal operation and during events for which the building vent pressure setpoint is not exceeded, leakage across the RB boundary to the environment will bypass the filter. The flow through cracks and small openings will vary in the same manner as the filter, as a function of the pressure gradient from the inside of the RB to the environment and the flow coefficient for the opening. Over the range of flow rates expected for radiologically important events, the flow coefficients for the filter and for the building leaks can be assumed to be constant. Therefore, the fraction of flow which bypasses the filter is constant regardless of the variation in pressure over the entire blowdown event. This bypass fraction can be controlled by selecting the filter characteristics or by altering the architectural features in the RB to reduce the leak rate. In practice, it is probably easier to control the bypass fraction by increasing filter area.

RB filtration options identified by URS-WD include:

1. HEPA/Charcoal: Effective in addressing the source term, with up to 99% iodine removal, except for any bypass allowance.
2. Dry Sand/Gravel Bed with downstream HEPA/Charcoal: Dry bed may serve to protect HEPA/charcoal bed from pressure or other adverse upstream conditions, if any.
3. Wet Sand/Gravel Bed with downstream HEPA/Charcoal: Wet bed could contain sodium hydroxide and sodium thiosulfate in solution to enhance iodine removal, possibly eliminating the need for downstream filtration.
4. Combination of a venturi scrubber with a pool containing sodium hydroxide and sodium thiosulfate solution and a packed filtration bed integrated into a single unit. This system provides iodine removal, but the effectiveness may be reduced at low flow rates. Downstream HEPA/charcoal filtration may be needed to achieve acceptable iodine DFs. Scrubbing through submerged discharge into a pool may prove to be more predictable and effective than packed bed design options.

Final selection of a filtration option will depend on factors such as iodine chemical form, and physical filter protection from the effects of anticipated accident scenarios such as pressure, shock, flow, temperature, and particulate loadings. HEPA/charcoal is probably the most cost-effective option, but requires protection from high-pressure and high-moisture events. This issue is one justification for directing pressure relief valve blowdown to a separate filter compartment [GA 1993]. Although design of vented filtration systems for a VHTR VLPC presents some challenges, the designs are less demanding than those for an LWR containment because of the much lower water inventories during hypothetical VHTR accident scenarios.

6.2.2 Filter on the Primary Helium Pressure Relief Line

This design option is shown schematically in Fig. 6-2. The physical location for the filter is not constrained, although it may be necessary to surround the filter with radiation shielding and to protect it from external hazards. It is desirable to locate the filter discharge outside of the RB and to connect the relief valves to the filter discharge with piping that penetrates the RB boundary. An exterior discharge will reduce hazards to personnel and also preclude contamination of the RB and need for subsequent cleanup. It is expected that a shielded filter room would be required, which would contribute some additional cost to the RB.

The piping network which connects the relief valve trains to the filter can be connected to a common header which connects to the filter discharge line. The pressure relief lines must be sized such that maximum allowable head loss requirements are not exceeded. As shown in Fig. 6-2, the 450 MWt MHTGR design included three pressure relief lines. The low-pressure rupture disk shown on the outlet of the filter vessel is proposed as a means of excluding moisture and foreign material from the filter during standby conditions. Candidate filter design parameters are given in Table 6-3.

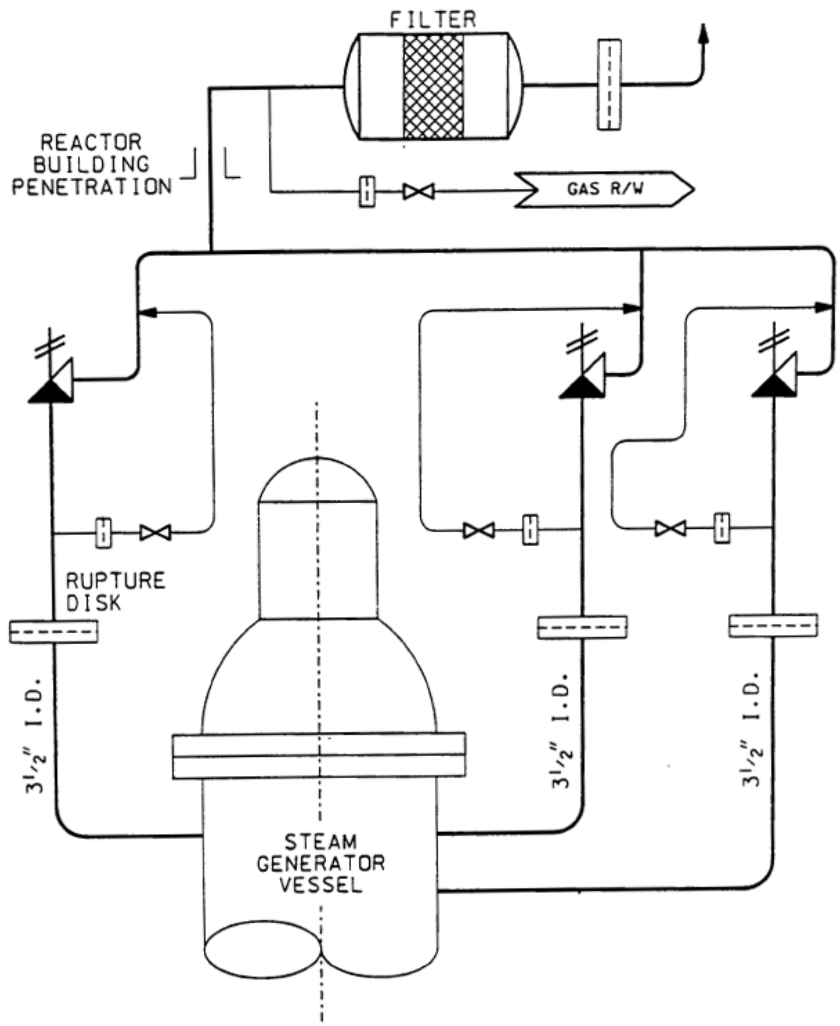


Figure 6-2. Concept for Filter on Primary Coolant Pressure Relief Train (450 MWt MHTGR)

Table 6-3. Candidate Design Parameters for Filter on Pressure Relief Line

Filter medium	Glass Fiber	Glass Fiber	Sand/Gravel	Sand/Gravel	Sand/Gravel
Bed characteristics	11 lb/ft^3	10 lb/ft^3	Granule size 0.04 in.	Granule size 0.06 in.	Granule size 0.024 in.
Bed depth (ft)	35	40	8.7	7.7	5
Frontal area (ft^2)	19.6	78.5	4200	1400	4200
Max. head loss during blowdown (psid)	50	23	122	288	88
Vessel diameter (ft)	5	10	73	43	73
Vessel length (ft)	45	45	19	18	15
Decontamination Factors					
Noble Gases	1	1	1	1	1
Halogens	4	100	100	10	1000
Particulates	1,500	4,000	>100	>10	>1000

The halogen DFs given in Table 6-3 assume that formation of organic iodine is limited. If test data show that formation of organic iodine is expected under prototypical VHTR accident conditions, addition of charcoal as a filter medium may be required to achieve acceptable DFs for iodine.

6.2.3 Issues that Affect Filtration Requirements

A final determination of VLPC performance and filtration requirements will need to be based on 600 MWt design specific analyses, modeling a full spectrum of isotopic sources, and radioactivity transport timing commensurate with standard time periods used for breathing rates and χ/Q determinations. The licensing basis analyses should consider the following issues:

- Fuel quality and other NRC issues as discussed in [PSER 1996]
- ASME code considerations
- Control room habitability
- Atmospheric Dispersion Factors

Fuel quality and other issues raised by the NRC are discussed in [PSER 1996] and [Dilling 1993]. For this study, it is assumed the Radionuclide Design Criteria given in Section 2.3 are applicable.

<u>ASME Code Considerations for Relief Valve Discharge</u>

ASME Boiler and Pressure Vessel Code Section III, Division 1, § NB-7141 [17] applies to installation of pressure relief devices for Class 1 systems; it prohibits discharge piping backpressure from reducing the relief capacity below that required to provide overpressure protection. ASME III NB-7143 addresses draining of pressure relief devices, and requires drains to be provided for pressure relief devices that are designed to accommodate collection of liquid or residue on the discharge side of the disk, with additional provisions for thermal discharge and collection of drainage. ASME III requirements are codified (with additional conditions imposed by the NRC) via 10CFR50.55a for LWRs, and should be considered as licensing requirements for NGNP. Primary vessel pressure relief system design would therefore require consideration of the following:

- Characterization of discharge flow with respect to potential moisture and particulate content (regardless of whether downstream filtration is installed), and demonstration of operability under design basis conditions. The HTR-10 pebble-bed HTGR installed a particulate (dust) filter upstream of the safety relief valve to avoid damage to seating surfaces and reduce occupational exposures during valve maintenance and testing [Dong 2007]. If similar provisions are considered for NGNP to protect the valve, then the

[17] The 2007 edition of the ASME Code is used for the purposes of this discussion. System design should be based on the selected Code of Record for NGNP.

general pressure relief device criteria of ASME III NB-7141, including open and unobstructed flow between the system and relief device, must be addressed.

- Effect of filtration on backpressure should be minimized such that it does not adversely affect pressure relief capability.
- If a rupture disk is used in conjunction with a relief valve, then the rupture disk may only be installed downstream of the valve.[18]
- The set pressure of the valve, stamped burst pressure of the rupture disk, and outlet piping pressure must all be considered in determining the pressure relief system capability (ASME III, NB-7600).
- Provisions for effective removal of moisture and particulates from the valve seating surfaces must be included in the discharge line.

Control Room Habitability

Experience with LWRs, especially in the current environment of control room habitability concerns described in NRC Generic Letter 2003-01, is that design basis radiological dose considerations are generally determined by the control room considerations. A major thrust for these concerns by the NRC are recent tracer gas test results on unfiltered control room leakage measurements that were orders of magnitude in excess of expectations. Based on these tracer tests, the assumed design basis accident releases from the containment systems, whether filtered or not at the containment release location, would arrive at the control room (after application of the appropriate χ/Q values) at levels that would be more consistent with a control room that did not have dedicated safety-related filtration systems. Although the plant level requirements defined in [SRM 2007] Criterion 3.1.3.5 indicate that "the plant design shall require no reliance on the operator, the control room and its contents, or any AC-Powered equipment to satisfy the NRC design basis accident limits/requirements," control room habitability remains a major NRC issue based on current advanced LWR experience. Additionally, NRC criteria per NUREG-0737 would likely require accident progression evaluation capabilities at a Technical Support Center with habitability considerations similar to a control room, and a habitable Emergency Operations Facility. These plant features are assumed to be needed to support an Emergency Plan, notwithstanding the NGNP goal of establishing the LPZ at the EAB. Other considerations include air intake layout with respect to planned and potential accident release pathways, and procedures for habitability.

[18] Approval of an ASME code case would be required for the design concept shown in Fig. 6-2, which utilizes rupture disks upstream of the relief valves to provide protection from moisture and coolant impurities during normal operation. Alternatively, these protections could be incorporated into the design of the relief valve.

Elimination of Control Room habitability considerations may require the determination that the Control Room can be evacuated along with the rest of the site, and the plant monitored from a distance where χ/Q values are no worse than those at the EAB, with Control Room occupancy assumptions.

Atmospheric Dispersion Factors

[PSID 1992] used RG 1.4 to determine χ/Q values. The current methodology for calculating χ/Q values for a given site is to use the ARCON and PAVAN Codes. These codes require several years of meteorological data for a given site. Less conservative χ/Q values for a specific site should be calculated to estimate more realistic offsite doses.

6.3 Elevated VLPC Vent Stack

Offsite doses during accidents can be reduced by equipping the VHTR RB with a tall exhaust stack, as illustrated in Fig. 6-3. All radioactive exhaust streams released through the stack would be considered elevated releases, resulting in greater dilution and dispersion of the released radioactivity before it reaches the EAB. For the generic MHTGR site with a stack height of 90 m, radionuclide concentrations at an EAB of 425 m would be reduced by approximately a factor of 3 [GA 1993].[19] Exhaust streams that could be routed through the stack include the nuclear island HVAC exhaust and VLPC vent exhaust (with or without filters). If the primary coolant pressure relief line is equipped with a filter, this exhaust stream could also be directed to the stack. Radioactivity released from normal building leakage would still be considered a ground-level release.

URS-WD has performed site-specific estimates of χ/Q values for the Peach Bottom Units 2&3 site for both non-elevated and elevated releases. As indicated in Table 6-4, the χ/Q values are reduced by factors ranging from 100 to 500 for an elevated release at this site.

If a free standing, elevated vent stack is included in the NGNP design and is taken credit for in control room habitability analyses, then Regulatory Guide 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants", revision 0, June 2003 (RG 1.194), position 3.2.2 should be considered applicable, requiring:

> "Such a stack should be more than 2-1/2 times the height of the adjacent structures or be located more than 5L downwind of the trailing edge of upwind buildings, and more than 2L upwind of the leading edge of downwind buildings, and more than 0.5L crosswind of the closest edge of crosswind buildings where L is the lesser of the height or width of the building creating the downwind, upwind,

[19] The overall effect of increased stack height on radiological consequences should be based on a site-specific, time-dependent release.

or crosswind wake. Since L will be dependent on wind direction for most building clusters, it will generally be necessary to assess the zone of influence for all directions within the 90 degree wind direction sector centered on the line of sight between the stack and the control room intake."

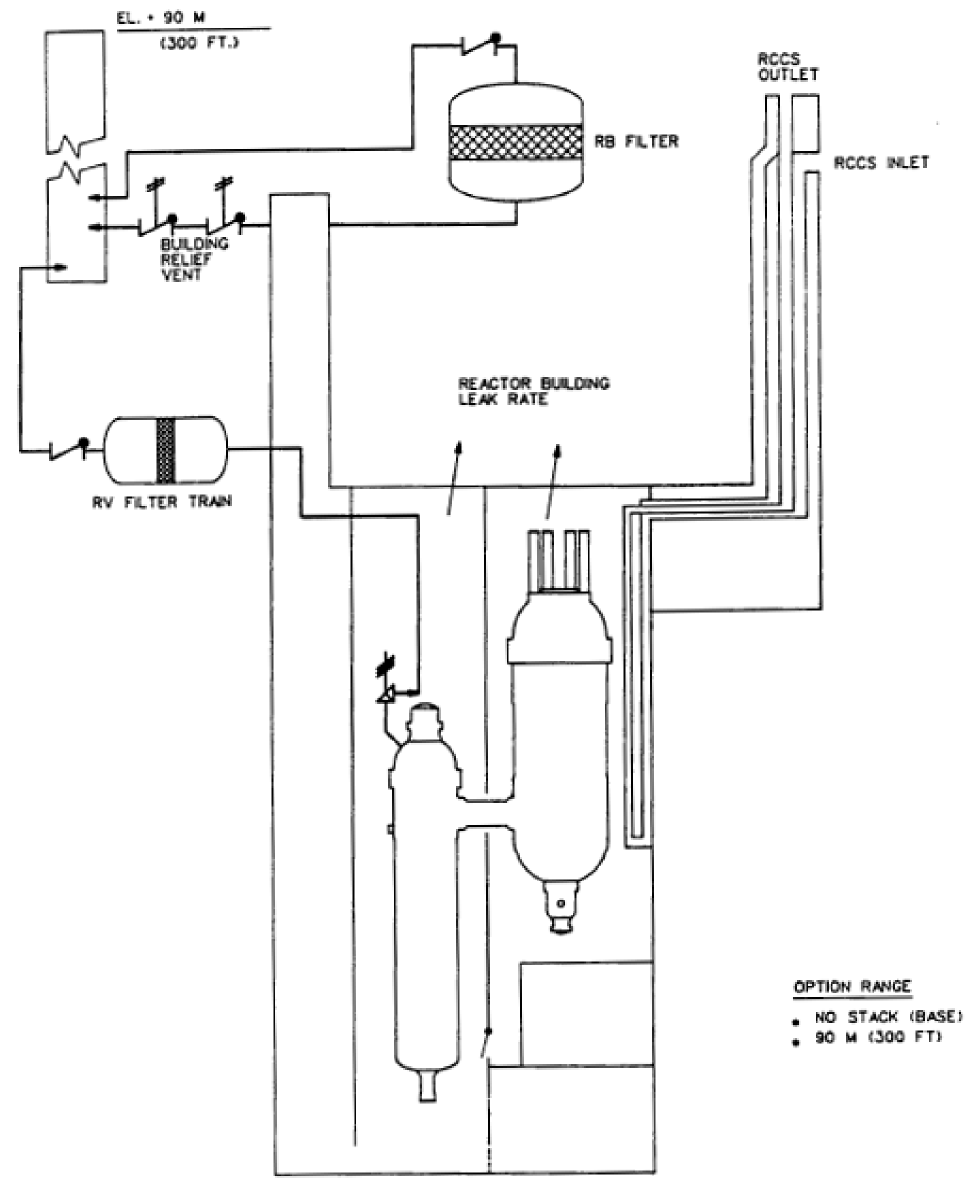


Figure 6-3. Reactor Building with Elevated Vent Path

Table 6-4. Effect of Elevated Release on Atmospheric Dispersion

	MHTGR PSID Ground Release	Peach Bottom Ground Release	Peach Bottom Elevated Release
EAB (m)	425	823	823
χ/Q (s/m^3)			
0 – 2 hr		9.11×10^{-4}	9.17×10^{-6}
0 – 8 hr	1.21×10^{-3}	4.67×10^{-4}	3.24×10^{-6}
8 – 24 hr	6.34×10^{-4}	3.35×10^{-4}	1.92×10^{-6}
1 – 4 d	2.30×10^{-4}	1.64×10^{-4}	6.22×10^{-7}
4 – 30 d	5.22×10^{-5}	6.26×10^{-5}	1.23×10^{-7}

Notes
1. Building wake correction factors are accounted for.
2. Elevated stack credit requires consideration of a 0.5 h fumigation period where dispersion factors are considerably higher.
3. Stack location for Peach Bottom likely maximizes credit for elevated release compared with other sites.

6.4 Extension of the Exclusion Area Boundary

An increase in the EAB may be a cost-effective means of reducing offsite doses, particularly if an increase in EAB size is warranted by other user requirements (e.g., security considerations). NGNP top level design criteria in [SRM 2007] include establishing the EAB and Emergency Planning Zone (EPZ) boundaries at 425 meters from the reactor. The EAB and EPZ sizes may be increased (requiring SRM revision) to achieve dose reduction.

Extending the EAB from 425 m to 805 m (1/2 mile) was evaluated as part of the studies described in [Dilling 1993] and [GA 1993]. Figure 6-4 illustrates the extended EAB concept. URS-WD has performed site-specific estimates of χ/Q values using RG 1.4 methodology for both a 425 m EAB and an 805 m EAB. As indicated in Table 6-5, the χ/Q values are reduced by factors ranging from 2 to 30 for an EAB extended to 805 m.

Depending on the size of the EAB, the hydrogen facility may be located "offsite" with respect to the nuclear plant and would be subject to public dose limits. The normal operational dose limits to the public are regulated by Appendix I of 10CFR50 and 40CFR190, and the occupational dose limits are regulated by 10CFR 20.1201. Table 6-6 shows the limits for the public at the EAB and occupational limits. The occupational dose limit for the NGNP is 500 mrem (10% of 5 rem).

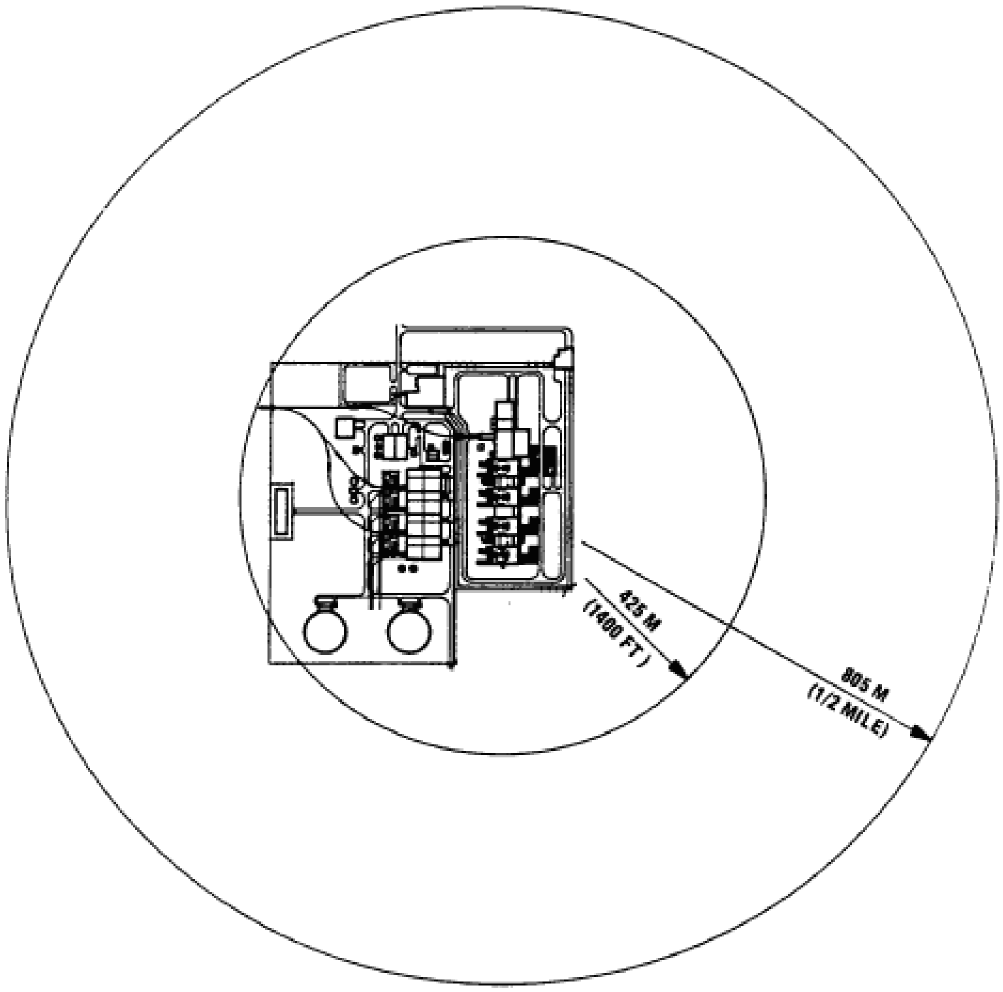


Figure 6-4. Concept for Extension of Exclusion Area Boundary

Table 6-5. Effect of EAB on Atmospheric Dispersion

	EAB	
	425 m	**805 m**
Time Period	χ/Q (s/m^3)	
0 – 8 hr	1.30×10^{-3}	7.03×10^{-4}
8 – 24 hr	7.00×10^{-4}	2.15×10^{-4}
1 – 4 d	2.30×10^{-4}	7.60×10^{-5}
4 – 30 d	5.60×10^{-4}	1.75×10^{-5}

Notes
1. Building wake correction factor for 425 m EAB is 2.1.
2. Building wake correction factor for 805 m EAB is 1.35
3. Elevated stack credit requires consideration of a 0.5 h fumigation period where dispersion factors are considerably higher.

Table 6-6. EAB Occupational and Public Dose Limits

Dose Limits to Employees

	TEDE[2]	Thyroid
occupational dose limits 20.1201	5 rem/yr	N/A

Dose Limits to Members of the Public

	Whole body	Thyroid
PAG accident limits (EPA-520/1-75-001)	1 rem(accident)	5 rem/accident
10CFR50.67 limits	25 rem(accident) (TEDE)[2]	N/A
40CFR190 limits	25 mrem/yr	75 mrem/yr
Appendix I dose limits[1]	10 mrem/yr gamma gaseous effluents / 20 mrem/yr beta gaseous effluents and 3 mrem/yr liquid effluents	10 mrem/yr liquid effluents
10CFR20.1301 limits	100 mrem/yr (TEDE)[2]	N/A

Notes: 1. Appendix I for effluents does not include shine dose. 40CFR190 does include shine dose.
2. TEDE is defined in RG 1.183 as the whole body dose + (0.03)*(the thyroid dose). TEDE is used for alternate source term calculations.

10CFR835.602(a) applies to DOE facilities and may be a licensing requirement for the first-of-a-kind (FOAK) NGNP, requiring an access point to each radiologically controlled area. Individuals who enter only controlled areas without entering radiological areas or radioactive material areas are not expected to receive a total effective dose equivalent greater than 0.1 rem in a year.

As shown in the above table, the design for accident scenarios is controlled by the PAG limits and normal operational doses for offsite exposure are controlled by 40CFR190. Occupational dose limits will be controlled by either 10CFR20.1201 (with the NGNP-imposed 10% limit) or 10CFR835.

It is recommended to include the hydrogen production facility as part of the reactor site, which allows for a larger margin in dose limits for the hydrogen production workers. However, additional training for these workers may be required. The type and level of training is expected to be similar to that given to workers at a nuclear facility that do not have access to radiation control areas.

A greater distance to the EAB/EPZ boundary would be viewed favorably from a regulatory standpoint, particularly for the EPZ because of the significant difference between the SRM criterion (defined by user/utility requirements) and current practice for commercial LWRs. 10CFR50, Appendix E, includes the following:

> "The size of the EPZs for a nuclear power plant shall be determined in relation to local emergency response needs and capabilities as they are affected by such conditions as demography, topography, land characteristics, access routes, and jurisdictional boundaries. The size of the EPZs also may be determined on a case-by-case basis for gas cooled nuclear reactors and for reactors with an authorized power level less than 250 MW thermal. Generally, the plume exposure pathway EPZ for nuclear power plants with an authorized power level greater than 250 MW thermal shall consist of an area about 10 miles (16 km) in radius and the ingestion pathway EPZ shall consist of an area about 50 miles (80 km) in radius."

Modifications to [SRM 2007] should be considered to replace the prescribed EAB and EPZ distance with a performance-based or user requirement.

7. CONCLUSIONS AND RECOMMENDATIONS

Based on this study and studies performed for previous MHTGR concepts, the VLPC concept is recommended for the NGNP RB design. Based on this study and studies performed for previous MHTGR concepts, the VLPC concept is recommended for the NGNP RB design. The different effects that embedment of the NGNP RB can have on the design, construction, maintenance and operation of the plant are evaluated and discussed considering three possible alternatives where the RB is either fully embedded, partially embedded, or partially embedded with backfill. The most economical solution for the RB design is dependant on site specific conditions such as depth of rock, seismic conditions and elevation of water table. It is recommended that site investigations be performed at potential candidate sites so the RB embedment can be designed based on site-specific conditions. A greater embedment depth reduces the overall height of the RB above grade, which is driven by requirements for refueling equipment. Deeper embedment also provides greater protection against natural hazards and external threats. Design considerations for deeper embedment include the following:

- Design measures to protect the embedded portion of the RB against external flooding may include providing a water barrier on all exterior concrete members subjected to ground water via a combination of installing a waterstop and applying waterproofing chemicals to the concrete.
- Deeper embedment poses challenges in pipe tunnel design and pipe routing, particularly for main steam and feedwater lines. A pipe tunnel would assure protection and accessibility of the penetration area and piping outside the RB. The depth of a pipe tunnel is dictated by the elevation of the penetrations. Greater embedment of a pipe tunnel increases the overburden pressures and excavation costs.
- The magnitude of wind, tornado, and external hazard loads as compared with the magnitude of dynamic soil pressure loads (i.e., these loads are essentially a trade-off with respect to embedment depth).

VLPC design alternatives that can reduce doses at the EAB include filtered pathways on the RB and on the primary coolant pressure relief line, an elevated stack, and an extension of the EAB. Neither the capital costs, nor the O&M implications of these VLPC design alternatives are expected to have a significant negative impact on the operability or cost of the NGNP.

If the NGNP includes a SG in the primary loop (as assumed for this study), accidents involving water-ingress are expected to result in the most severe radiological consequences. However, the doses at the EAB for these events are expected to be below the EPA PAG limits and the radiological consequences can be further mitigated if one or more of the alternatives identified above are incorporated into the VLPC design. In particular, a filtered pathway on the primary coolant pressure relief line offers several advantages with only modest cost implications. In addition to significantly reducing radionuclide release to the environment during water ingress events, this design option can also improve worker safety by eliminating the possibility of discharge into the RB.

Regulatory and Licensability Issues and Recommendations

The RB is expected to be licensed under a regulatory framework using current NRC regulations and guidance modified as needed to account for unique design features of the NGNP, consistent with NRC policy to achieve "enhanced margins of safety and/or simplified, inherent, passive or other means to accomplish safety and security functions." Defense-in-depth (D-i-D) remains a governing principle that guides NRC development of advanced reactor licensing policy. Issues and recommendations related to licensing include the following:

- Resolution of remaining issues in the NRC Preliminary Safety Evaluation Report [PSER 1996].
- Resolution of issues regarding Regulatory Treatment of Non-Safety Systems (RTNSS) should consider:
  - Regulatory criteria for safety classification for NGNP may be essentially the same as that applied to current reactors (i.e., 10CFR50 Appendix A, General Design Criterion 1). RB functions of equipment support and protection, and radionuclide control warrant a safety-related, Seismic Category I classification.
  - Active systems to back up the passive safety functions and provide investment protection should be examined with a full scope Probabilistic Risk Assessment (PRA) and considered for special regulatory treatment (e.g., assigned reliability and availability goals). The RB HVAC system is a potential candidate for RTNSS that could help address licensability issues associated with reliance on the RCCS.
- For an embedded RB, potential issues related to regulatory requirements for inspection and condition monitoring of the foundation should be investigated further.

Key Issues Requiring Additional R&D and Engineering Studies

Because of the previous severe accidents at the Chernobyl and Windscale reactors, demonstrating the safety case for beyond design basis events with air ingress may become an important issue for the NGNP. As part of this study, independent assessments of a cross-vessel rupture event were performed by FES and KAERI. Both of these assessments show that air ingress does not affect peak fuel temperatures reached during the accident and has a relatively small impact on the overall temperature response of the core during the accident. The total amount of graphite oxidation is limited to a few percent and is confined to the lower graphite structures and bottom-most layer of the active core. For these reasons, the incremental radiological consequences associated with air ingress and graphite oxidation should be small compared to heatup of the core, which is largely driven by decay heat. However, this event should continue to be analyzed in increasing detail, including more detailed modeling of oxidation in the lower graphite structures and assessments of the impacts of oxidation on structural integrity. Design measures to mitigate air ingress, including the SCAD concept developed by JAEA, should be evaluated in more detail.

If the NGNP includes a SG in the primary loop, more detailed assessments must be performed for the SRDC-6 water-ingress event, including assessing the effects of higher core temperatures on fuel hydrolysis and flammable gas generation.[20] The more recent data obtained on fuel hydrolysis should be analyzed in more detail and used to develop improved models for fuel hydrolysis. A preliminary evaluation of this data [Richards 1990] indicated the hydrolysis reaction rate may become more limited at high water vapor pressures than is currently assumed in existing models.

As the NGNP progresses into the conceptual design phase, optimization studies should be performed for the VLPC, including selection of design pressure, leak rate, embedment depth, and filtration requirements.

Accurate models for VLPC environment/atmosphere and surface temperatures during normal operation and accidents should be developed. The VLPC atmosphere is the boundary condition for determining cross vessel and SG (or IHX) vessel temperatures, which are limited to about 350°C during normal operation if SA-508/533 steel is to be used for these components. The VLPC atmosphere and surface conditions/temperatures are important parameters for determining the rates for plateout and deposition of radionuclides in the VLPC. In particular, it is important to characterize the behavior of radiological important isotopes of iodine in the VLPC during accident conditions. A VLPC model is important for both modeling these events and for defining the conditions for technology development programs for obtaining iodine transport data.

As shown in Fig. 2-1, the configuration assumed for this study includes a SG in the primary loop that produces steam at 540°C with a reactor helium outlet temperature in the range 900°C to 950°C. Although the SG tube temperatures are controlled primarily by the water/steam temperatures in the tubes, a tritium migration assessment should be performed if these point design conditions are adopted. There may also be additional design and operational issues for a SG with these design points that should be assessed.

[20] For this study, the NGNP design point for the primary helium outlet temperature is assumed to be in the range 900°C to 950°C. Previous analyses of the SRDC-6 event were based on MHTGR designs operating with coolant outlet temperatures of approximately 700°C.

8. REFERENCES

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[10CFR] NRC Regulations Title 10, Code of Federal Regulations

- 10 CFR 20, Standards for Protection against Radiation
- 10 CFR 50, Domestic Licensing of Production and Utilization Facilities
- Appendix A to Part 50--General Design Criteria for Nuclear Power Plants
- Appendix B to Part 50--Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants
- Appendix E to Part 50--Emergency Planning and Preparedness for Production and Utilization Facilities
- Appendix G to Part 50--Fracture Toughness Requirements
- Appendix H to Part 50--Reactor Vessel Material Surveillance Program Requirements
- Appendix I to Part 50--Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion "As Low as is Reasonably Achievable" for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents
- Appendix J to Part 50--Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors
- Appendix N to Part 50—Standardization of Nuclear Power Plant Designs: Permits To Construct and Licenses To Operate Nuclear Power Reactors of Identical Design at Multiple Sites
- Appendix Q to Part 50--Pre-application Early Review of Site Suitability Issues
- Appendix R to Part 50--Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979
- Appendix S to Part 50--Earthquake Engineering Criteria for Nuclear Power Plants
- 10 CFR 51, Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions
- 10 CFR 52, Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants
- 10 CFR 73, Physical Protection of Plants and Materials
- 10 CFR 100, Reactor Site Criteria

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[SRP 2.5]	NUREG-0800 §§: 2.5.1, Rev. 4 Basic Geologic and Seismic Information 03/2007 2.5.2, Rev. 4 Vibratory Ground Motion 03/2007 2.5.3, Rev. 4 Surface Faulting 03/2007 2.5.4, Rev. 3 Stability of Subsurface Materials and Foundations 03/2007 2.5.5, Rev. 3 Stability of Slopes 03/2007
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[SRP 3.4]	NUREG-0800 §§ 3.4.1, Rev. 3 Internal Flood Protection for Onsite Equipment Failures 03/2007 3.4.2, Rev. 3 Analysis Procedures 03/2007

[SRP 3.5]	NUREG-0800 §§ 3.5.1.1, Rev. 3 Internally Generated Missiles (Outside Containment) 03/2007 3.5.1.2, Rev. 3 Internally Generated Missiles (Inside Containment) 03/2007 3.5.1.3, Rev. 3 Turbine Missiles 03/2007 3.5.1.4, Rev. 3 Missiles Generated by Tornadoes and Extreme Winds 03/2007 3.5.1.5, Rev. 4 Site Proximity Missiles (Except Aircraft) 03/2007 3.5.1.6, Rev. 3 Aircraft Hazards 03/2007 3.5.2, Rev. 3 Structures, Systems, and Components To Be Protected From Externally-Generated Missiles 03/2007 3.5.3, Rev. 3 Barrier Design Procedures 03/2007
[SRP 3.6]	NUREG-0800 §§ 3.6.1, Rev. 3 Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment 03/2007 3.6.2, Rev 2 Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping 03/2007 3.6.3, Rev. 1 Leak-Before-Break Evaluation Procedures 03/2007
[SRP 3.7]	NUREG-0800 §§ 3.7.1, Rev. 3 Seismic Design Parameters 03/2007 3.7.2, Rev. 3 Seismic System Analysis 03/2007 3.7.3, Rev. 3 Seismic Subsystem Analysis 03/2007 3.7.4, Rev. 2 Seismic Instrumentation 03/2007
[SRP 3.8]	NUREG-0800 §§ 3.8.1, Rev. 2 Concrete Containment 03/2007 3.8.2, Rev. 2 Steel Containment 03/2007 3.8.3, Rev. 2 Concrete and Steel Internal Structures of Steel or Concrete Containments 03/2007 3.8.4, Rev. 2 Other Seismic Category I Structures 03/2007 3.8.5, Rev. 2 Foundations 03/2007
[SRP 3.11]	NUREG-0800 §3.11, Rev. 3 Environmental Qualification of Mechanical and Electrical Equipment 03/2007
[SRP 6.2]	NUREG-0800 §§ 6.2.1, Rev. 3 Containment Functional Design 03/2007 6.2.1.2, Rev. 3 Subcompartment Analysis 03/2007 6.2.2, Rev. 5 Containment Heat Removal Systems 03/2007 6.2.3, Rev. 3 Secondary Containment Functional Design 03/2007 6.2.4, Rev. 3 Containment Isolation System 03/2007 6.2.5, Rev. 3 Combustible Gas Control in Containment 03/2007 6.2.6, Rev. 3 Containment Leakage Testing 03/2007 6.2.7, Rev. 1 Fracture Prevention of Containment Pressure Boundary 03/2007

[SRP 6.3]	NUREG-0800 §6.3, Rev. 3 Emergency Core Cooling System 03/2007
[SRP 8.2]	NUREG-0800 §8.2, Rev. 4 Offsite Power System 03/2007
[SRP 8.3]	NUREG-0800 §§ 8.3.1, Rev. 3 AC Power Systems (Onsite) 03/2007 8.3.2, Rev. 3 DC Power Systems (Onsite) 03/2007
[SRP 9.1]	NUREG-0800 §§ 9.1.1, Rev. 3 Criticality Safety of Fresh and Spent Fuel Storage and Handling 03/2007 9.1.2, Rev. 4 New and Spent Fuel Storage 03/2007 9.1.3, Rev. 2 Spent Fuel Pool Cooling and Cleanup System 03/2007 9.1.4, Rev. 3 Light Load Handling System (Related to Refueling) 03/2007 9.1.5, Rev. 1 Overhead Heavy Load Handling Systems 03/2007
[SRP 11.2]	NUREG-0800 §11.2, Rev. 3 Liquid Waste Management System 03/2007
[SRP 11.3]	NUREG-0800 §11.3, Rev. 3 Gaseous Waste Management System 03/2007
[SRP 11.4]	NUREG-0800 §11.4, Rev. 3 Solid Waste Management System 03/2007
[SRP 14.3]	NUREG-0800 §14.3 Inspections, Tests, Analyses, and Acceptance Criteria 03/2007
[SRP 15]	NUREG-0800 Chapter 15, Accident Analysis
[SRP 15.7]	NUREG-0800 §§ 15.7.3, Rev. 2 Postulated Radioactive Releases Due to Liquid-Containing Tank Failures (content of this section has been relocated to BTP 11-6), 07/1981 15.7.4, Rev. 2 Radiological Consequences of Fuel Handling Accidents 07/1981 15.7.5, Rev. 2 Spent Fuel Cask Drop Accidents 07/1981
[SRP 18.0]	NUREG-0800 §18.0, Rev. 2 Human Factors Engineering 03/2007
[SRP 19.0]	NUREG-0800 §19.0, Rev. 2 Probabilistic Risk Assessment and Severe Accident Evaluation

APPENDIX A – Summary of Regulations and Guidance

SUMMARY OF REGULATIONS AND GUIDANCE

Regulations

Regulation	Summary of Requirement
10 CFR Part 20	**Standards for protection against radiation**
10 CFR Part 20 Subpart A	General Provisions
10 CFR Part 20 Subpart B	**§ 20.1101 Radiation protection programs.** (a) Each licensee shall develop, document, and implement a radiation protection program commensurate with the scope and extent of licensed activities and sufficient to ensure compliance with the provisions of this part. (See § 20.2102 for recordkeeping requirements relating to these programs.) (b) The licensee shall use, to the extent practical, procedures and engineering controls based upon sound radiation protection principles to achieve occupational doses and doses to members of the public that are as low as is reasonably achievable (ALARA). (c) The licensee shall periodically (at least annually) review the radiation protection program content and implementation. (d) To implement the ALARA requirements of § 20.1101 (b), and notwithstanding the requirements in § 20.1301 of this part, a constraint on air emissions of radioactive material to the environment, excluding Radon-222 and its daughters, shall be established by licensees other than those subject to § 50.34a, such that the individual member of the public likely to receive the highest dose will not be expected to receive a total effective dose equivalent in excess of 10 mrem (0.1 mSv) per year from these emissions. If a licensee subject to this requirement exceeds this dose constraint, the licensee shall report the exceedance as provided in § 20.2203 and promptly take appropriate corrective action to ensure against recurrence.
10 CFR Part 20 Subpart C	**§ 20.1201 Occupational dose limits for adults.** (a) The licensee shall control the occupational dose to individual adults, except for planned special exposures under § 20.1206, to the following dose limits. (1) An annual limit, which is the more limiting of-- (i) The total effective dose equivalent being equal to 5 rems (0.05 Sv); or (ii) The sum of the deep-dose equivalent and the committed dose equivalent to any individual organ or tissue other than the lens of the eye being equal to 50 rems (0.5 Sv). (2) The annual limits to the lens of the eye, to the skin of the whole body, and to the skin of the extremities, which are: (i) A lens dose equivalent of 15 rems (0.15 Sv), and (ii) A shallow-dose equivalent of 50 rem (0.5 Sv) to the skin of the whole body or to the skin of any extremity.
10 CFR Part 20 Subpart D	**§ 20.1301 Dose limits for individual members of the public.** (a) Each licensee shall conduct operations so that — (1) The total effective dose equivalent to individual members of the public from the licensed operation does not exceed 0.1 rem (1 mSv) in a year, exclusive of the dose contributions from background radiation, from any administration the individual has received, from exposure to individuals administered radioactive material and released under § 35.75, from voluntary participation in medical research programs, and from the licensee's disposal of radioactive material into sanitary sewerage in accordance with § 20.2003, and (2) The dose in any unrestricted area from external sources, exclusive of the dose contributions from patients administered radioactive material and released in accordance with § 35.75, does not exceed 0.002 rem (0.02 millisievert) in any one hour. (b) If the licensee permits members of the public to have access to controlled areas, the limits for members of the public continue to apply to those individuals. (c) Notwithstanding paragraph (a)(1) of this section, a licensee may permit visitors to an individual who cannot be released, under § 35.75, to receive a radiation dose greater than 0.1 rem (1 mSv) if— (1) The radiation dose received does not exceed 0.5 rem (5 mSv); and (2) The authorized user, as defined in 10 CFR Part 35, has determined before the visit that it is appropriate.
10 CFR Part 20 Subpart E	Radiological Criteria for License Termination **§ 20.1406 Minimization of contamination.** (a) Applicants for licenses, other than early site permits and manufacturing licenses under part 52 of this chapter and

	renewals, whose applications are submitted after August 20, 1997, shall describe in the application how facility design and procedures for operation will minimize, to the extent practicable, contamination of the facility and the environment, facilitate eventual decommissioning, and minimize, to the extent practicable, the generation of radioactive waste. (b) Applicants for standard design certifications, standard design approvals, and manufacturing licenses under part 52 of this chapter, whose applications are submitted after August 20, 1997, shall describe in the application how facility design will minimize, to the extent practicable, contamination of the facility and the environment, facilitate eventual decommissioning, and minimize, to the extent practicable, the generation of radioactive waste.
10 CFR Part 20 Subpart F	Surveys and Monitoring **§ 20.1501 General.** (a) Each licensee shall make or cause to be made, surveys that-- (1) May be necessary for the licensee to comply with the regulations in this part; and (2) Are reasonable under the circumstances to evaluate-- (i) The magnitude and extent of radiation levels; and (ii) Concentrations or quantities of radioactive material; and (iii) The potential radiological hazards. (b) The licensee shall ensure that instruments and equipment used for quantitative radiation measurements (e.g., dose rate and effluent monitoring) are calibrated periodically for the radiation measured.
10 CFR Part 20 Subpart G	Control of Exposure From External Sources in Restricted Areas **§ 20.1601 Control of access to high radiation areas.** (a) The licensee shall ensure that each entrance or access point to a high radiation area has one or more of the following features-- (1) A control device that, upon entry into the area, causes the level of radiation to be reduced below that level at which an individual might receive a deep-dose equivalent of 0.1 rem (1 mSv) in 1 hour at 30 centimeters from the radiation source or from any surface that the radiation penetrates; (2) A control device that energizes a conspicuous visible or audible alarm signal so that the individual entering the high radiation area and the supervisor of the activity are made aware of the entry; or (3) Entryways that are locked, except during periods when access to the areas is required, with positive control over each individual entry. (b) In place of the controls required by paragraph (a) of this section for a high radiation area, the licensee may substitute continuous direct or electronic surveillance that is capable of preventing unauthorized entry. (c) A licensee may apply to the Commission for approval of alternative methods for controlling access to high radiation areas. (d) The licensee shall establish the controls required by paragraphs (a) and (c) of this section in a way that does not prevent individuals from leaving a high radiation area. **§ 20.1602 Control of access to very high radiation areas.** In addition to the requirements in § 20.1601, the licensee shall institute additional measures to ensure that an individual is not able to gain unauthorized or inadvertent access to areas in which radiation levels could be encountered at 500 rads (5 grays) or more in 1 hour at 1 meter from a radiation source or any surface through which the radiation penetrates.
10 CFR Part 20 Subpart H	Respiratory Protection and Controls to Restrict Internal Exposure in Restricted Areas **§ 20.1701 Use of process or other engineering controls.** The licensee shall use, to the extent practical, process or other engineering controls (*e.g.,* containment, decontamination, or ventilation) to control the concentration of radioactive material in air. [64 FR 54556, Oct. 7, 1999]

	§ 20.1702 Use of other controls. (a) When it is not practical to apply process or other engineering controls to control the concentrations of radioactive material in the air to values below those that define an airborne radioactivity area, the licensee shall, consistent with maintaining the total effective dose equivalent ALARA, increase monitoring and limit intakes by one or more of the following means-- (1) Control of access; (2) Limitation of exposure times; (3) Use of respiratory protection equipment; or (4) Other controls. (b) If the licensee performs an ALARA analysis to determine whether or not respirators should be used, the licensee may consider safety factors other than radiological factors. The licensee should also consider the impact of respirator use on workers' industrial health and safety.
10 CFR Part 20 Subpart I	Storage and Control of Licensed Material **§ 20.1801 Security of stored material.** The licensee shall secure from unauthorized removal or access licensed materials that are stored in controlled or unrestricted areas. **§ 20.1802 Control of material not in storage.** The licensee shall control and maintain constant surveillance of licensed material that is in a controlled or unrestricted area and that is not in storage.
10 CFR Part 20 Appendix B	Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage
10 CFR Part 50	**Domestic licensing of production and utilization facilities**
10 CFR Part 50 §§ 50.1 - 50.9	General Provisions
• §50.2	Definitions, including EAB, LPZ, source term, TEDE and the following definition of safety related SSCs:
	Safety-related structures, systems and components means those structures, systems and components that are relied upon to remain functional during and following design basis events to assure: (1) The integrity of the reactor coolant pressure boundary (2) The capability to shut down the reactor and maintain it in a safe shutdown condition; or (3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the applicable guideline exposures set forth in § 50.34(a)(1) or § 100.11 of this chapter, as applicable.
10 CFR Part 50 §§ 50.10 – 50.13	Requirement of License, Exceptions
10 CFR Part 50 §§ 50.20 – 50.23	Classification and Description of Licenses • § 50.22 Class 103 licenses; for commercial and industrial facilities
10 CFR Part 50 §§ 50.30 – 50.39	Applications for Licenses, Certifications, and Regulatory Approvals; Form; Contents; Ineligibility of Certain Applicants
▪ 10 CFR 50.33,	
▪ 10CFR50.33(g)	Requires applicants to submit emergency response plans for EPZ
▪ 10CFR50.34	Contents of construction permit and operating license applications; technical information
▪ 10CFR50.34a	Design objectives for equipment to control releases of radioactive material in effluents—nuclear power reactors
▪ 10CFR50.36	Technical specifications
▪ 10CFR50.36a	Technical specifications on effluents from nuclear power reactors
▪ 10CFR50.36b	Environmental conditions.
10 CFR Part 50 §§ 50.40 – 50.49	Standards for Licenses, Certifications, and Regulatory Approvals
• 10CFR50.43	Additional standards and provisions affecting class 103 licenses and certifications for commercial power 10CFR50.43(e) Applications for a design certification, combined license, manufacturing license, or operating license that propose nuclear reactor designs which differ significantly from light-water reactor designs that were licensed before 1997, or use simplified, inherent, passive, or other innovative means to accomplish their

	safety functions, will be approved only if: (1)(i) The performance of each safety feature of the design has been demonstrated through either analysis, appropriate test programs, experience, or a combination thereof; (ii) Interdependent effects among the safety features of the design are acceptable, as demonstrated by analysis, appropriate test programs, experience, or a combination thereof; and (iii) Sufficient data exist on the safety features of the design to assess the analytical tools used for safety analyses over a sufficient range of normal operating conditions, transient conditions, and specified accident sequences, including equilibrium core conditions; or (2) There has been acceptable testing of a prototype plant over a sufficient range of normal operating conditions, transient conditions, and specified accident sequences, including equilibrium core conditions. If a prototype plant is used to comply with the testing requirements, then the NRC may impose additional requirements on siting, safety features, or operational conditions for the prototype plant to protect the public and the plant staff from the possible consequences of accidents during the testing period.
• 10CFR50.44	Combustible gas control for nuclear power reactors 10CFR50.44(a)(1) *Inerted atmosphere* means a containment atmosphere with less than 4 percent oxygen by volume. 10CFR50.44 (d) *Requirements for future non water-cooled reactor applicants and licensees and certain water-cooled reactor applicants and licensees.* The requirements in this paragraph apply to all construction permits and operating licenses under this part, and to all design approvals, design certifications, combined licenses, or manufacturing licenses under part 52 of this chapter, for non water-cooled reactors and water-cooled reactors that do not fall within the description in paragraph (c), footnote 1 of this section, any of which are issued after October 16, 2003. Applications subject to this paragraph must include: (1) Information addressing whether accidents involving combustible gases are technically relevant for their design, and (2) If accidents involving combustible gases are found to be technically relevant, information (including a design-specific probabilistic risk assessment) demonstrating that the safety impacts of combustible gases during design-basis and significant beyond design-basis accidents have been addressed to ensure adequate protection of public health and safety and common

	defense and security.
• 10CFR50.47	Emergency Plans
• 10CFR50.48	Fire Protection
• 10CFR50.49	Environmental qualification of electric equipment important to safety for nuclear power plants
10 CFR Part 50 §§ 50.50 – 50.68	Issuance, Limitations, and Conditions of Licenses and Construction Permits
• 10CFR50.54	Conditions of licenses
• 10CFR50.55	Conditions of construction permits, early site permits, combined licenses, and manufacturing licenses.
• 10CFR50.55a	Codes and standards
• 10CFR50.65	Requirements for monitoring the effectiveness of maintenance at nuclear power plants
• 10CFR50.67	Accident source term *(identified as LWR-specific and N/A in NUREG-1860)*
10 CFR Part 50 §§ 50.69 – 50.76	Inspections, Records, Reports, Notifications
• 10CFR50.69	Risk-informed categorization and treatment of structures, systems and components for nuclear power reactors *(to be replaced with risk-informed, performance-based NGNP licensing framework; e.g., see NUREG-1860)*
10 CFR Part 50 §§ 50.100 – 50.103	Revocation, Suspension, Modification, Amendment of Licenses and Construction Permits, Emergency Operations by the Commission
10 CFR Part 50 § 50.109	Backfitting
10 CFR Part 50 §§ 50.110 – 50.120	Enforcement
10 CFR Part 50 Appendix A	General Design Criteria for Nuclear Power Plants
10CFR50 App. A GDCs 1 – 5	*I. Overall Requirements*
• **Definitions and Explanations**	*Nuclear power unit*. A nuclear power unit means a nuclear power reactor and associated equipment necessary for electric power generation and includes those structures, systems, and components required to provide reasonable assurance the facility can be operated without undue risk to the health and safety of the public. *Loss of coolant accidents*. Loss of coolant accidents mean those postulated accidents that result from the loss of reactor coolant at a rate in excess of the capability of the reactor coolant makeup

	system from breaks in the reactor coolant pressure boundary, up to and including a break equivalent in size to the double-ended rupture of the largest pipe of the reactor coolant system.[1] *Single failure*. A single failure means an occurrence which results in the loss of capability of a component to perform its intended safety functions. Multiple failures resulting from a single occurrence are considered to be a single failure. Fluid and electric systems are considered to be designed against an assumed single failure if neither (1) a single failure of any active component (assuming passive components function properly) nor (2) a single failure of a passive component (assuming active components function properly), results in a loss of the capability of the system to perform its safety functions.[2] *Anticipated operational occurrences*. Anticipated operational occurrences mean those conditions of normal operation which are expected to occur one or more times during the life of the nuclear power unit and include but are not limited to loss of power to all recirculation pumps, tripping of the turbine generator set, isolation of the main condenser, and loss of all offsite power.
• GDC 1	Quality Standards and Records
• GDC 2	Design Bases for Protection Against Natural Phenomena
• GDC 3	Fire Protection
• GDC 4	Environmental and Dynamic Effects Design Bases
• GDC 5	Sharing of Structures, Systems, and Components *(N/A to NGNP RB study)*
10CFR50 App. A GDCs 10 – 19	*II.* Protection by Multiple Fission Product Barriers
• GDC 10	Reactor Design
• GDC 11	Reactor inherent Protection
• GDC 12	Suppression of Reactor Power Oscillations *(N/A to NGNP RB study)*
• GDC 13	Instrumentation and Control
• GDC 14	Reactor Coolant Pressure Boundary
• GDC 15	Reactor Coolant System Design
• GDC 16	Containment Design
• GDC 17	Electric Power Systems
• GDC 18	Inspection and Testing of Electric Power Systems
• GDC 19	Control Room
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• GDC 20	Protection System Functions
• GDC 21	Protection System Reliability and Testability
• GDC 22	Protection System Independence
• GDC 23	Protection System Failure Modes

• GDC 24	Separation of Protection and Control Systems *(N/A to NGNP RB study)*
• GDC 25	Protection System Requirements for Reactivity Control Malfunctions *(N/A to NGNP RB study)*
• GDC 26	Reactivity Control System Redundancy and Capability *(N/A to NGNP RB study)*
• GDC 27	Combined Reactivity Control Systems Capability *(N/A to NGNP RB study)*
• GDC 28	Reactivity Limits
• GDC 29	Protection Against Anticipated Operational Occurrences
10CFR50 App. A GDCs 30 – 46	*IV. Fluid Systems*
▪ GDC 30	Quality of Reactor Coolant Pressure Boundary
▪ GDC 31	Fracture Prevention of Reactor Coolant Pressure Boundary
▪ GDC 32	Inspection of Reactor Coolant Pressure Boundary
▪ GDC 33	Reactor Coolant Makeup *(N/A to NGNP RB study)*
▪ GDC 34	Residual Heat Removal
▪ GDC 35	Emergency Core Cooling
▪ GDC 36	Inspection of Emergency Core Cooling System
▪ GDC 37	Testing of Emergency Core Cooling System
▪ GDC 38	Containment Heat Removal
▪ GDC 39	Inspection of Containment Heat Removal System
▪ GDC 40	Testing of Containment Heat Removal System
▪ GDC 41	Containment Atmosphere Cleanup
▪ GDC 42	Inspection of Containment Atmosphere Cleanup Systems
▪ GDC 43	Testing of Containment Atmosphere Cleanup Systems
▪ GDC 44	Cooling Water
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▪ GDC 46	Testing of Cooling Water System
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▪ GDC 51	Fracture Prevention of Containment Pressure Boundary
▪ GDC 52	Capability for Containment Leakage Rate Testing
▪ GDC 53	Provisions for Containment Testing and Inspection
▪ GDC 54	Systems Penetrating Containment
▪ GDC 55	Reactor Coolant Pressure Boundary Penetrating Containment
▪ GDC 56	Primary Containment Isolation
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▪ GDC 62	Prevention of criticality in fuel storage and handling
▪ GDC 63	Monitoring fuel and waste storage

▪ GDC 64	Monitoring radioactivity releases
10 CFR Part 50 Appendix B	Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants
10 CFR Part 50 Appendix E	Emergency Planning and Preparedness for Production and Utilization Facilities
10 CFR Part 50 Appendix I	Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion "As Low as is Reasonably Achievable" for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents
10 CFR Part 50 Appendix J	Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors
10 CFR Part 50 Appendix N	Standardization of Nuclear Power Plant Designs: Permits To Construct and Licenses To Operate Nuclear Power Reactors of Identical Design at Multiple Sites
10 CFR Part 50 Appendix S	Earthquake Engineering Criteria for Nuclear Power Plants
10 CFR Part 51	**Environmental Protection Regulations For Domestic Licensing And Related Regulatory Functions**
10 CFR Part 51 §51.30	Environmental Assessment
10 CFR Part 52	**Licenses, Certifications, And Approvals For Nuclear Power Plants**
10 CFR Part 73	Physical protection of plants and materials
10 CFR Part 74	Material control and accounting of special nuclear material
10 CFR Part 100	Reactor site criteria
10 CFR Part 100 Appendix A	Seismic And Geologic Siting Criteria For Nuclear Power Plants

GUIDANCE DOCUMENTS

I. NUREG-0800 "Standard Review Plan (SRP) For The Review Of Safety Analysis Reports For Nuclear Power Plants"

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II NRC REGULATORY GUIDES

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--	**Division 1, Power Reactors**		
1.1	Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps (Safety Guide 1) (ML003739925)	--	11/1970
1.2	(Withdrawn -- See 56 FR 36175, 07/31/1991)	--	--
1.3	Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors (Rev. 2, ML003739601)	--	11/1970
1	06/1973		
2	06/1974		
1.4	Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors (Rev. 2, ML003739614)	--	11/1970
1	06/1973		
2	06/1974		
1.5	Assumptions Used for Evaluating the Potential Radiological Consequences of a Steam Line Break Accident for Boiling Water Reactors (Safety Guide 5) (ML003739923)	--	03/1971
1.6	Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems (Safety Guide 6) (ML003739924)	--	03/1971
1.7	Control of Combustible Gas Concentrations in Containment (ML070290080) (Draft was issued as DG-1117, 08/2002, ML022210067) (Rev. 2, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident," was issued 11/1978, ML003739927) (Rev. 1 was issued 09/1976) (Rev. 0 was issued 03/1971)	3	03/2007
1.8	Qualification and Training of Personnel for Nuclear Power Plants (Draft RS 807-5, Proposed Revision 2, published 02/1979; Draft RS 807-5, Second Proposed Revision 2, published 09/1980; Draft OL 403-5, Third Proposed Revision 2, published 01/1985; Draft DG-1012, Proposed Revision 3, published 09/1996, DG-1084, Second Proposed Revision 3, published 03/1999) (Rev. 2, ML003739928; Rev. 3, ML003706932)	3	05/2000
1.9	Application and Testing of Safety-Related Diesel Generators in Nuclear Power Plants (Rev. 4, ML070380553) (Draft Rev. 4 was issued as DG-1172, 11/2006, ML062650307; see the staff's responses to public comments on DG-1172, ML070380364) (Rev. 3, "Selection, Design, Qualification, and Testing of Emergency Diesel Generator Units Used as Class 1E Onsite Electric Power Systems at Nuclear Power Plants," was issued 07/1993, ML003739929) (Second Draft Rev. 3 was issued as DG-1021, 04/1992) (First Draft Rev. 3 was issued as RS 802-5, 11/1988) (Rev. 2 was issued 12/1979) (Rev. 1 was issued 11/1978) (Rev. 0 was issued 03/1971)	4	03/2007
1.10	(Withdrawn -- See 46 FR 37579, 07/21/1981)	--	--
1.11	Instrument Lines Penetrating Primary Reactor Containment (Safety Guide 11) Supplement to Safety Guide 11, Backfitting Considerations (ML003739934)	--	03/1971
1.12	Nuclear Power Plant Instrumentation for Earthquakes (Draft MS 140-5, Proposed Revision 2, published 07/1981) (DG-1016, the Second Proposed Revision 2, published 11/1992) (DG-1033, the Second Proposed Revision 2, published 02/1995) (Rev. 1, ML003739947; Rev. 2, ML003739944)	2	03/1997
1.13	Spent Fuel Storage Facility Design Basis (Rev. 2, ML070310035) (Second Draft Rev. 2 was issued as DG-1162, 10/2006, ML062680291; see the staff's responses to public comments on DG-1162, ML070300300) (First Draft Rev. 2 was issued as CE 913-5, 12/1981) (Rev. 1 was issued for comment 12/1975, ML003739943) (Rev. 0 was issued 03/1971)	2	03/2007
1.14	Reactor Coolant Pump Flywheel Integrity (Rev. 1, ML003739936)	1	08/1975
1.15	(Withdrawn -- See 46 FR 37579, 07/21/1981)	--	--
1.16	Reporting of Operating Information -- Appendix A Technical Specifications (for Comment) (Rev. 4, ML003739954)	4	08/1975
1.17	(Withdrawn -- See 56 FR 30777, 07/5/1991)	--	--
1.18	(Withdrawn -- See 46 FR 37579, 07/21/1981)	--	--
1.19	(Withdrawn -- See 46 FR 37579, 07/21/1981)	--	--
1.20	Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing (Rev. 3, ML070260376) (Draft Rev. 3 was issued as DG-1163, 11/2006, ML062750162; see the staff's responses to public comments on DG-1163, ML070260382) (Rev. 2 was issued 05/1976, ML003739957) (Rev. 1 was issued 06/1975) (Rev. 0 was issued 12/1971)	3	03/2007
1.21	Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants (Rev. 1, ML003739960)	1	06/1974
1.22	Periodic Testing of Protection System Actuation Functions (Safety Guide 22)	--	02/1972
1.23	Meteorological Monitoring Programs for Nuclear Power Plants (Rev. 1, ML070350028)	1	03/2007
1.24	Assumptions Used for Evaluating the Potential Radiological Consequences of a Pressurized Water Reactor Radioactive Gas Storage Tank Failure (Safety Guide 24)	--	03/1972
1.25	Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors (Safety Guide 25)	--	03/1972
1.26	Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants (Rev. 4, ML070290283) (Draft Rev. 4 was issued as DG-1152, 11/2006, ML063100379; no comments were received) (Rev. 3 was issued for comment, 02/1976, ML003739964) (Rev. 2 was issued 06/1975) (Rev. 1 was issued 09/1974) (Rev. 0 was issued 03/1972)	4	03/2007

1.27	Ultimate Heat Sink for Nuclear Power Plants (for Comment) (Rev. 2, ML003739969)	2	01/1976
1.28	Quality Assurance Program Requirements (Design and Construction) (Rev. 3, ML003739981) (Draft RS 002-5, Proposed Revision 3, published 03/1981) (Draft DG-1010, Proposed Revision 4, published 11/1992)	3	08/1985
1.29	Seismic Design Classification (Rev. 4, ML070310052) (Draft Rev. 4 was issued as DG-1156, 10/2006, ML062540294; see the staff's responses to public comments on DG-1156, ML070300537) (Rev. 3 was issued 09/1978, ML003739983) (Rev. 2 was issued 02/1976) (Rev. 1 was issued 08/1973) (Rev. 0 was issued 06/1972)	4	03/2007
1.30	Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment (Safety Guide 30) (ML081270243)	--	08/1972
1.31	Control of Ferrite Content in Stainless Steel Weld Metal (Rev. 3, ML003739986)	3	04/1978
1.32	Criteria for Power Systems for Nuclear Power Plants (Rev. 2, ML003739990) (DG-1079, Proposed Revision 3, issued 04/2003, ML031280598) (Rev. 3, ML040680488)	3	03/2004
1.33	Quality Assurance Program Requirements (Operation) (Draft RS 902-4, Proposed Revision 3, published 08/1979) (Draft RS 902-4, Second Proposed Revision 3, published 11/1980) (Rev. 2, ML003739995)	2	02/1978
1.34	Control of Electroslag Weld Properties (ML003739997)	--	12/1972
1.35	Inservice Inspection of Ungrouted Tendons in Prestressed Concrete Containments (Rev. 2, ML003740001) (Draft SC 810-4, Proposed Revision 3, published 04/1979) (Rev. 3, ML003740007)	3	07/1990
1.35.1	Determining Prestressing Forces for Inspection of Prestressed Concrete Containments (ML003740040) (Draft SC 807-4 published 04/1979)	--	07/1990
1.36	Nonmetallic Thermal Insulation for Austenitic Stainless Steel (ML003740046)	--	02/1973
1.37	Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants (Rev. 1, ML070250571) (Draft Rev. 1, was issued as DG-1165, 10/2006, ML063040652; see the staff's responses to public comments on DG-1165, ML070300798) (Rev. 0 was issued 03/1973, ML003740051)	1	03/2007
1.38	Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants (Rev. 2, ML003740057)	2	05/1977
1.39	Housekeeping Requirements for Water-Cooled Nuclear Power Plants (Rev. 2, ML003740067)	2	09/1977
1.40	Qualification Tests of Continuous-Duty Motors Installed Inside the Containment of Water-Cooled Nuclear Power Plants (ML003740083)	--	03/1973
1.41	Preoperational Testing of Redundant On-Site Electric Power Systems To Verify Proper Load Group Assignments (ML003740090)	--	03/1973
1.42	(Withdrawn -- See 41 FR 11891, 03/22/1976)	--	--
1.43	Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components (ML003740095)	--	05/1973
1.44	Control of the Use of Sensitized Stainless Steel (ML003740109)	--	05/1973
1.45	Guidance on Monitoring and Responding to Reactor Coolant System Leakage (Rev. 1, ML073200271)	1	05/2008
1.46	(Withdrawn -- See 50 FR 9732, 03/11/1985)	--	--
1.47	Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems (ML003740127)	--	05/1973
1.48	(Withdrawn -- See 50 FR 9732, 03/11/1985)	--	--
1.49	(Withdrawn -- See 72 FR 36737, 07/05/2007)	--	--
1.50	Control of Preheat Temperature for Welding of Low-Alloy Steel (ML003740136)	--	05/1973
1.51	(Withdrawn -- See 40 FR 30510, 07/21/1975)	--	--
1.52	Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants (Rev. 2, ML003740139) (DG-1102, Proposed Revision 3, issued 10/00, ML003756180) (Rev. 3, ML011710176)	3	06/2001
1.53	Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems (ML003740182) (Draft DG-1118, Proposed Revision 1, ML021260080, published 05/2002) (Rev.1, ML032670945) (Rev. 2, ML033220006)	2	11/2003
1.54	Service Level I, II, and III Protective Coatings Applied to Nuclear Power Plants (ML003740187) (Draft DG-1976, Proposed Revision 1, ML003739156, published 03/1999) (Rev. 1, ML003714475)	1	07/2000
1.55	(Withdrawn -- See 46 FR 37579, 07/21/1981)	--	--
1.56	Maintenance of Water Purity in Boiling Water Reactors (for Comment) (Rev. 1, ML003740192)	1	07/1978
1.57	Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components (Rev. 1, ML070310029) (Draft Rev. 1 was issued as DG-1158, 10/2006, ML063000278; see the staff's responses to public comments on DG-1158, ML070240173) (Rev. 0 was issued 06/1973, ML003740195)	1	03/2007
1.58	(Withdrawn -- See 56 FR 36175, 07/31/1991)	--	--
1.59	Design Basis Floods for Nuclear Power Plants (Errata published 07/30/1980) (Rev. 2, ML003740388)	2	08/1977
1.60	Design Response Spectra for Seismic Design of Nuclear Power Plants (Rev. 1, ML003740207)	1	12/1973
1.61	Damping Values for Seismic Design of Nuclear Power Plants (Rev. 1, ML070260029)	1	03/2007
1.62	Manual Initiation of Protective Actions (ML003740216)	--	10/1973
1.63	Electric Penetration Assemblies in Containment Structures for Nuclear Power Plants (Draft EE 405-4, Proposed Revision 3, published 06/1986) (Rev. 3, ML003740219)	3	02/1987
1.64	(Withdrawn -- See 56 FR 36175, 07/31/1991)	--	--
1.65	Materials and Inspections for Reactor Vessel Closure Studs (ML003740228)	--	10/1973
1.66	(Withdrawn -- See 42 FR 54478, 10/6/1977)	--	--
1.67	(Withdrawn -- See 48 FR 19101, 04/27/1983)	--	--
1.68	Initial Test Programs for Water-Cooled Nuclear Power Plants (Rev. 3, ML070260039) (Draft Rev. 3 was issued	3	03/2007

	as DG-1166, 11/2006, ML062750316; see the staff's responses to public comments on DG-1166, ML070360680) (Rev. 2 was issued 08/1978) (Rev. 1 was issued 01/1977) (Rev. 0 was issued 11/1973)		
1.68.1	Preoperational and Initial Startup Testing of Feedwater and Condensate Systems for Boiling Water Reactor Power Plants (Rev. 1, ML003740230)	1	01/1977
1.68.2	Initial Startup Test Program To Demonstrate Remote Shutdown Capability for Water-Cooled Nuclear Power Plants (Rev. 1, ML003740258)	1	07/1978
1.68.3	Preoperational Testing of Instrument and Control Air Systems (Draft RS 709-4, a proposed revision to Regulatory Guide 1.80, published 10/1980) (ML003740231)	--	04/1982
1.69	Concrete Radiation Shields for Nuclear Power Plants (ML003740235)	--	12/1973
1.70	Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition) (Rev. 2, ML010610289) (Rev. 3 in three parts, ML011340072, ML011340108, and ML011340116)	3	11/1978
1.71	Welder Qualification for Areas of Limited Accessibility (Rev. 1, ML070320476)	1	03/2007
1.72	Spray Pond Piping Made from Fiberglass-Reinforced Thermosetting Resin (Rev. 2, ML003740253)	2	11/1978
1.73	Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants (ML003740261)	--	01/1974
1.74	(Withdrawn -- See 54 FR 38919, 09/21/1989)	--	--
1.75	Physical Independence of Electric Systems (Rev. 2, ML003740265) (Rev. 3, ML043630448)	3	02/2005
1.76	Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants (Rev. 1, ML070360253) (Draft Rev. 1 was issued as DG-1143, 01/2006, ML053140225; see the staff's responses to public comments on DG-1143, ML070360258) (Rev. 0, "Design-Basis Tornado for Nuclear Power Plants," was issued 04/1974, ML003740273)	1	03/2007
1.77	Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors (ML003740279)	--	05/1974
1.78	Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release (ML003740298) (Proposed Revision 1, DG-1087, published 02/2001, ML010440064) (Revision 1 incorporates guidance from withdrawn Regulatory Guide 1.95) (Revision 1, ML013100014)	1	12/2001
1.79	Preoperational Testing of Emergency Core Cooling Systems for Pressurized Water Reactors (Rev. 1, ML003740351)	1	09/1975
1.80	(Withdrawn -- See 47 FR 19258, 05/4/1982) Reissued as Regulatory Guide 1.68.3, a renumbered revision to this guide with an expanded scope that addresses control air systems.	--	--
1.81	Shared Emergency and Shutdown Electric Systems for Multi-Unit Nuclear Power Plants (Rev. 1, ML003740343)	1	01/1975

1.82	Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident (Draft MS 203-4, Proposed Revision 1, published 05/1983) (Rev. 1, ML003740236) (Draft DG-1038, Proposed Revision 2, ML003739202, published 07/1995) (Rev. 2, ML003740249) (Rev. 3, ML033140347)	3	11/2003
1.83	Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes (Rev. 1, ML003740256)	1	07/1975
1.84	Design, Fabrication, and Materials Code Case Acceptability, ASME Section III	34	10/2007
1.85	(Withdrawn 06/2003) Materials Code Case Acceptability -- ASME Section III, Division 1 (Guidance incorporated into Revision 32 of Regulatory Guide 1.84, published 06/2003, ML041620482)	31	05/1999
1.86	Termination of Operating Licenses for Nuclear Reactors (ML003740243)	--	06/1974
1.87	Guidance for Construction of Class 1 Components in Elevated-Temperature Reactors (Supplement to ASME Section III Code Cases 1592, 1593, 1594, 1595, and 1596) (Rev. 1, ML003740252)	--	06/1975
1.88	(Withdrawn -- See 56 FR 36175, 07/31/1991)	--	--
1.89	Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants (1974, ML012880422) (Draft EE 042-2, Proposed Revision 1, published 02/1982) (Rev. 1, ML003740271)	1	06/1984
1.90	Inservice Inspection of Prestressed Concrete Containment Structures with Grouted Tendons (Rev. 1, ML003740281)	1	08/1977
1.91	Evaluations of Explosions Postulated To Occur on Transportation Routes Near Nuclear Power Plants (Rev. 1, ML003740286)	1	02/1978
1.92	Combining Modal Responses and Spatial Components in Seismic Response Analysis (Rev. 0, 12/1974) (Rev. 1, ML003740290, 02/1976) (Proposed Revision 2, published as DG-1108, 08/2001) (Second Proposed Revision 2, published as DG-1127, ML050230006, 02/2005) (Rev.2, ML053250475, 07/2006)	2	07/2006
1.93	Availability of Electric Power Sources (ML003740292)	--	12/1974
1.94	Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants (Rev. 1, ML003740305) (Draft RS 908-5, Proposed Revision 2, published 09/1979)	1	04/1976
1.95	(Withdrawn January 2002) Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release (Rev. 1, ML003740310) (Guidance incorporated into Revision 1 of Regulatory Guide 1.78, published 01/2002) (Withdrawn 12/26/2001, ML020020470)	--	--
1.96	Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants (Rev. 1, ML003740263)	1	06/1976
1.97	Criteria For Accident Monitoring Instrumentation For Nuclear Power Plants (Rev. 0, entitled "Instrumentation for Light-Water-Cooled Nuclear Power Plants To Assess	4	06/2006

	Plant and Environs Conditions During and Following an Accident," 12/1975) (Rev. 1, 08/1977) (Proposed Rev. 2, Draft RS 917-4, 12/1979) (Rev. 2, ML060750525, 12/1980) (Rev. 2 Errata, ML060750527, 07/1981) (Rev. 3, ML003740282, 05/1983) (Proposed Rev. 4, DG-1128, ML052150210, 08/2005) (Rev.4, ML061580448, 06/2006)		
1.98	Assumptions Used for Evaluating the Potential Radiological Consequences of a Radioactive Offgas System Failure in a Boiling Water Reactor (for Comment) (ML003740259)	--	03/1976
1.99	Radiation Embrittlement of Reactor Vessel Materials (Draft ME 305-4, Proposed Revision 2, published 02/1986) (Rev. 2, ML003740284)	2	05/1988
1.100	Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants (Draft EE 108-5, Proposed Revision 2, published 08/1987) (Rev. 2, ML003740293)	2	06/1988
1.101	Emergency Planning and Preparedness for Nuclear Power Reactors (Revision 1 to this guide, "Emergency Planning for Nuclear Power Plants," was withdrawn--see 45 FR 69610, 10/21/1980.) (Draft DG-1022, Proposed Revision 3, published 02/1992) (Rev. 3, ML003740302) (Draft DG-1075, Proposed Revision 4, published 03/2000, ML003687957) (Rev. 4, ML032020276) (Rev. 5, ML050730286)	5	06/2005
1.102	Flood Protection for Nuclear Power Plants (ML003740308)	--	10/1975
1	09/1976		
1.103	(Withdrawn -- See 46 FR 37579, 07/21/1981)	--	--
1.104	(Withdrawn -- See 44 FR 49321, 08/22/1979) See NUREG-0554, "Single-Failure-Proof Cranes for Nuclear Power Plants."	--	--
1.105	Setpoints for Safety-Related Instrumentation (Draft IC 010-5, Proposed Revision 2, published 12/81) (Draft DG-1045, Proposed Revision 3, ML003739248, published 10/96) (Rev. 2, ML003740318; Rev. 3, ML993560062)	3	12/1999
1.106	Thermal Overload Protection for Electric Motors on Motor-Operated Valves (Rev. 1, ML003740323)	1	03/1977
1.107	Qualifications for Cement Grouting for Prestressing Tendons in Containment Structures (Rev. 1, ML003740374)	1	02/1977
1.108	(Withdrawn -- See 58 FR 41813, 08/5/1993) (Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants)	--	--
1.109	Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I (Rev. 1, ML003740384)	1	10/1977
1.110	Cost-Benefit Analysis for Radwaste Systems for Light-Water-Cooled Nuclear Power Reactors (for Comment) (ML003740332)	--	03/1976
1.111	Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors (Rev. 1, ML003740354)	1	07/1977
1.112	Calculation of Releases of Radioactive Materials in	1	03/2007

	Gaseous and Liquid Effluents from Light-Water-Cooled Power Reactors (Rev. 1, ML070320241) (Draft Rev. 1, was issued as DG-1160, 10/2006, ML062680253; see the staff's responses to public comments on DG-1160, ML070360671) (Rev. 0-R was issued as errata, 05/1977, ML003740361) (Rev. 0 was issued 04/1976)		
1.113	Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I (Rev. 1, ML003740390)	1	04/1977
1.114	Guidance to Operators at the Controls and to Senior Operators in the Control Room of a Nuclear Power Unit (Draft HF 601-4, Proposed Revision 2, published 12/1986) (Rev. 2, ML003740393)	2	05/1989
1.115	Protection Against Low-Trajectory Turbine Missiles (Rev. 1, ML003739456)	1	07/1977
1.116	Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems (Rev. 0-R, ML003739465)	0-R	05/1977
1.117	Tornado Design Classification (Rev. 1, ML003739346)	1	04/1978
1.118	Periodic Testing of Electric Power and Protection Systems (Rev. 3, ML003739468) (DG-1028, Proposed Revision 3, published 09/1994) (Rev. 3, ML003739468)	3	04/1995
1.119	(Withdrawn -- See 42 FR 33387, 06/30/1977)	--	--
1.120	Withdrawn August 15, 2001 (Fire Protection Guidelines for Nuclear Power Plants) (issued 06/1976, Revision 1 issued 11/1977) (Rev. 1 ML003739360, Withdrawal ML012340082)		
1.121	Bases for Plugging Degraded PWR Steam Generator Tubes (for Comment) (ML003739366)	--	08/1976
1.122	Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components (Rev. 1, ML003739367)	1	02/1978
1.123	(Withdrawn -- See 56 FR 36175, 07/31/1991)	--	--
1.124	Service Limits and Loading Combinations for Class 1 Linear-Type Supports (Rev. 2, ML070160591)	2	02/2007
1.125	Physical Models for Design and Operation of Hydraulic Structures and Systems for Nuclear Power Plants (Rev. 1, ML003739388)	1	10/1978
1.126	An Acceptable Model and Related Statistical Methods for the Analysis of Fuel Densification (Rev. 1, ML003739385)	1	03/1978
1.127	Inspection of Water-Control Structures Associated with Nuclear Power Plants (Rev. 1, ML003739392)	1	03/1978
1.128	Installation Design and Installation of Vented Lead-Acid Storage Batteries for Nuclear Power Plants (Rev. 2, ML070080013) (Draft Rev. 2, was issued as DG-1154, 10/2006, ML062220170; see the staff's responses to public comments on DG-1154, ML070030154) (Rev. 1, "Installation Design and Installation of Large Lead Storage Batteries for Nuclear Power Plants," was issued 10/1978, ML003740099) (Rev. 0 was issued 04/1977)	2	02/2007
1.129	Maintenance, Testing, and Replacement of Vented Lead-Acid Storage Batteries for Nuclear Power Plants (Rev. 2, ML063490110) (Draft Rev. 2, was issued as DG-1155, 10/2006, ML062540343; see the staff's responses to public comments on DG-1155, ML063490093) (Rev. 1,	2	02/2007

	"Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Nuclear Power Plants," was issued 02/1978, ML003740104) (Rev. 0 was issued 04/1977)		
1.130	Service Limits and Loading Combinations for Class 1 Plate-and-Shell-Type Component Supports (Rev. 2, ML070170053)	2	03/2007
1.131	Qualification Tests of Electric Cables, Field Splices, and Connections for Light-Water-Cooled Nuclear Power Plants (for Comment) (ML003740128)	--	08/1977
1.132	Site Investigations for Foundations of Nuclear Power Plants (Rev. 2, ML032800710) (DG-1101, Proposed Revision 2, issued 02/2001, ML010510162) (Rev. 1, ML003740350)	2	10/2003
1.133	Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors (Rev. 1, ML003740137)	1	05/1981
1.134	Medical Evaluation of Licensed Personnel at Nuclear Power Plants (Draft OL 401-5, Proposed Revision 2, published 11/1984) (Draft DG-1068, Proposed Revision 3, ML003739137, published 02/1997) (Rev. 2, ML003740138) (Rev. 3, ML003740140)	3	03/1998
1.135	Normal Water Level and Discharge at Nuclear Power Plants (for Comment) (ML003740143)	--	09/1977
1.136	Design Limits, Loading Combinations, Materials, Construction, and Testing of Concrete Containments (Rev. 3, ML070310045) (Draft Rev. 3 was issued as DG-1159, 10/2006, ML063000430; see the staff's responses to public comments on DG-1159, ML070240146) (Rev. 2, "Materials, Construction, and Testing of Concrete Containments (Articles CC-1000, -2000, and -4000 through -6000 of the 'Code for Concrete Reactor Vessels and Containments')," was issued 06/1981, ML003740155) (Draft Rev. 2 was issued as SC 814-5, 11/1979) (Rev. 1 was issued 10/1978) (Rev. 0 was issued 11/1977)	3	03/2007
1.137	Fuel-Oil Systems for Standby Diesel Generators (Rev. 1, ML003740180)	1	10/1979
1.138	Laboratory Investigations of Soils and Rocks for Engineering Analysis and Design of Nuclear Power Plants (04/1978, ML003740184) (DG-1109, Proposed Revision 1, published 08/2001, ML012420328) (Revision 1 was not published) (Revision 2, ML033510166)	2	12/2003
1.139	(Withdrawn -- See 73 FR 32750, 06/10/2008)		
1.140	Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Normal Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants (DG-1103, Proposed Revision 2, issued 10/2000) (Rev. 1, ML003740190; Rev. 2, ML011710150)	2	06/2001
1.141	Containment Isolation Provisions for Fluid Systems (for Comment) (ML003740194)	--	04/1978
1.142	Safety-Related Concrete Structures for Nuclear Power Plants (Other than Reactor Vessels and Containments) (DG-1098, Proposed Revision 2, published 08/2000) (Rev. 1, ML003740197; Rev. 2, ML013100274)	2	11/2001
1.143	Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants (DG-1100, Proposed	2	11/2001

	Revision 2, published 08/2000) (Rev. 1, ML003740200; Rev. 2, ML013100305)		
1.144	(Withdrawn -- See 56 FR 36175, 07/31/1991)	--	--
1.145	Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants (Reissued 02/1983 to correct page 1.145-7) (Rev. 1, ML003740205)	1	11/1982
1.146	(Withdrawn -- See 56 FR 36175, 07/31/1991)	--	--
1.147	Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1	15	10/2007
1.148	Functional Specification for Active Valve Assemblies in Systems Important to Safety in Nuclear Power Plants (ML003739979) (Draft SC 704-5 published 02/1979)	--	03/1981
1.149	Nuclear Power Plant Simulation Facilities for Use in Operator Training and License Examinations (Draft RS 110-5 published 07/1980) (Draft OL 402-5, Proposed Revision 1, published 11/1984) (Draft DG-1043, Proposed Revision 2, published 06/1995) (DG-1080, Proposed Revision 3, ML003739149, published 08/1999) (Rev. 1, ML003739984; Rev. 2, ML003739988; Rev. 3, ML012770164)	3	10/2001
1.150	(Withdrawn-- See 73 FR 7766, 02/11/2008)	--	--
1.151	Instrument Sensing Lines (ML003740003) (Draft IC 126-5 published 03/1982)	--	07/1983
1.152	Criteria for Digital Computers in Safety Systems of Nuclear Power Plants (11/85) (ML003740088) (Draft IC 127-5 published 03/1983) (Draft DG-1039, Proposed Revision 1, published 05/1995) (Rev. 1, ML003740015) (Draft DG-1130, Proposed Revision 2, Published 12/2004) (Rev. 2, ML053070150)	2	01/2006
1.153	Criteria for Safety Systems (12/85) (Draft IC 609-5 published 12/1982) (Draft DG-1042, Proposed Revision 1, published 11/1995) (ML003740019) (Rev. 1, ML003740022)	1	06/1996
1.154	Format and Content of Plant-Specific Pressurized Thermal Shock Safety Analysis Reports for Pressurized Water Reactors (ML003740028) (Draft SI 502-4 published 01/1986)	--	01/1987
1.155	Station Blackout (Issued June 1988, reissued August 1988 with corrected tables) (ML003740034) (Draft SI 501-4 published 03/1986)	--	08/1988
1.156	Environmental Qualification of Connection Assemblies for Nuclear Power Plants (ML003740042) (Draft EE 404-4 published 05/1987)	--	11/1987
1.157	Best-Estimate Calculations of Emergency Core Cooling System Performance (ML003739584) (Draft RS 701-4 published 03/1987)	--	05/1989
1.158	Qualification of Safety-Related Lead Storage Batteries for Nuclear Power Plants (ML003740047) (Draft EE 006-5 published 08/1987)	--	02/1989
1.159	Assuring the Availability of Funds for Decommissioning Nuclear Reactors (Draft DG-1003 published 05/1989, ML003739365) (Draft DG-1106, Proposed Revision 1, published 05/2001, ML010170350) (08/90, ML003740066) (Rev. 1, ML032790365)	1	10/2003

1.160	Monitoring the Effectiveness of Maintenance at Nuclear Power Plants (Draft DG-1020 published 11/1992) (Draft DG-1031, Proposed Revision 1, published 06/1994) (Draft DG-1051, Proposed Revision 2, ML003739233, published 08/1996) (Rev. 1, ML031430362) (Rev. 2, ML003761662)	2	03/1997
1.161	Evaluation of Reactor Pressure Vessels with Charpy Upper-Shelf Energy Less Than 50 Ft-Lb (ML003740038) (Draft DG-1023 published 09/1993)	--	06/1995
1.162	Format and Content of Report for Thermal Annealing of Reactor Pressure Vessels (ML003740052) (Draft DG-1027 published 10/1994)	--	02/1996
1.163	Performance-Based Containment Leak-Test Program (ML003740058) (Draft DG-1037 published 02/1995) (Errata to NEI 94-01 published 03/1996)	--	09/1995
1.164	(Not yet issued)		
1.165	Identification and Characterization of Seismic Sources and Determination of Safe Shutdown Earthquake Ground Motion (ML003740084) (Draft DG-1015 issued 11/1992, Draft DG-1032, issued 02/1995)	--	03/1997
1.166	Pre-Earthquake Planning and Immediate Nuclear Power Plant Operator Postearthquake Actions (ML003740089) (Draft DG-1017 issued 11/1992, Draft DG-1034, ML003739203, issued 02/1995)	--	03/1997
1.167	Restart of a Nuclear Power Plant Shut Down by a Seismic Event (ML003740093) (Draft DG-1018 issued 11/1992, Draft DG-1035, ML003739196, issued 02/1995)	--	03/1997
1.168	Verification, Validation, Reviews, and Audits for Digital Computer Software Used in Safety Systems of Nuclear Power Plants (DG-1054, ML003739159, issued 08/1996) (09/97, ML003740098) (DG-1123, Proposed Revision 1, 01/2003, ML030270328) (Rev. 1, ML040410189)	1	02/2004
1.169	Configuration Management Plans for Digital Computer Software Used in Safety Systems of Nuclear Power Plants (ML003740102) (Draft DG-1055, ML003739153, issued 08/1996)	--	09/1997
1.170	Software Test Documentation for Digital Computer Software Used in Safety Systems of Nuclear Power Plants (ML003740105) (Draft DG-1056, ML003739146, issued 08/1996)	--	09/1997
1.171	Software Unit Testing for Digital Computer Software Used in Safety Systems of Nuclear Power Plants (ML003740108) (Draft DG-1057, ML003739141, issued 08/1996)	--	09/1997
1.172	Software Requirements Specifications for Digital Computer Software Used in Safety Systems of Nuclear Power Plants (ML003740094) (Draft DG-1058, ML003739228, issued 08/1996)	--	09/1997
1.173	Developing Software Life Cycle Processes for Digital Computer Software Used in Safety Systems of Nuclear Power Plants (ML003740101) (Draft DG-1059, ML003740101, issued 08/1996)	--	09/1997
1.174	An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis (ML003740133)	1	11/2002
1.175	An Approach for Plant-Specific, Risk-Informed Decisionmaking: Inservice Testing (ML003740149) (Issued with SRP Chapter 3.9.7) (Draft DG-1062, ML003739158, issued 06/1997)	--	08/1998
1.176	(Withdrawn-- See 73 FR 7766, 02/11/2008)	--	--
1.177	An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications (ML003740176) (Issued with SRP Chapter 16.1) (Draft DG-1065, ML003739150, issued 06/1997)	--	08/1998
1.178	An Approach for Plant-Specific Risk-Informed Decisionmaking for Inservice Inspection of Piping (9/98, ML003740181) (Issued with SRP Chapter 3.9.8) (Draft DG-1063, ML003739154, issued 10/1997) (Revision 1, ML032510128, issued 09/2003) (SRP, ML032510135)	1	09/2003
1.179	Standard Format and Content of License Termination Plans for Nuclear Power Reactors (ML003740212) (Draft DG-1078, ML003739152, issued 04/1998)	--	01/1999
1.180	Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems (01/00, ML003740218)	1	10/2003
1.181	Content of the Updated Final Safety Analysis Report in Accordance with 10 CFR 50.71(e) (ML003740112) (Draft DG-1083 issued 03/1999)	--	09/1999
1.182	Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants (ML003740117) (Draft DG-1082 issued 12/1999)	--	05/2000
1.183	Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors (ML003716792) (Draft DG-1081, ML993490086, issued 12/1999)	--	07/2000
1.184	Decommissioning of Nuclear Power Reactors (ML003701137) (Draft DG-1067, ML003739144, issued 06/1997) (Errata to update Reference 1 (ML040920341), published 04/2004)	--	07/2000
1.185	Standard Format and Content for Post-Shutdown Decommissioning Activities Report (ML003701163) (Draft DG-1071, ML003739227, issued 12/1997) (Errata to update Reference 1 (ML040920341), published 04/2004)	--	07/2000
1.186	Guidance and Examples for Identifying 10 CFR 50.2 Design Bases (ML003754825) (Draft DG-1093, ML003691412, published 04/2000)	--	12/2000
1.187	Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments (ML003759710) (Draft DG-1095, ML003698165, issued 04/2000)	--	11/2000
1.188	Standard Format and Content for Applications To Renew Nuclear Power Plant Operating Licenses (ML012010322) (Drafts were issued as DG-1104, ML003736097, 08/2000; DG-1047, ML003739244, 08/1996; and DG-1009, 12/1990.) (Proposed Rev. 1 was issued as DG-1140, ML050230010, 01/2005.) (Rev. 1, ML051920430, 09/2005.)	1	09/2005
1.189	Fire Protection for Nuclear Power Plants (Rev. 1, ML070370183	01	03/2007
1.190	Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence (ML010890301) (Drafts were DG-1053 and DG-1025 (09/1993)) (DG-1025,	--	03/2001

	ML003739334)		
1.191	Fire Protection Program for Nuclear Power Plants During Decommissioning and Permanent Shutdown (ML011500010) (Draft DG-1069, ML003739129, published 07/1998)	--	05/2001
1.192	Operation and Maintenance Code Case Acceptability, ASME OM Code (ML030730430) (Draft guide was issued as DG-1089, 12/01, ML013120051)	--	06/2003
1.193	ASME Code Cases Not Approved for Use	2	10/2007
1.194	Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants (ML031530505) (Draft guide was issued as DG-1111, 12/01, ML013130132)	--	06/2003
1.195	Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors (ML031490640) (Draft guide was issued as DG-1113, 01/02, ML020160023)	--	05/2003
1.196	Control Room Habitability at Light-Water Nuclear Power Reactors (ML063560144)	1	01/2007
1.197	Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors (ML031490664) (Draft guide was issued as DG-1115, 03/02, ML020790191)	--	05/2003
1.198	Procedures and Criteria for Assessing Seismic Soil Liquefaction at Nuclear Power Plant Sites (ML033280143) (Draft guide was issued as DG-1105, 03/01, ML010650295)	--	11/2003
1.199	Anchoring Components and Structural Supports in Concrete (ML033360660) (Draft was issued as DG-1099, 07/02, ML021910490)	--	11/2003
1.200	An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities, ML070240001 (Clarification to RG 1.200, Rev. 1, ML071940235)	1	01/2007
1.201	Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to Their Safety Significance (Rev. 0, ML060260164) (Draft guide was issued as DG-1121, 05/03, ML031430373) (Rev. 1, ML061090627, 05/2006)	1	05/2006
1.202	Standard Format and Content of Decommissioning Cost Estimates for Nuclear Power Reactors (ML050230008) (Draft guide was issued as DG-1085, 11/2001, ML013100099)	--	02/2005
1.203	Transient and Accident Analysis Methods (ML053500170) (Issued with ML053550265) SRP Section 15.0.2 , ML053550265, 12/2005) (Drafts were issued as DG-1096, ML003770849, 12/2000, and DG-1120, ML030160652, 12/2002)	--	12/2005
1.204	Guidelines for Lightning Protection of Nuclear Power Plants (ML052290422) (Draft guide was issued as DG-1137, 02/ 2005, ML050480101)	--	11/2005
1.205	Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants (ML061100174) (Draft was issued as DG-1139, ML042740308, dated September 2004)	--	05/2006
1.206	Combined License Applications for Nuclear Power Plants	--	06/2007
	(LWR Edition). Publication of the associated revision to Title 10 Code of Federal Regulations Part 52 is pending. (Draft guide was issued as DG-1145, dated September 2006)		
1.207	Guidelines for Evaluating Fatigue Analyses Incorporating the Life Reduction of Metal Components Due to the Effects of the Light-Water Reactor Environment for New Reactors (Rev. 0, ML070380586)	--	03/2007
1.208	A Performance-Based Approach to Define the Site-Specific Earthquake Ground Motion (Rev. 0, ML070310619) (Draft was issued as DG-1146, 10/2006, ML063030291; see the staff's responses to public comments on DG-1146, ML070380437)	--	03/2007
1.209	Guidelines for Environmental Qualification of Safety-Related Computer-Based Instrumentation and Control Systems in Nuclear Power Plants (Rev. 0, ML070190294) (Second Draft was issued as DG-1142, 10/2006, ML063040591; see the staff's responses to public comments on DG-1142, ML070260193) (First Draft was issued as DG-1077, "Guidelines for Environmental Qualification of Microprocessor-Based Equipment Important to Safety in Nuclear Power Plants," dated September 2001, ML012710073)	--	03/2007
1.210	Qualification of Safety-Related Battery Chargers and Inverters for Nuclear Power Plants (ML080640184)	--	06/2008
--	**Division 4, Environmental and Siting (RG Used in Reactor Building Study)**		
4.21	Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning	--	06/2008

APPENDIX B – Geotechnical Considerations for the INL Site

Geotechnical Considerations for the Idaho National Laboratories Site

1.0 Site Lithology

Reference 1 summarizes the results of geotechnical investigation performed at several locations within the INL site that is located in southeastern Idaho. The site is within a tectonic region, which is thought to represent the track of the Yellowstone Hotspot currently located beneath the Yellowstone National Park, Wyoming. Hotspot volcanism produced large volume silicic eruptions that were followed by predominantly basaltic volcanism which have resulted in 1000 to 2000 m deep strata of basalt lava flows and sediments. The basalt-sediment sequence overlies rhyolitic deposits associated with Yellowstone Hotspot volcanism. The thickness of the sedimentary interbeds range from 0 to 97 m with thicker interbeds occurring within the deeper parts of the basalt sections.

In general, massive interior zones with vertical cooling fractures and fractured rubble zones at their upper and lower contacts characterize the basalt lava flows. The basalt lava flows form an irregular bedrock surface and is characterized with cracks, joints, and fissures in the tops of lava flows. The previous site investigations indicate bedrock surfaces irregularities over small areas beneath individual facilities as well as over the facility area.

2.0 Depth of Bedrock

Fine-grained alluvium sediments rest above the basalt bedrock. In portions of the site, lithologic logs indicate the presence of a clay layer 0.4 to 23 ft thick resting at the top of the bedrock. The overall depth of the soil strata over the bedrock is characterized with a great level of variation from 20 ft to 65 ft due to irregular bedrock topography. The depth of the bedrock and its variation at the specific NGNP site location at INL has to be established since it is important input for the conceptual design.

3.0 Ground Water Elevation Bedrock

Reference 1 indicates that at one location the water table elevation is below the alluvial soil sediments. The input depth of the water table has to be established. The previous geological surveys indicate fractures and fractured rubble zones in the bedrock which can affect the design with regard to ground water considerations.

4.0 Soil Density Profiles

Based on measurements of shear wave velocity and lithologic logs, Reference 2 distinguished two layers of fine grained alluvium soil designated as Upper Alluvium Soil (UAS) and Lower Alluvium Soil (LAS). Per Reference 2, the measured densities for the UAS range from 96 to 114 pcf and LAS varies from 71 to 133 pcf. The densities for basalt range from 127 to 184 pcf. The site response calculations in Reference 2 used 118 pcf for alluvial soil and 159 fps for the bedrock.

5.0 Shear Wave Velocity Profiles

Based on the filed measurements, Reference 2 used the following input mean values Vs and coefficients of variation COV for the small-strain shear wave velocities:

Stratum	Vs (fps)	COV (Log_{10})
UAS	971	0.068
LAS	1539	0.045
Basalt	3718	0.049

Table 9 in Reference 2 provides for the soil-structure interaction analyses, the following lower bound (LB), best estimate (BE), and upper bound (UB) profiles for the damping, shear and compression wave velocities of the soil that are compatible to the strains generated by design earthquake with 10,000 year reoccurrence probability.

Stratum	Vs (fps)			Vp (fps)			Damping (%)		
	LB	BE	UB	LB	BE	UB	LB	BE	UB
UAS	610	858	1206	1057	1486	2090	6.0	3.5	2.1
LAS	1074	1390	1800	1860	2408	3118	4.9	3.4	2.4
Basalt	2786	3623	4712	5800	7543	9809			

Measurements are required to establish soil profiles for the selected NGNP location. The profiles provide the required input for site response analyses that serve as basis for development of the ground motion response spectra for site-specific earthquake. The results of site response analyses for the iterated strain compatible properties can be used for development of profiles for soil-structure interaction analyses.

Laboratory tests of the alluvial soils are required to establish input soil properties that are needed for the conceptual design such as angle of internal friction, dynamic degradation curves to represent the non-linear strain-stress constitutive relationship.

6.0 References

1. S. J. Payne, "Modeling of the Sedimentary Interbedded Basalt Stratigraphy for the Idaho National Laboratory Probabilistic Seismic Hazard Analysis," INL/EXT-05-01047, April 2006.
2. S. J. Payne, "Development of Soil Design Basis Earthquake (DBE) Parameters for Moderate and High Hazard Facilities at RTC," Rev. 2 INL/EXT-03-00942, April 2006.

APPENDIX C – KAERI Report on RB Response and Source Term

KAERI

Nuclear Hydrogen Project

Calculation Note

Document No: NHDD-KA-08-RD-CD-050

Title : **Analysis of Reactor Containment Building P-T and SRDC-11 source terms with the MELCOR**

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Approved by :	Won-Jae Lee		Date : Sep. 13, 2008

SUMMARY

This report is on a containment building and function study, which is a part of NGNP CDS subtask, WBS NHS.000.S02-Containment building and function Study. The first purpose of the task is to assess the mass and energy discharge into the reactor building and to provide the discharge data for the containment pressure transient analysis. The major events selected for this study are the main steam line break (MSLB), pressure safety valve (PSV) opening and the loss of helium coolant by cross vessel guillotine break for the 600 MWt NGNP. The events are analyzed by the GAMMA+ system thermo-fluid analysis code developed by KAERI. The second purpose is to predict the containment peak pressures from three representative accidents and to estimate the total amount of I - 131 release to the atmosphere for the SRDC - 11 accident using the MELCOR code. The MELCOR code used is the 1.8.5 QZ. MELCOR version that was developed by Sandia National Laboratories for USNRC. The transient boundary conditions such as the mass and energy release rate for the MELCOR code were obtained from the thermo-fluid analysis using the GAMMA+ code. Other input data were provided by GA. Calculation results show that the containment peak pressures are 13 psig and 1.2 psig for the MSLB and PSV opening accidents respectively. For the cross vessel guillotine break accident, the peak pressure was predicted between 210 psig to 180 psig. If the iodine release is in the form of I_2 gas, then it is released to atmosphere completely without any removal in the containment. But, if it is released in the chemical form of CsI, the amount of iodine release to atmosphere is turned out to be one order lower (4~5%) than that of the total I-131 released from the reactor vessel.

Record of Revisions

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NHDD-KA-08-RD-CD-050

NHS.000.S02

Analysis of Reactor Containment Building P-T and SRDC-11 source terms with the MELCOR

September 15, 2008

Containment building and function Study

NGNP Project - Conceptual Design Studies

Korea Atomic Energy Research Institute

1. Pressure Transient analysis

The purpose of this study is to predict the peak pressures from three representative accidents for the NGNP 600 MWt with the MELCOR code. Three accident sequences for the NGNP such as ‘main steam line break (MSLB)’ , ‘cross vessel guillotine break’ and ‘primary safety valve (PSV) opening’ are considered for estimating the transient pressure behaviors with the MELCOR. The MELCOR code used is the version of 1.8.5 QZ. The MELCOR code is a fully integrated, engineering level computer code that models the progression of severe accidents in both LWR and BWR. It was developed by Sandia national laboratories for the USNRC.

The Figure 1.1 shows the pressure transient analytical model for the MELCOR. For each accident sequence, the same analytical model was applied to estimate the pressure transient behavior during the short period of 5 seconds but the boundary conditions for the mass and the energy release rate for both steam and helium and their release locations were different depending on the each accident sequences.

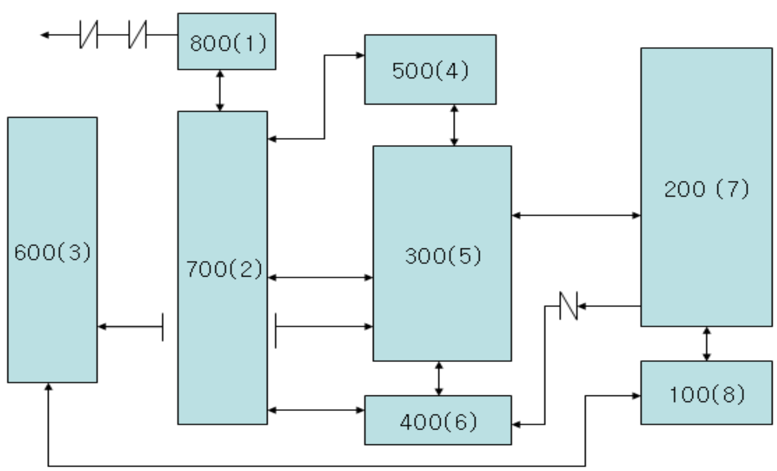

Figure 1.1 Pressure transient analytical model for MELCOR

The dimensional data for all the compartments in RCB such as their volume sizes and flow areas are summarized in the table 1.1 and table 1.2.

Table 1.1 Volume for the compartments in RCB

CV number	Volume[m^3]
S/D cooling system maintenance (100)-8	1245.941
Reactor Cavity Space (200)-7	1529.11
Steam Generator Space (300)-5	1030.73
SG below, around Feed nozzle (400)-6	297.3269
Space above Main circulator (500)-4	622.9705
Equipment shaft space (600)-3	699.426
Vent below Space (700)-2	1030.73
Vent above, above Steam & Feed (800)-1	622.9706

Table 1.2 Flow area among the compartments in RCB

CV number	Flow area [m^2]
From 'S/D cooling' to 'RX cavity' (100→200)	2.48
From' S/D cooling' to 'Eq-shaft' (100→600)	9.47
From 'RX cavity' to 'SG cavity' (200→300)	0.18
From 'RX cavity' to 'SG below space' (200→400)	4.64
From 'SG cavity' to 'SG below space' (300→400)	1.45
From 'SG cavity' to 'above M-circulat' (300→500)	2.48
From 'SG cavity' to 'vent below' (300→700)	0.24
From 'SG below space' vent below' (400→700)	13.93
From 'M-circulator' to 'vent below' (500→700)	1.85
From 'SG Cavity to 'Eq-shaft space' (300→600)	1.39
From 'vent below' to 'vent above' (700→800)	20.43
From 'Vent above' to 'atmosphere' (800→900)	10.40

The important assumptions for this study are as follows. It was assumed that the set point for opening the vent valve is 1 psid between the atmosphere and the vent space (control volume number is 800). If the vent valve opens, thereafter, it was assumed that the valve keeps to be opened over the transient. The atmosphere pressure was assumed to be kept at 14.5 psi. The reverse flow from the 'SG below space' to the 'reactor cavity space' was not allowed.

The mass and energy release rates for both steam and helium were provided by the calculation results from the GAMMA+ code. The dimensional data, the Initial condition for the RCB were provided by GA Company. But these dimensional data were originally for the 450 MWt plant. Therefore, there were modifications of the volume size on some compartment spaces for 600

MWt plant. The SG space, the equipment shaft space and the vent below space were increased in their volume sizes as much as 30% from that of 450 MWt plant respectively.

1.1 Accident analysis for the internal events

The purpose of this analysis is to assess the mass and energy discharge into the reactor building and to provide the discharge data for the containment pressure transient analysis. The major events selected for this purpose are the main steam line break and the loss of helium coolant. The events are analyzed by the GAMMA+ system thermo fluid analysis code.

1.1.1 System analysis model for NGNP 600 MWt with the GAMMA+

The selected system configuration for the internal events is shown at Figure 1.2 The configuration, one among the configurations suggested in “Steam Generator Alternatives Study Summary and Conclusions” by M. Labar, is the worst one since the reactor vessel system is directly affected by the steam line break.

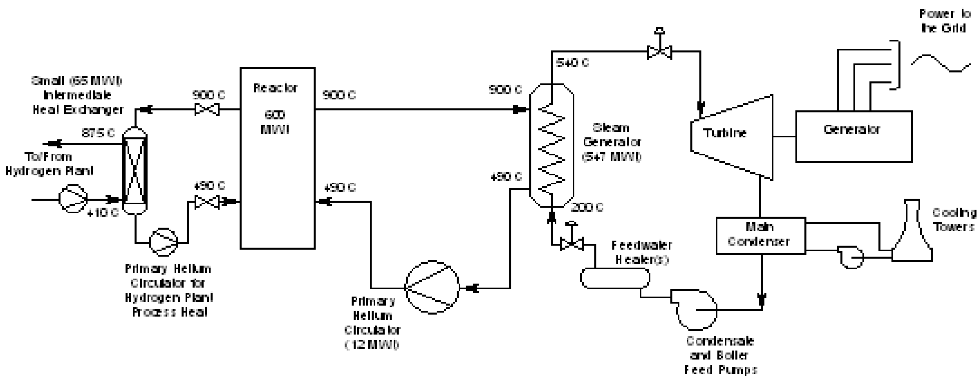

Figure 1.2 NGNP system configuration selected for analysis purpose

The GAMMA+ system nodalization shown at Figure 1.3 is used for the accident analysis of the internal events. The reactor pressure vessel model is most identical to that used in the NGNP CDS Phase B cooled vessel analysis task, except that the internal vessel cooling concept is applied. In order to simulate the loop transient behavior, the SG loop for electricity production and IHX loop for hydrogen production are added based on the geometry and operation data supplied by GA.

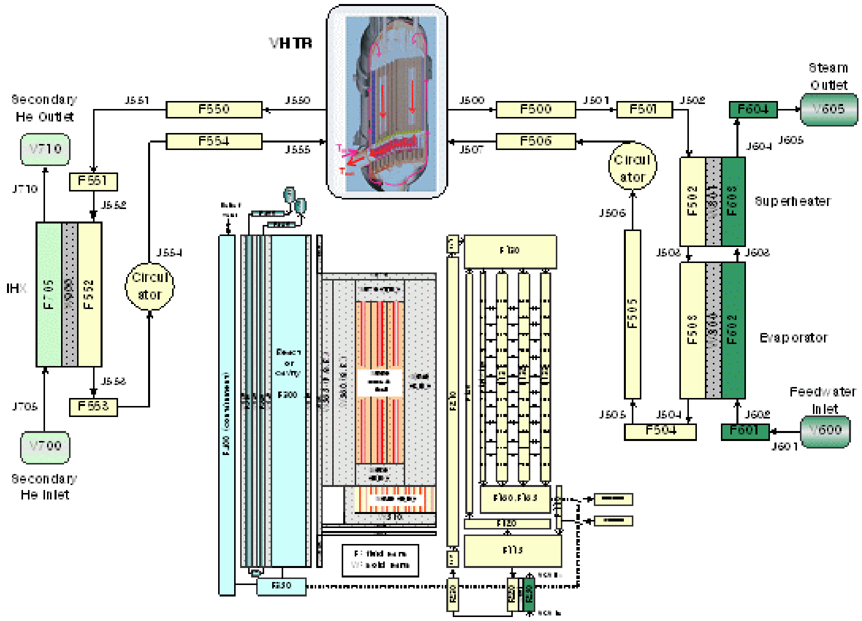

Figure 1.3 System nodalization for analyzing the internal events

1.1.2 Main Steam Line Break (MSLB)

For this event the break area of 0.1662 m2 (ID 0.46 m) is used. The guillotine type pipe break is assumed to occur at the upstream of steam isolation valve and assuming the failure of the check valve closure. The reactor trip is assumed to occur at 1 s with a time delay of 1 s. Following the reactor trip and time delay, the primary circulator flow runs down in 10 seconds but secondary feed water flow runs down linearly in 5 seconds. At the same time the steam isolation valve closes linearly with a 10 s stroke time. Figure 1.4 shows the discharged mass flows from the openings to the steam generator silo. The break flow rates are governed by the size of main steam line and the stroking times of feed water and steam isolation valve. The discharged fluid is kept steam phase during the transient.

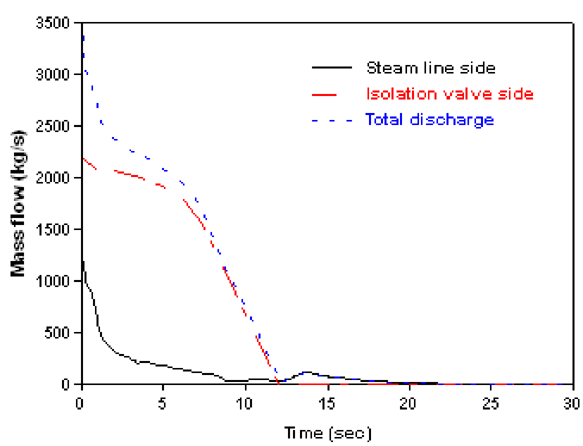

Figure 1.4 break discharges into the steam generator silo (MSLB)

1.1.3 Loss of Helium Leak for PSV Opening

For this event the primary safety valve is assumed to fail open with the break area of 0.0082 m2 (ID 0.1022 m). Helium coolant discharges into the reactor silo (1 atm). The reactor trip occurs at 10.1 s with a time delay of 1 s by low primary pressure signal (6.24 MPa). Following the reactor trip and time delay, the primary circulator flow runs down in 10 seconds but secondary feed water flow runs down linearly in 5 seconds. At the same time the steam isolation valve closes linearly with a 10 s stroke time. Figure 1.5 shows the helium discharge rate into the reactor silo. The fuel temperature reaches maximum (1501^{o}C) at 80 hours shown at Figure 1.6.

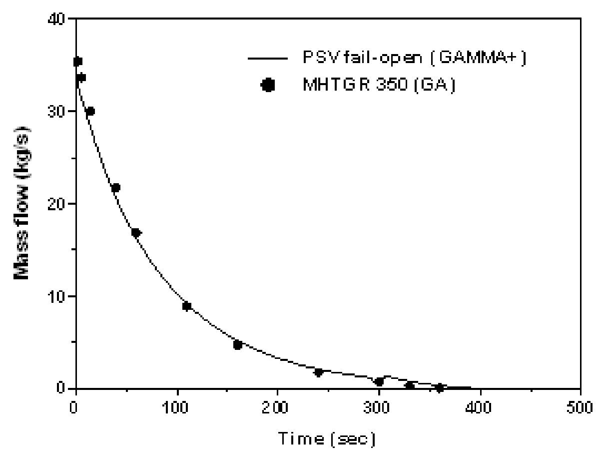

Figure 1.5 break discharges into the reactor silo (PSV opening)

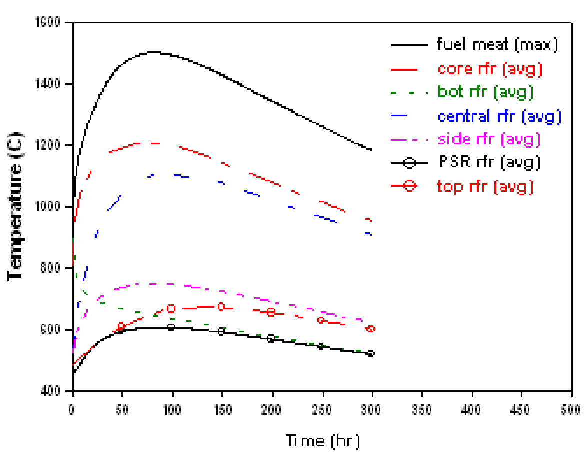

Figure 1.6 core temperature transients (PSV opening)

1.1.4 Loss of Helium Coolant for Cross Vessel Rupture

For this event the cross vessel is assumed to rupture with the break areas of 2.158 m2 at the cold annulus of the cross vessel and 1.606 m2 at the hot pipe of the cross vessel. Helium coolant rapidly discharges into the reactor silo (1 atm). Immediately following the rupture, the reactor trip occurs at 0.005 s with a time delay of 1 s by low primary pressure signal (6.24 MPa). Following the reactor trip and time delay, the primary circulator flow runs down in 10 seconds but secondary feed water flow runs down linearly in 5 seconds. At the same time the steam isolation valve closes linearly with a 10 s stroke time. Figure 1.7 shows the helium discharge rate from the openings to the reactor silo. The fuel temperature reaches maximum (1497oC) at 80 hours shown at Figure 1.8.

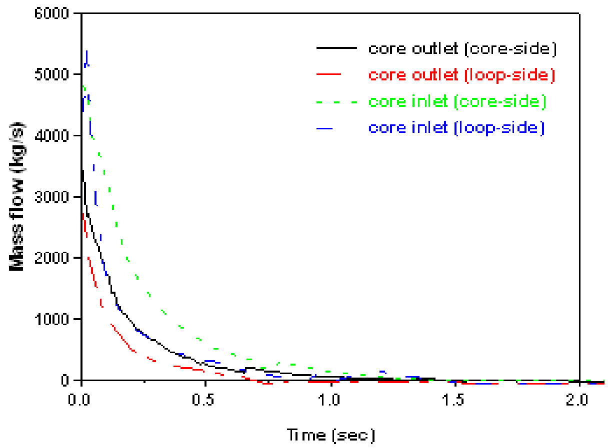

Figure 1.7 break discharges into the reactor silo (cross – vessel rupture)

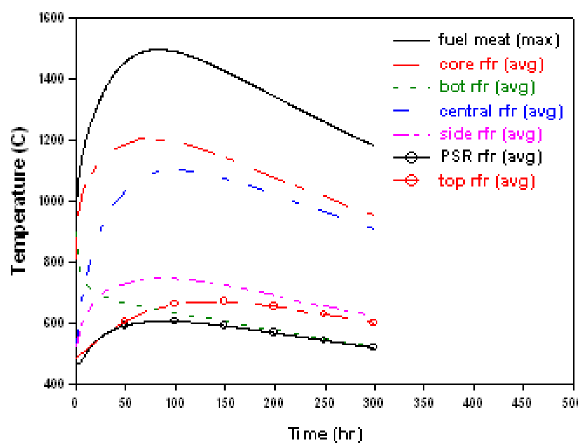

Figure 1.8 core temperature transients (cross vessel rupture)

The complete rupture of the cross vessel considered above belongs to very rare hypothetical event. Therefore more probable but still much less likely failure is considered. The opening areas by cross vessel failure are assumed to be 1.16 m2 at the outer duct of the cross vessel and 0.82 m2 at the inner duct of the cross vessel, as limited by the steam generator vessel constraints. The event sequence is identical to that of the above cross vessel rupture case. Figure 1.9 shows the helium discharge rate from the openings to the reactor silo.

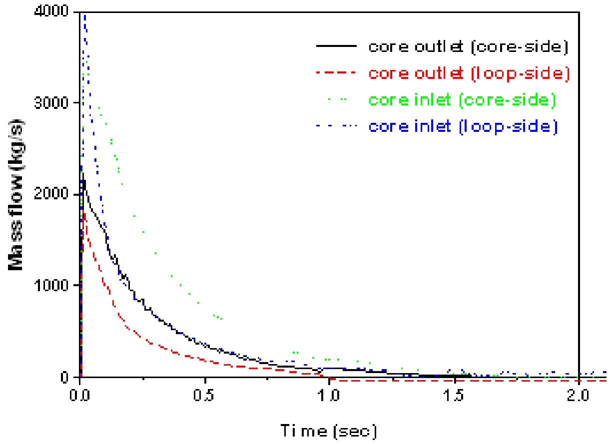

Figure 1.9 break discharges into the reactor silo (cross vessel failure)

1. 2 pressure transient (P - T) analysis with the MELCOR

The purpose of this analysis is to predict the peak pressure for the three accidents such as the 'main steam line break', the 'cross vessel guillotine break' and the 'safety vale fail open'. These P - T calculations are performed with the MELCOR code.

1.2.1 Pressure transient from MSLB

Figure 1.10 shows the calculated pressure transient for the three compartments such as the SG space, the reactor cavity space and the vent below space during 5 seconds. The peak pressure was occurred at the vent below space where the steam is discharged. The peak pressure was 13 psig. The vent area was 10.4 m^2.

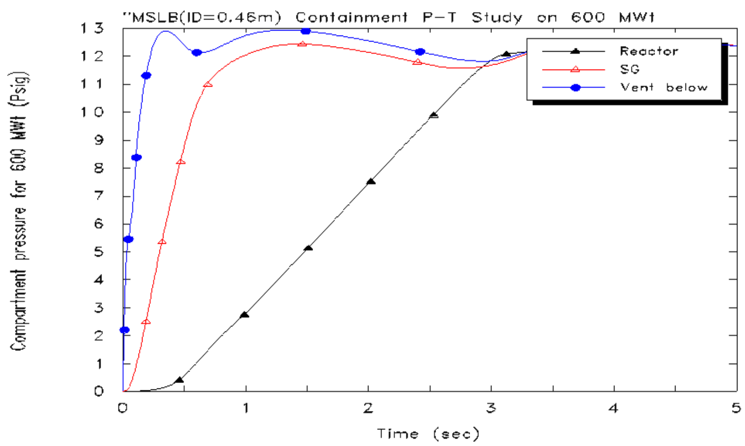

Figure 1.10 pressure transient for SG, reactor and vent below space (MSLB)

1.2.2 Pressure transient from the cross vessel guillotine break (CVGB)

Figure 1.11 shows the calculated pressure transients for the three compartments from CVGB accident with the break area of 3.7 m^2 during 5 seconds. The peak pressure was occurred at the reactor cavity space where the helium is discharged. The pressure was increased to 210 psig with maximum. The vent area was 10.4 m^2.

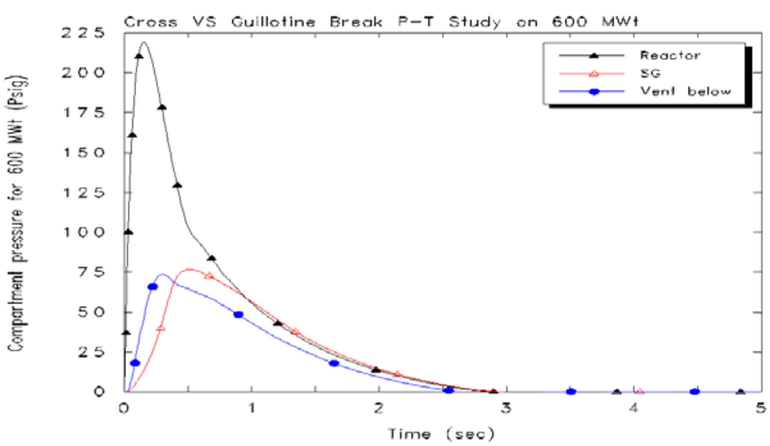

Figure 1.11 Pressure transient for SG, Reactor and vent below space with the break area of 3.7 m^2 (CVGB)

Figure 1.12 shows the sensitivity calculation for the pressure transient from CVGB that the break area was reduced to 2.2 m^2. The peak pressure was occurred at the reactor cavity space where the helium is discharged. But the pressure was increased to 180 psig with maximum. The vent area was 10.4 m^2.

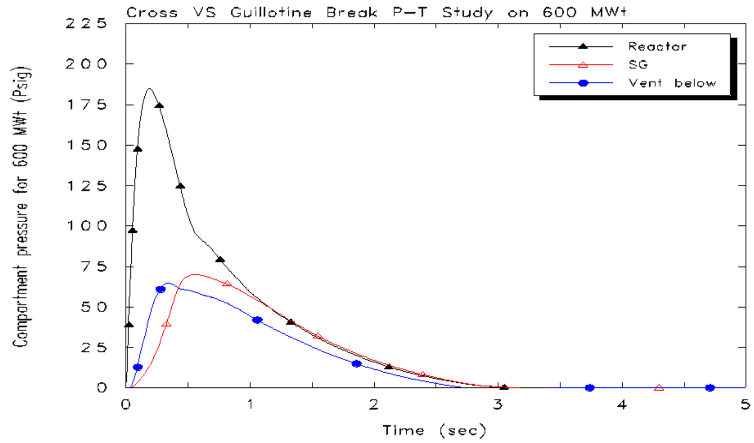

Figure 1.12 Pressure transient for SG, Reactor and vent below space with the break area of 2.2 m^2 (CVGB)

1.2.3 Pressure transient from the PSV opening

Figure 1.13 shows the calculated pressure transients for the three compartments from PSV opening accident with the break area of 0.32 cm^2 during 5 seconds. The peak pressure was occurred at the SG cavity space. The peak pressure was 1.2 psig. The vent area was 10.4 m^2.

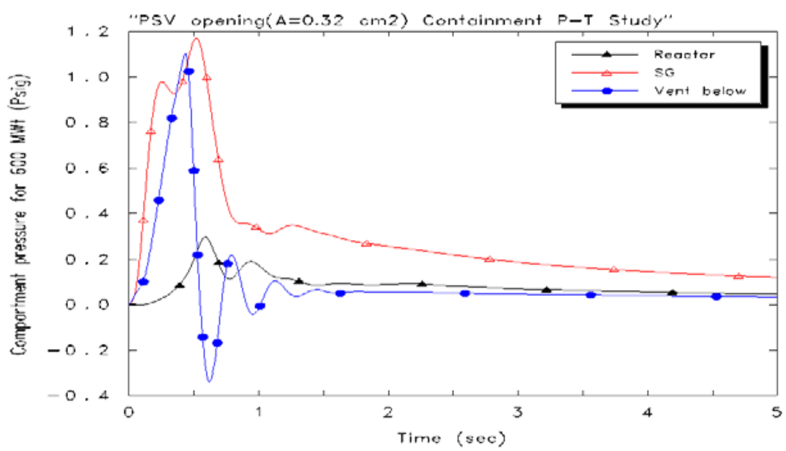

Figure 1.13 Pressure transient for SG, Reactor and vent below space with the break area of 0.32 cm^2 (PSV opening)

2. **SRDC - 11 Transient Source Terms Analysis with the MELCOR**

The purpose of this study is to estimate the total amount of I - 131 release activity from the reactor cavity space in RCB to the atmosphere in case of the SRDC - 11 accident for the 600 MWt plant with the MELCOR. The same 'system analytical model' as that for the pressure transient analysis was used to predict the source terms from I - 131 for SRDC - 11 accident with the MELCOR. Figure 2.1 shows the system analytical model to estimate the release activity of I - 131 to the atmosphere from SRDC - 11 accident with the MELCOR.

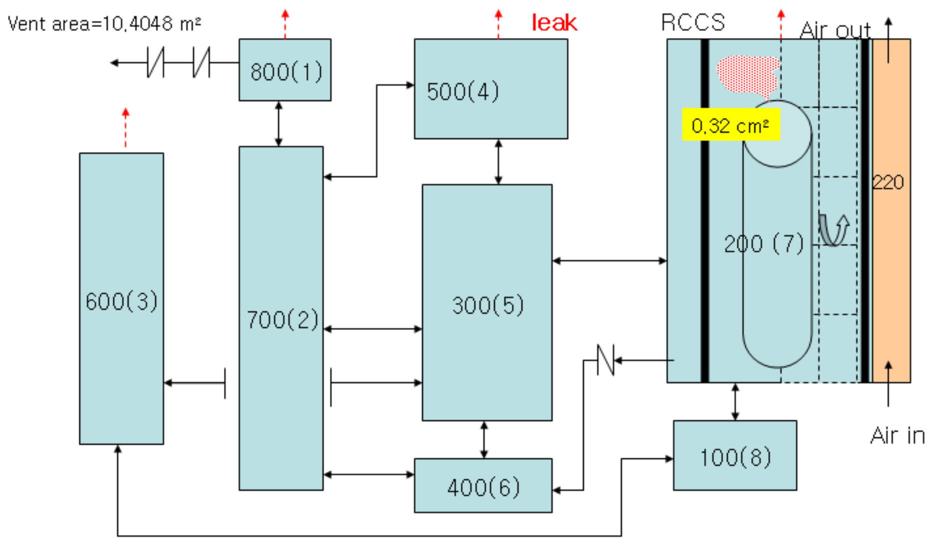

Figure 2.1 System analytical model for SRDC - 11 accident with the MELCOR

But it was required that two features should be included into the above mentioned 'system analytical model' additionally. The first feature was the inclusion of the RCCS (reactor cavity cooling system). RCCS is the cooling panel and it surrounds the reactor vessel over its full circumference and length to remove some heat from the surface of the reactor vessel. Therefore it was expected that the RCCS can play an important role not only on the thermal hydraulic parameter but also on the amount of fission product deposition.

The second feature was the implementation of phenomena that helium leaks from the four compartment spaces to the atmosphere. The compartment spaces for leaking were the 'cavity space', the 'SG above space', the 'vent space' and the 'equipment shaft space' respectively. For this source terms study, two type of leak rates were considered. The first leak rate was defined as the 100% of the RCB volume should be released to atmosphere during one day under the condition of 25 psi. From now on, this leak rate will be called as the100 v/0 leak. The other type of leak rate was defined as the 30% of the RCB volume should be released to atmosphere during one day under the condition of 25 psi. From now on, this leak rate will be called as the30 v/0 leak.

The actual modeling of these leak rates for the MELCOR code were reappeared by the determination of the corresponding total leak area from the four compartment spaces. The total helium flow rate for the leak type of 100 v/0 can be derived based on the volume size of RCB and the time of a day. Two volumes were modeled with the MELCOR for finding out the corresponding flow area. It was assumed that one volume is under the constant pressure of 25 psi with the same volume size as that of RCB and the other volume (=atmosphere) is under the constant pressure of 14.5 psi. Then these two volumes were connected with the flow path that the flow area can be defined by user. Then, the total leak area corresponding to the helium flow rate for 100 v/0 leak can be found out easily. The partition of total flow area for the '100 v/0 leak' into the four different compartments was done based on the partition fractions given by GA.

Table 2.1 Leak areas from 4 compartments under 100 v/0 and 30 v/0 leak

	100 vol/0 leak per day Area [m^2]	30 vol/0 leak per day Area [m^2]
RX cavity (200)	8.1656E-05	3.287E-05
SG above (500)	2.9736E-05	1.197E-05
Vent above (800)	8.968E-05	3.61E-05
Eq-shaft (600)	2.70928E-04	1.0906E-04
Total area	0.472E-03	0.19E-3

Before the TMI - 2 accident, the most of the operating reactors for PWR were designed and licensed based on the 'TID - 14844 source term'. In TID - 14844, the physical form of the released iodine was considered as a gas as like a noble gases. But following the TMI - 2 accident, it has been considered that the iodine release did not closely follow the pattern that might be expected based on the TID - 14844. Therefore, it has been considered that the most of the iodine is presumed to be release to the containment as particulate but 5% is taken to be gaseous. Although the iodine forms in VHTR might be different from that of PWR with considering its high operating temperature of around 1000 °C and its different chemical condition under the helium gas, this is the state of art on the known form of release iodine until now. Therefore, to apply the exact chemical form of iodine, it will be necessary to acquire much more experimental evidences on the chemical form of iodine in a VHTR.

Based on the above mentioned backgrounds concerning the chemical form of the released iodine, two different forms of iodine are accounted for simulating the transport of iodine up to the atmosphere. The first form was the molecular iodine (=I_2), which exists as a gas over the transient. The second form was the particulate iodine (=CsI), which can exist as an aerosol or a gas depending on its vapor pressure value as a function of the surface temperature. Table 2.2

are summarized the initial and dimensional data for simulating the NGNP 600 MWt system with the MELCOR.

Table 2.2 the initial and dimensional data for NGNP 600 MWt system with the MELCOR.

Control Volume number	Atmosphere Temperature [K]	Volume [m^3]	Deposit area [m^2]
100	297.15	1245.9	126.72
200	423.0	1529.5	7305.6
300	311.15	1030.7	565.76
400	297.15	297.3	227.46
500	313.15	622.9	445.48
600	316.15	699.4	501.44
700	316.15	1030.7	670.58
800	316.15	622.9	314.9
220(atm)	313.15	150	NA
900(atm)	313.15	NA	NA

Total cumulative I - 131 release activity was 65.5 Ci at 100 hrs. The amount of I - 131 release was turned out to be too small to open the vent valve. The release rate of I - 131 activity should be converted to the mass release rate of I - 131 for the MELCOR input deck. This conversion was performed based on the I - 131 specific activity of 1.24×10^8 [Ci/kg]. The helium release rate from the leak area of 0.32 cm^2 on the reactor vessel was derived using the MELCOR.

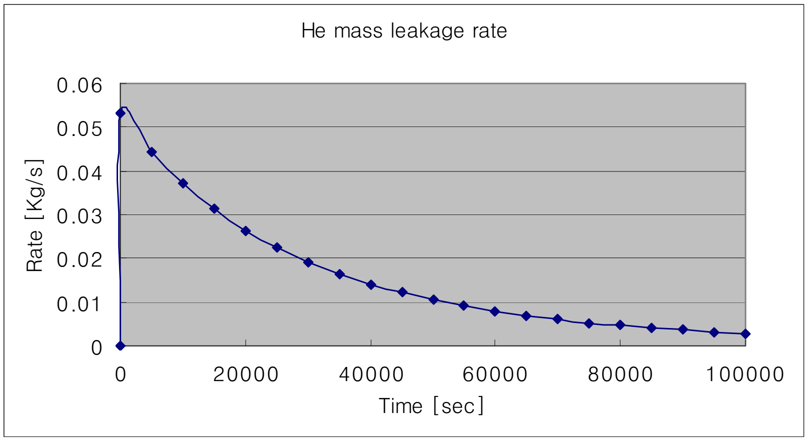

Figure 2.2 helium mass release rate for SRDC - 11 accident with the MELCOR

The one volume that was pressurized with 70 bars and has the same volume size as that of reactor vessel was modeled to derive the helium release rate under the leak area of 0.32 cm^2. Figure 2.2 and 2.3 shows the applied helium release rate from the reactor vessel to the reactor cavity space and its temperature.

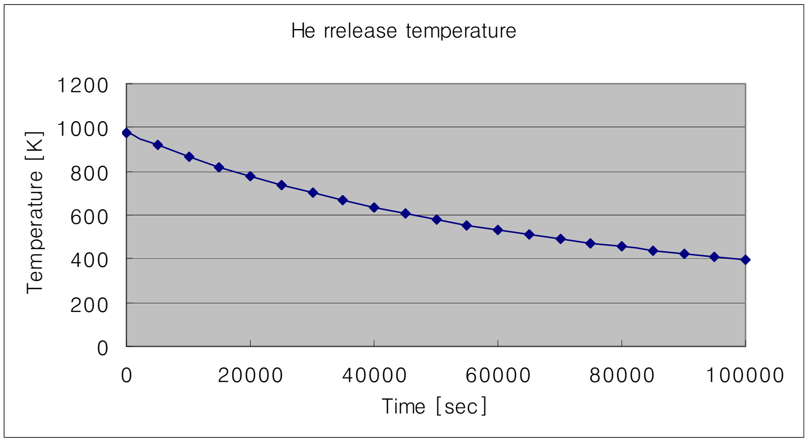

Figure 2.3 release helium temperature for SRDC - 11 accident with the MELCOR

The calculation was performed up to 1000 hrs. Figure 2.4 showed the results predicted from the MELCOR with the assumption of CsI form under the condition of the ‘100 v/0 leak’. In the initial phase, most of the released iodine suspended until 4 hours. But after 4 hours, the deposition starts to become the dominant phenomena. The deposit phenomena continue to 100 hours. The total amount of release I - 131 activity was predicted as 2.95 Ci and it corresponds to the 4.8% of the total I - 131 activity being released from the reactor vessel.

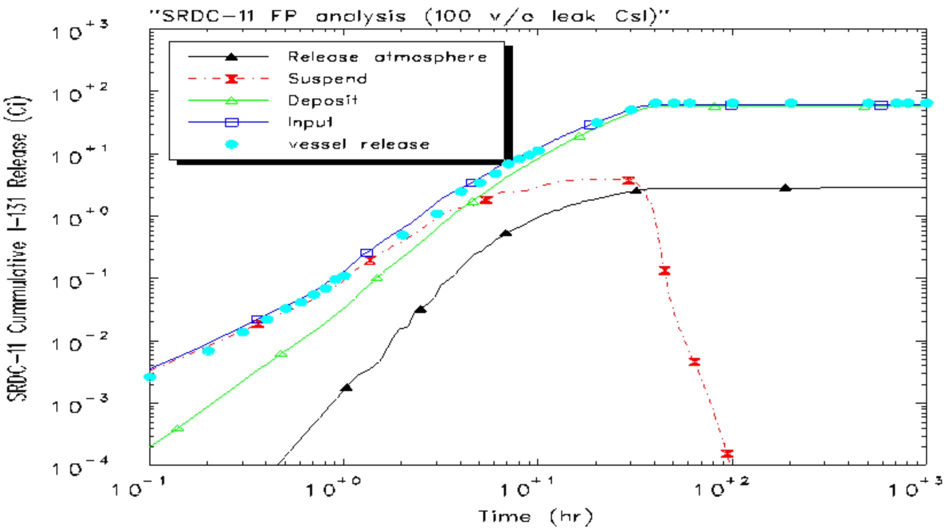

Figure 2.4 predicted I-131 release from SRDC - 11 with the MELCOR (100 v/0 leak CsI)

Table 2.3 summarized the distribution of I - 131 over the system at 1000 hours. More than half of the I - 131 was removed in the reactor cavity by deposition on the surface of reactor vessel, RCCS and containment wall. The second largest removal was occurred in the SG cavity.

Table 2.3 Distribution of I-131 activity for SRDC-11 at 1000 hrs (100 v/0, CsI)

Control volume number	Suspend activity [%]	Deposit activity [%]
100	0	3.76
200	0	67.24
300	0	10.26
400	0	2.01
500	0	3.26
600	0	3.20
700	0	4.44
800	0	0.97
900(atm)	4.84	

Figure 2.5 showed the results predicted from the MELCOR with the assumption of I_2 (gas) form under the condition of the ‘30 v/0 leak. All the released iodine suspended until 0.5 hours. But after 0.5 hours, the suspended I_2 gas starts to release to the atmosphere. The release of I_2 gas continued to 1000 hours.

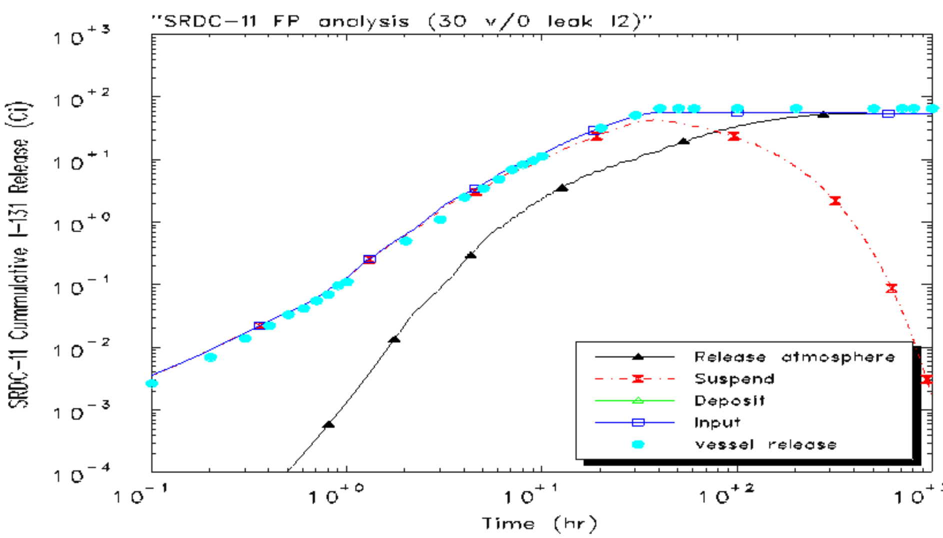

Figure 2.5 predicted I-131 release from SRDC - 11 with the MELCOR (30 v/0 leak I_2)

Consequently, all the release I_2 gas was released to the atmosphere completely. It corresponds to the 100% of the total I - 131 activity being released from the reactor vessel. However if the helium release did not continue until 1000 hr, then the suspended I_2 gas might be confined in the RCB. Table 2.4 summarized the distribution of I - 131 over the system at 1000 hours for 30 v/0 leak and with assumption of I_2 chemical form.

Table 2.4 Distribution of I-131 activity for SRDC-11 at 1000 hrs (30 v/0, I_2)

Control volume number	Suspend activity [%]	Deposit activity [%]
100	0	3.76
200	0	67.24
300	0	10.26
400	0	2.01
500	0	3.26
600	0	3.20
700	0	4.44
800	0	0.97
900(atm)	4.84	

3. **Summary and conclusions**

From the P-T analyses results for NGNP 600 MWt with the MELCOR, 'PSV opening' were satisfied with the goal (below 9 psig) under the current NGNP 600 MWt design and the release data provided from the GAMMA+ using the MELCOR1.8.5 QZ. But the MSLB (ID=0.46) of peak pressure 13 psig was not satisfied the goal. It needs a reduction of the peak pressure. The Guillotine break rupture at cross vessel were not satisfied the goal and the peak pressures were predicted as 180~210 psig for the break area of 2.2 m^2 and 3.7 m^2 respectively.

If the iodine release as I_2 gas, then it remains as a gas over the transient. The helium release from the reactor vessel continued to 1000 hrs, therefore, the suspended I_2 gas was released to the atmosphere completely without any removal in the RCB such as a condensation or a settling. But if the helium release to the reactor cavity did not continue to 1000 hrs, then suspended I_2 gas will stop the release to the atmosphere.

But with the assumption of the CsI chemical form, the amount of iodine release to atmosphere turned out to be one order lower (4~5%) than that of the total I-131 released from RV.

APPENDIX D – KAERI Report on Assessment of Air Ingress

KAERI

Nuclear Hydrogen Project

Calculation Note

Document No: NHDD-RD-CA-08-008

Title : Air-ingress Analysis for NGNP Reactor

Prepared by :	Nam-il Tak	Nam-Il Tak	Date : Sep. 10, 2008
Reviewed by :	Hong-Sik Lim		Date : Sep. 11, 2008
Approved by :	Won Jae Lee		Date : Sep. 12, 2008

SUMMARY

This document reports the air-ingress analysis for the NGNP which is a part of NGNP CDS Subtask WBS NHS.000.S02 - Reactor Containment, Embedment Depth, and Building Functions. The purpose of the present analysis is to perform the system thermo-fluid and graphite oxidation calculations during air-ingress events of the NGNP reactor. The potential threat posed by air-ingress lies in the chemical reaction of oxygen with hot graphite which causes reaction heat as well as graphite corrosion. Two methodologies (i.e., containment refreshment and containment vent) were applied to model the air supply into the containment. Based on the two methodologies, five cases were analyzed to provide conservative results as well as best-estimate ones. The GAMMA+ code developed by KAERI was adopted. The results of the GAMMA+ calculations show that massive air-ingress is delayed up to ~550 hr for the considered design having the internal flow path for the vessel cooling. It is also found that chemical reaction mainly attacks the lower part of the active core and the lower reflector due to the significantly delayed air-ingress. The total corroded volume of graphite is predicted no more than 3% until 900 hr.

Record of Revisions

No.	Date	Description	Prepared by
00	Sep. 10, 2008	Initial Issue	Nam-il Tak

NHDD-RD-CA-08-008

Air-ingress Analysis for NGNP Reactor

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WBS NHS.000.S02. Reactor Containment, Embedment Depth, and Building Functions

NGNP Phase B Conceptual Design Studies

Korea Atomic Energy Research Institute

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1. INTRODUCTION

The present work is performed under NGNP Conceptual Design Studies Subtask NHS.000.S02 - Reactor Containment, Embedment Depth, and Building Functions. The purpose of the present analysis is to perform the system thermo-fluid and graphite oxidation calculations during an air-ingress event of the NGNP reactor. The results of the present analysis will be used to calculate additional source term release by the air-ingress, and finally assess the effect of the air-ingress on the dose rate.

2. GEOMETRY OF REACTOR CORE AND CONTAINMENT

The reference system configuration for the present work is shown in Fig. 2.1. The configuration was proposed by the previous NGNP study [1].

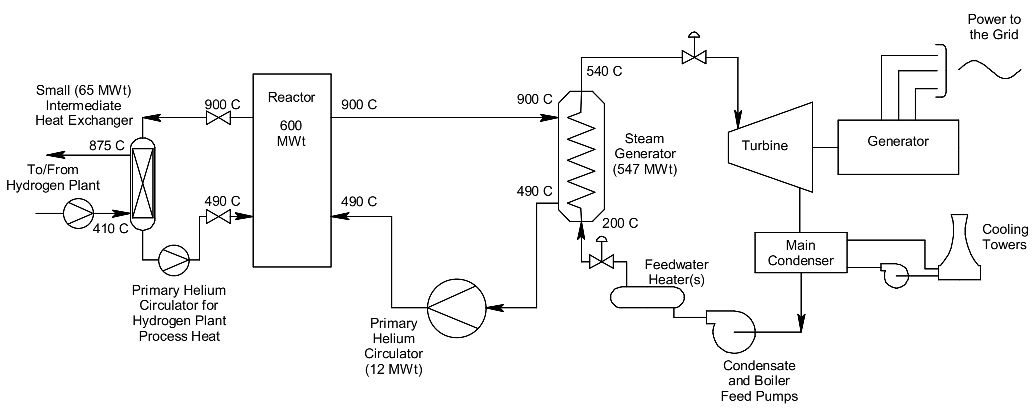

Fig. 2.1 The reference system configuration.

Most of the reactor design parameters used in the previous study of KAERI [2] are kept in this work. That is, based on the design parameters of GT-MHR [3], some modifications are made to reflect the recent NGNP studies. For example, the coolant inlet/outlet temperatures are modified to 490^{o}C and 900^{o}C, respectively. The geometries of the coaxial pipes are also modified by the General Atomics (GA). The inner and outer diameters of the coaxial pipe connected with the steam generator are 1.58 and 2.29 m, respectively. The natural pathway for the vessel cooling system (VCS) is considered. Figs. 2.2 and 2.3 show the natural pathway recently proposed. A small portion of the inlet coolant flow is bypassed at the lower plenum, cooled down by the shutdown cooling system (SCS) heat exchanger, flows through the annular space between the core barrel and the reactor vessel, and mixes with main coolant flow at the upper plenum after all. The coolant pressure drop across the core under the normal operating condition is 60 kPa.

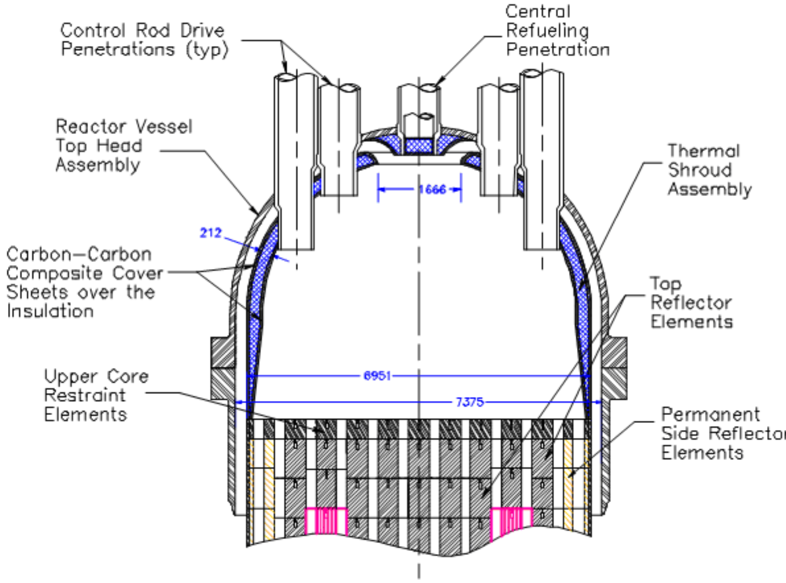

Fig. 2.2 The schematics of the upper part of the NGNP reactor [4].

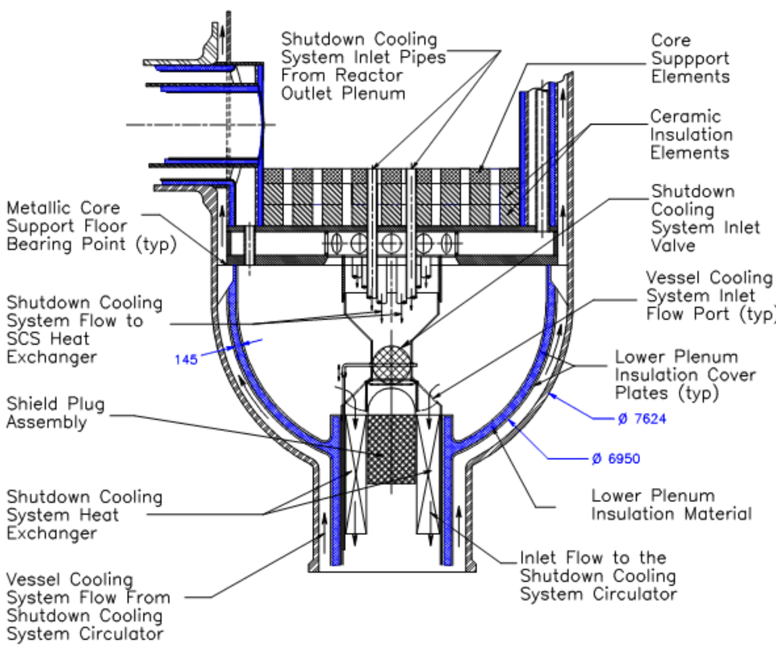

Fig. 2.3 The schematics of the lower part of the NGNP reactor [4].

The vented low pressure containment (VLPC) is adopted in this work. This concept was studied for the GA designs of 350 MWth MHTGR [5] and 450 MWth MHTGR [6]. The VLPC is a normally closed space equipped with a vent. The vent will open if the pressure inside the VLPC exceeds its design setpoint. It protects the integrity of the containment and the reactor cavity cooling system (RCCS) against the discharged mass and energy during an initial blowdown phase after pipe break accidents. Although the vent allows the prompt release of fission products to environment, the release of associated gasses early in the accident removes the driving pressure which accelerates the release of the delayed source term out of the containment. The vent is designed to be closed following a transient.

Fig. 2.4 shows the damper type of a containment vent adopted in the 450 MWth MHTGR. Fission products which are released into the containment are reduced by plateout and deposition before release to the environment via the vent path. The vent dampers are maintained in a closed position by gravity, and the weight of the damper plate determines the relief setpoint pressure, which is the internal pressure needed to open the damper.

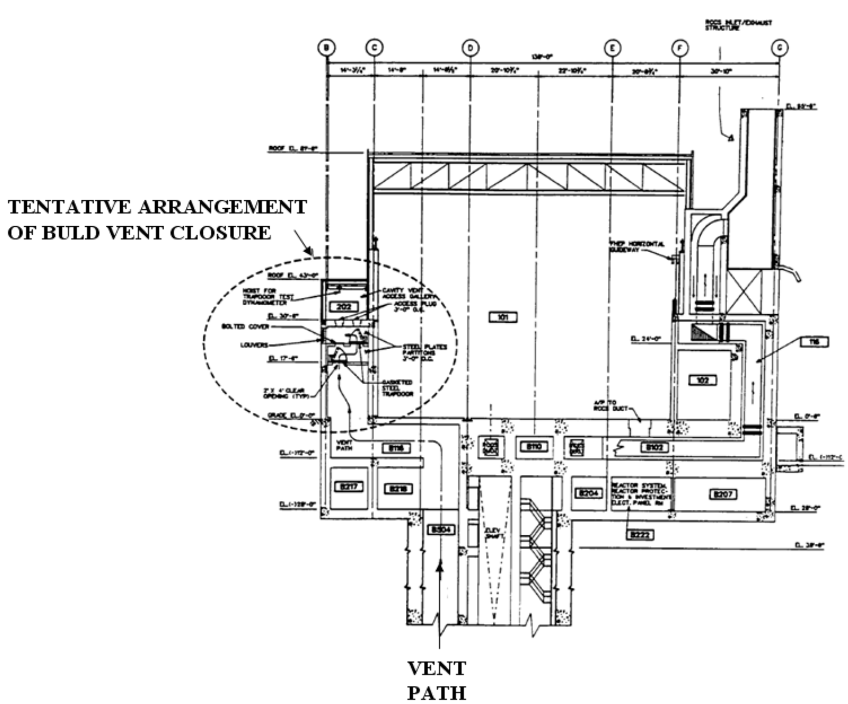

Fig. 2.4 Vent pathway for 450 MWth MHTGR with VLPC [6].

Fig. 2.5 shows the containment model for the pressure transient analysis of 450 MWth MHTGR. In the event of a large primary or secondary coolant discharge, gas or steam is able to flow from any compartment through the building and exits through the vent (relief valve or damper) to the atmosphere.

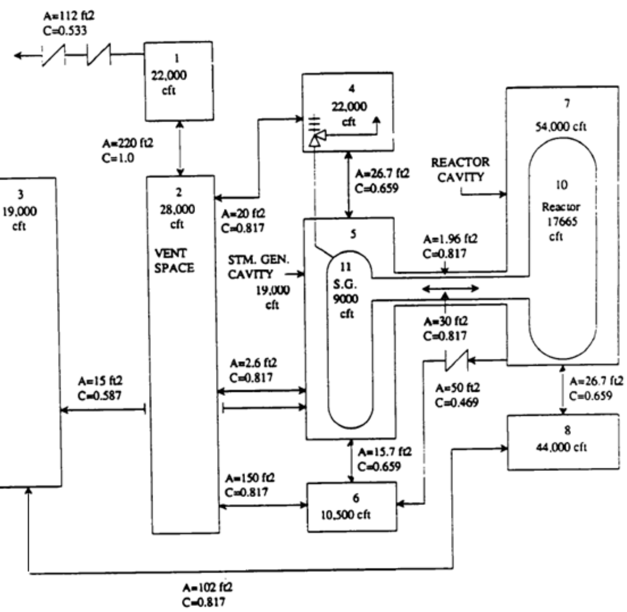

Fig. 2.5 Containment model of 450 MWth MHTGR for pressure transient analysis [6].

In the present work, the containment is modeled as single volume instead of modeling the detailed compartments. The containment volume of the 450 MWth MHTGR is adopted in this work. The parameters related with the containment vent are supplied by the GA. They are summarized in Table 2.1.

Table 2.1 Major parameters used for the containment

Parameters	Value
Containment volume (m^3)	6200
Containment leakage rate	1 volume per day
Vent opening setpoint	7 kPa
Vent area (m^2) - In case of damper - In case of relief valve	 10 0.0082
Vent open & close stroking time (1/s)	10
Environment air temperature (°C)	43[a)]

[a)] Matched with air temperature entering the RCCS.

3. ANALYSIS MODELS AND METHODOLOGIES

The GAMMA code [7] was developed for the analysis of VHTR thermo-fluid transients including air ingress phenomena. The code capability has been significantly extended in the GAMMA+ code, particularly for the following models; fluid transport and material properties, multi-dimensional heat conduction, multi-dimensional fluid flow, chemical reactions, multi-component molecular diffusion, fluid heat transfer and pressure drop, heat generation and dissipation, and radiation heat transfer.

3.1 Governing Equations

In the GAMMA+ code, the fluid flow and heat transport under air-ingress are described by the following conservation equations.

For the chemically reacting gases:

$$\frac{\partial \varphi\rho}{\partial t}+\frac{\partial \rho u_i}{\partial x_i}=\varphi\sum_s R_s \tag{3-1}$$

$$\frac{1}{\varphi}\frac{\partial \rho u_j}{\partial t}+\frac{u_i}{\varphi^2}\frac{\partial \rho u_j}{\partial x_i}=B_j-\frac{\partial P}{\partial x_j}+\frac{1}{\varphi}\frac{\partial}{\partial x_i}\left[\mu\left(\frac{\partial u_i}{\partial x_j}+\frac{\partial u_j}{\partial x_i}\right)\right]+\rho g_j \tag{3-2}$$

$$\frac{\partial \varphi\rho H}{\partial t}+\frac{\partial \varphi\rho u_i H}{\partial x_i}=\frac{\partial}{\partial x_i}\left(\varphi\lambda\frac{\partial T}{\partial x_i}\right)-\frac{\partial}{\partial x_i}\left(\varphi\sum_{s=1}^{m}H_s J_{si}\right)+\dot{q}_{ch}'''+\dot{q}_{sf}''' \tag{3-3}$$

$$\frac{\partial \varphi\rho Y_s}{\partial t}+\frac{\partial \varphi\rho u_i Y_s}{\partial x_i}=-\frac{\partial \varphi J_{si}}{\partial x_i}+\varphi R_s \tag{3-4}$$

For the solid parts:

$$\left(\rho C_p\right)_f\frac{\partial T_f}{\partial t}=\frac{1}{\xi}\frac{\partial}{\partial r}\left(\xi\lambda_f\frac{\partial T_f}{\partial r}\right)+\dot{q}_N'''-\dot{q}_{gf}''' \tag{3-5}$$

$$\left[(1-\varphi)\varphi_g\left(\rho C\right)_w\right]\frac{\partial T_w}{\partial t}=\frac{\partial}{\partial x_i}\left(\lambda_{eff}\frac{\partial T_w}{\partial x_i}\right)+\dot{q}_{het}'''-\dot{q}_{sf}'''+\dot{q}_{gf}''' \tag{3-6}$$

Eqs. (3.1) ~ (3-4) describe the spatially-averaged conservation equations for continuity, momentum, energy of the gas mixture, and the mass of each species, respectively. In Eqs. (3-1) and (3-4), the species generation and dissipation rate (R_s) due to chemical reaction are composed of the two parts, i.e., (1) the gas-phase homogeneous reaction describing CO combustion and (2) the heterogeneous reaction describing the graphite oxidation. The present analysis adopts the graphite oxidation model supplied by the GA, which will be described in detail in the next section. Eqs. (3-5) and (3-6) describe the heat conduction equations to solve the temperature distributions at the fuels, the graphite blocks, and the solid structures. These include the volumetric nuclear heat production ($\dot{q}_N'''$), the heat generation and dissipation rate ($\dot{q}_{het}'''$) due to the graphite oxidation, the heat exchange term ($\dot{q}_{sf}'''$) between the fluid and the solid part, and the additional heat exchange term ($\dot{q}_{gf}'''$) between the fueled zone and the unfueled graphite zone. More detailed descriptions of the numerical approaches of the GAMMA+ code can be referred in [7].

3.2 Graphite Oxidation Model

The reaction of oxygen in the air with the hot graphite can be described by:

$$C+(1-f_{CO}/2)O_2 \rightarrow f_{CO}\,CO+(1-f_{CO})CO_2 \tag{3.7}$$

where f_{CO} is the number of moles of CO formed per mole of C reacting. It has the relation with the CO/CO_2 molar ratio, R_{CO/CO_2} as follows:

$$f_{CO}=\frac{R_{CO/CO_2}}{1+R_{CO/CO_2}} \tag{3.8}$$

The quantity R_{CO/CO_2} depends on the temperature as [8]:

$$R_{CO/CO_2}=2512\exp(-6240/T_G) \tag{3.9}$$

where T_G is the graphite temperature in K.

Based on Eq. (3.7), the graphite mass-loss rate, $\dot{m}_C$ can be obtained by:

$$\dot{m}_C=\frac{1}{1-0.5f_{CO}}\frac{W_C}{W_{O_2}}A_w\dot{m}_{O_2,w}'' \tag{3.10}$$

where W_C is the molecular weight of graphite, W_{O_2} is the molecular weight of O_2, A_w is external area exposed to air, and $\dot{m}_{O_2,w}''$ is the mass flux of O_2 at the wall. The mass flux of O_2 at the wall is evaluated using Fick's law.

$$\dot{m}_{O_2,w}''=-\phi\rho D_{O_2}\left.\frac{dY_{O_2}}{dx}\right|_{x=0} \tag{3.11}$$

where ϕ is the tortuosity coefficient for diffusion in graphite ($\cong 0.01$), D_{O_2} is the binary diffusion coefficient for oxygen in nitrogen, ρ is the density of the gas mixture, and Y_{O_2} is the pore-volume-averaged mass fraction of O_2.

In order to evaluate Eq. (3.11), the species-conservation equation for O_2 in the graphite pore needs to be solved. With a quasi-steady approximation, the species-conservation equation is given for a slab geometry as follows [8]:

$$\frac{d^2 Y_{O_2}}{dx^2} = \delta \xi_{O_2} Y_{O_2}{}^n \tag{3.12}$$

where, the parameters δ and ξ_{O_2} are defined as:

$$\delta = \frac{F_c F_b k}{\phi D_{O_2} Y_{O_2}^n} \tag{3.13}$$

$$\xi_{O_2} = (1 - f_{CO}/2)\frac{W_{O_2}}{W_C}\frac{\rho_G}{\varepsilon\rho} \tag{3.14}$$

where F_c = catalysis factor to account for possible enhancement of the chemical reactivity by impurities, F_b = burnoff factor to account for the effect on chemical reactivity, ρ_G = graphite density, and ε = graphite void fraction (0.21 for H-451), and k = graphite intrinsic oxidation rate at zero burnoff for kinetically-controlled oxidation. For H-451 graphite, the intrinsic oxidation rate can be described as [8]

$$k = 7130 \exp(-20130/T_G)\, p_{O_2}^n \tag{3.15}$$

where p_{O_2} = oxygen partial pressure in atm, n = order of reaction with respect to p_{O_2} ($\cong 0.5$). In Eq. (3.15), the unit of k is s^{-1}. The burnoff factor for H-451 graphite can be obtained by [9]:

$$F_b = [1 - 70\log(1-b)]^{0.5} \tag{3.16}$$

where b is the fractional burnoff.

By integrating Eq. (3.12), it can be shown as

$$Y_{O_2}(x) = Y_{O_2,w}\left[1 - x(1-n)\left\{\frac{\delta\xi_{O_2} Y_{O_2,w}^{n-1}}{2n+2}\right\}^{0.5}\right]^{\frac{2}{1-n}} \tag{3.17}$$

where, $Y_{O_2,w}$ is the O_2 mass fraction at the wall (x = 0), which can be obtained by applying the convective boundary condition at the graphite-coolant channel interface:

$$\dot{m}''_{O_2,w} = -\phi \rho D_{O_2} \left.\frac{dY_{O_2}}{dx}\right|_{x=0} = \rho\beta\left(Y_{O_2,f} - Y_{O_2,w}\right) \qquad (3.18)$$

where, β is the convective mass transfer coefficient and $Y_{O_2,f}$ is the O_2 mass fraction in the bulk fluid. Using Eq. (3.17) and the definition of the Sherwood number, $Sh = \beta D_h / D_{O_2}$, where D_h is the hydraulic diameter, Eq. (3.18) can be rearranged by

$$\phi D_h \left[\frac{2\delta\xi_{O_2} Y_{O_2,w}^{n+1}}{n+1}\right]^{0.5} = Sh(Y_{O_2,f} - Y_{O_2,w}) \qquad (3.19)$$

Fig. 3.1 shows the oxygen mass fraction profile evaluated by Eq. (3.17) for H-451 graphite.

The oxygen penetration depth (x^*) is obtained from Eq. (3.17) by setting $Y_{O_2}(x) = 0$ and solving for x:

$$x^* = \left(\frac{1}{1-n}\right)\left(\frac{2n+2}{\delta\xi_{O_2} Y_{O_2,w}{}^{n-1}}\right)^{0.5} \qquad (3.20)$$

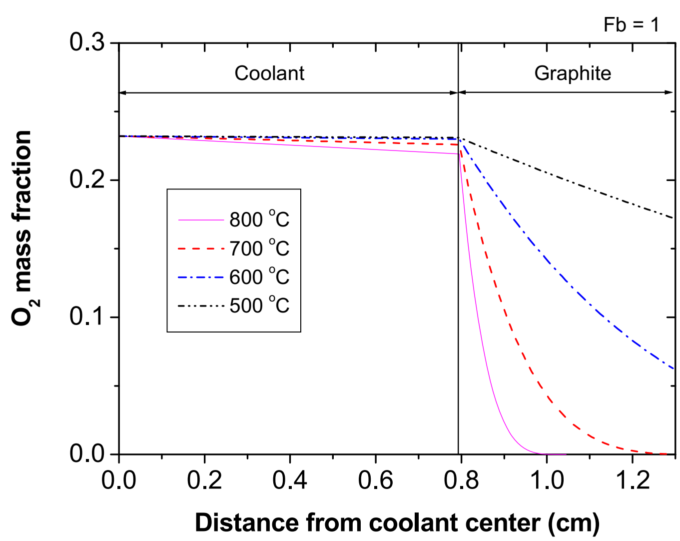

Fig. 3.1 Oxygen mass fraction profile for H-451 graphite.

The dissipation/generation rates for each gas are obtained from Eqs. (3.7) and (3.10). It is assumed that the heat generation due to the blowing effect ($=\dot{m}\times$ the specific enthalpy of graphite) is transferred to the fluid and the remaining exothermic reaction heat is deposited in the graphite.

3.3 System Model for Air-ingress Analysis

The input for the present GAMMA+ simulation is mainly based on the previous work [2]. The modifications are made to consider the natural path way for the VCS flow and the containment. The SCS is also considered to cool down the VCS flow. The system model for the GAMMA+ simulation is shown in Fig. 3.2.

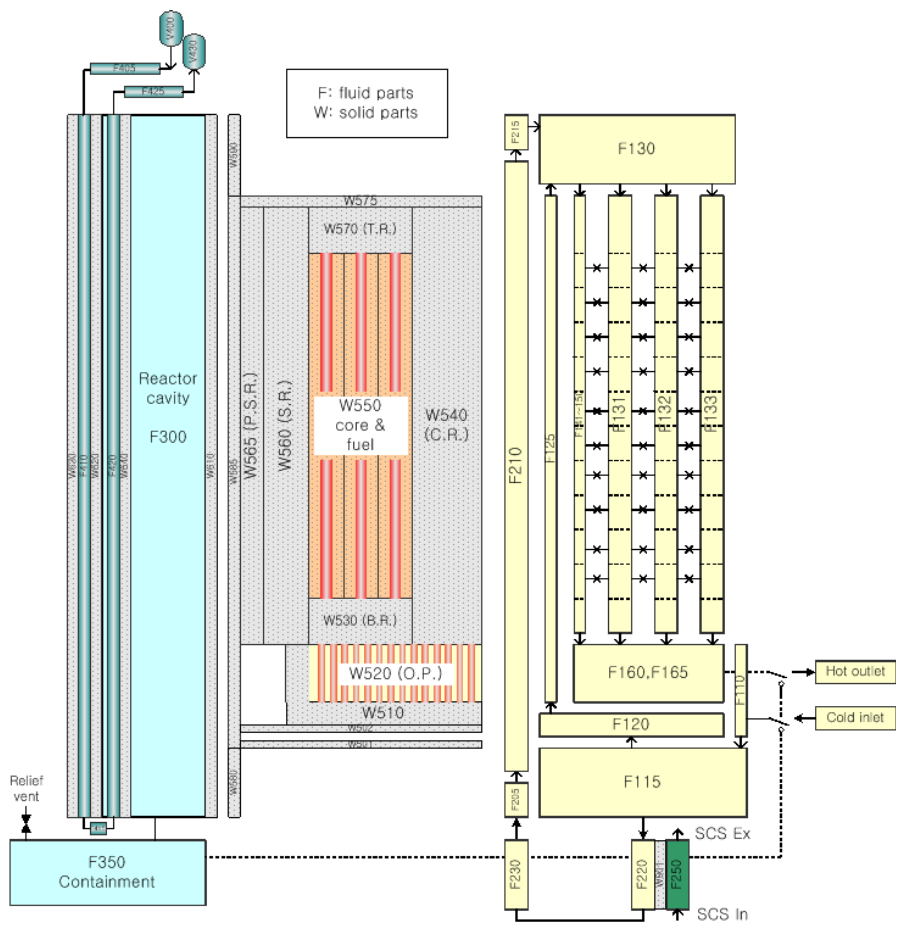

Fig. 3.2 System model for the GAMMA+ simulation.

The system model consists of the reactor coolant system, the reactor cavity and the RCCS, the SCS heat exchanger, and the containment. All solid regions are two- or three-dimensionally modeled. The fluid regions are modeled by the combination of two- and one-dimensional flow networks. In particular the reactor cavity and the annulus between the core barrel and the reactor pressure vessel (RPV) are modeled two-dimensionally in order to consider the local circulation flow characteristics. The thermal radiation heat transfers are considered in the top plenum, the annulus between the core barrel and the RPV, the reactor cavity containing the RCCS panels, and the annulus between the downcomer wall and the reactor cavity wall. The containment is modeled as single fluid volume. The air-cooled RCCS is modeled one-dimensionally, referencing the GT-MHR design. It is assumed that the ambient air outside the containment is at 1 bar and 43 ^{o}C.

3.4. Simulation Scenario

For the present air-ingress analysis, a guillotine-type rupture of the cross duct vessel connected with the steam generator is considered. This event allows both hot and cold legs of the core to be opened to the air, and it provides a pathway for natural circulation of air through the core.

Before the accident, the reactor is at full power. The break occurs at 0 s and helium coolant discharges into the reactor silo (1 atm) through the ruptured cold and hot legs with the areas of 2.16 and 1.61 m^2, respectively. Rapid depressurization of the core occurs and the reactor is tripped within 0.01 s with a time delay of 1 s by low primary pressure signal (6.24 MPa). The containment pressure is rapidly increased and the containment damper (or relief valve) is opened. The core decay heat is removed by conduction within the reactor vessel wall and then by radiation/convection to the air-cooled panels in the RCCS. The simulation is performed until 900 hours (37.5 days).

3.5. Air Supply Model into Containment

Air-ingress event is significantly affected by the air condition in the containment. Therefore, the following containment opening situations are investigated:

- Opening thru damper (or relief valve): The containment damper (or relief valve) is opened but failed to be closed after depressurization.
- Leak : The containment damper is opened and successfully closed after depressurization. Air leaks into the containment.

In addition, in order to simulate the air transportation into the containment, two kinds of methodologies are applied for the present analysis. The first methodology (Case A, Case B, Case C) assumes that the containment is continuously refreshed by air (See Fig. 3.3). Case A is to assume the complete refreshment of air in the containment, the worst case. Such a case may happen when no containment exists or significant amounts of the RCCS tubes are ruptured. Case

B and Case C are for the limited refreshment of air by the opening area of the relief valve or the leak path. In order to simulate the limited refreshment (Case B and Case C), the fixed flow rate of air is fed into the containment by force. The air feed rate is calculated by:

$$\dot{m} = \sqrt{2\,\rho_a\,\Delta P}\,A_f \tag{3.21}$$

where ρ_a is the density of the ambient air (43 ^oC, 1 atm), ΔP is the pressure difference (= 7 kPa), and A_f is the opening area (82, 6.6 cm^2 for Case B, Case C, respectively). The air feed rates calculated by Eq. (3.21) are 1.02 and 0.082 kg/s for Case B and Case C, respectively.

The containment refreshment methodology is adopted to provide conservative results whereas the containment vent methodology is for best-estimate ones. The containment vent methodology (Case D, Case E) assumes that ambient air is transported into the containment by natural phenomena (e.g., diffusion) through the opening area of the containment vent. Each case of Case D and Case E has a different opening area. Case D represents a damper type with a large opening area of 10 m^2 and Case E represents a relief valve type with a small opening area of 82 cm^2. Table 3.1 summarizes the five cases considered in the present work.

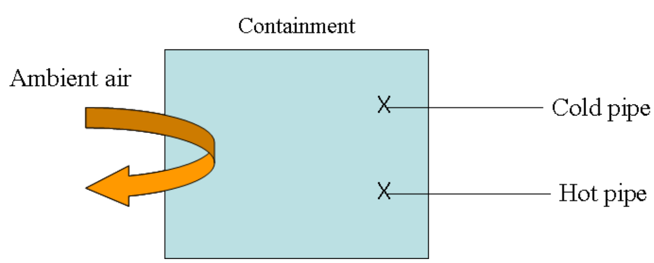

Fig. 3.3 Containment refreshment (Case A, Case B, Case C).

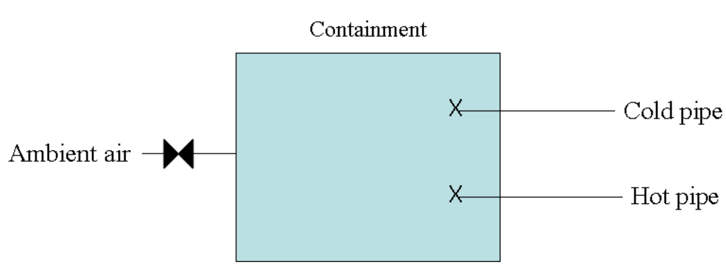

Fig. 3.4 Containment vent (Case D, Case E).

Table 3.1 The cases considered in the present work

Analysis Methodology	Cases		Description
Containment refreshed by air	Case A	Complete refreshment	The ruptured areas are exposed to the infinite volume of air (e.g., environment).
	Case B	Refreshment thru relief valve	Ambient air is fed into the containment with 1.02 kg/s.
	Case C	Refreshment by leakage	Ambient air is fed into the containment with 0.082 kg/s.
Containment vent through opening area	Case D	Damper vent	The containment damper is opened but failed to be closed. The opening area is 10 m^2.
	Case E	Relief valve vent	The containment relief valve is opened but failed to be closed. The opening area is 82 cm^2.

4. SYSTEM THERMO-FLUID ANALYSIS RESULTS

4.1 NATURAL CONVECTION

One of the most important parameters in the air-ingress event is the onset time of natural convection since the air-ingress rate into the core is very small before the natural convection. Fig. 4.1 shows the calculated mass flow rate from the containment into the reactor vessel at the hot pipe. The sudden increase of the mass flow rate in Fig. 4.1 indicates the onset of the natural convection occurs at ~550 hr (23 days) for the considered cases. Such a significant delay is mainly from the large fluid spaces in the lower head, the upper plenum, and the annulus for the VCS. It takes a long period of time to start the natural convection since the air has to fill up such large volumes by diffusion which is a slow process.

It should be noted that the large delay of the natural convection is assisted by the VCS flow path, which is the specific design adopted in this work. The VCS flow path connects the upper plenum with the annular space between the core barrel and the RPV. The connection enables a portion of the air in the upper plenum to diffuse into the annular space with a large volume. Therefore, it takes more time to fill up the upper plenum by air. A sensitivity calculation without this pathway showed that the natural convection occurs at ~360 hr. Earlier onset time of the natural convection may result in less graphite corrosion in the fuel blocks since more oxygen is consumed at the outlet plenum due to higher temperature. In other words, longer delay of the natural convection could lead more corrosion of the fuel blocks resulting in more production of radioactive source terms. Therefore, it is considered that the case of the significantly delayed natural convection is very important in terms of the release of the source terms although the operator has enough time to mitigate the accident.

The positive values of the mass flow rates at the hot pipe represent that the natural convection flow has an opposite direction of the nominal coolant flow. That is, the mixture gases flow upward at the active core due to the buoyancy. As shown in Fig. 4.1, the predicted mass flow rates are 0.15~0.2 kg/s for Case A ~ Case D. Case E has the smallest air supply of ~0.05 kg/s. In the case of Case E, about 13% of helium still exists in the core flow, but the contents of helium are negligible in the other cases. The smaller density of the mixture gas causes the smaller buoyancy head.

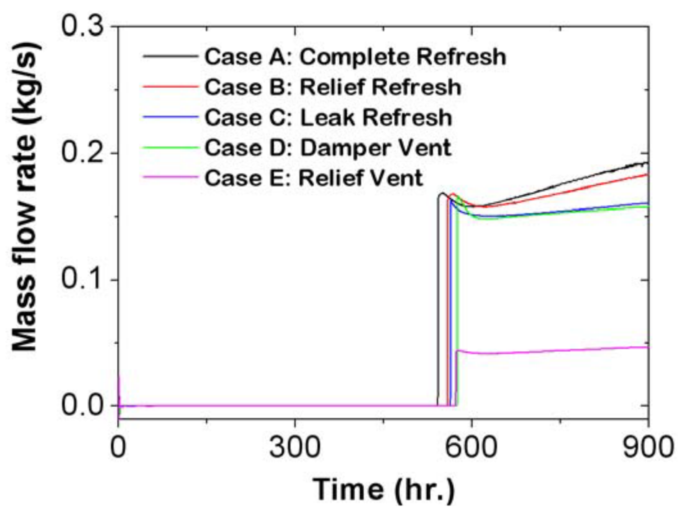

Fig. 4.1 Mass flow rate into the reactor vessel at the hot pipe.

4.2 Oxygen Concentration in Containment

Figs. 4.2 and 4.3 show the oxygen concentration behaviors in the containment for the considered cases. The figures show that each case has a unique behavior of the oxygen concentration in the containment. Obviously the oxygen mass fraction of Case A is kept as 23% due to an infinite air supply. For the other cases, massive air-ingress into the reactor after the natural convection leads a rapid change of the oxygen mass fraction in the containment. The consumed air in the reactor is determined by the rate of air supply into the containment. Compared with the containment refreshment methodology (Case A, Case B, Case C), the containment vent methodology (Case D and Case E) has lower oxygen mass fraction in the containment after the natural convection resulting in much lower oxygen supply for graphite corrosion.

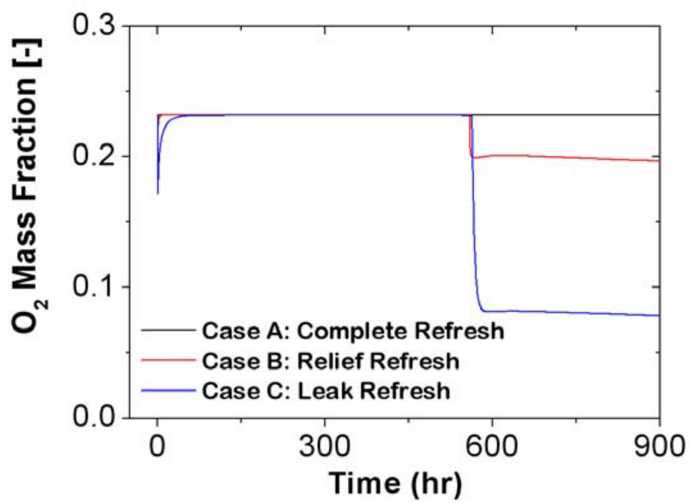

Fig. 4.2 Oxygen concentration in containment for containment refreshment methodology.

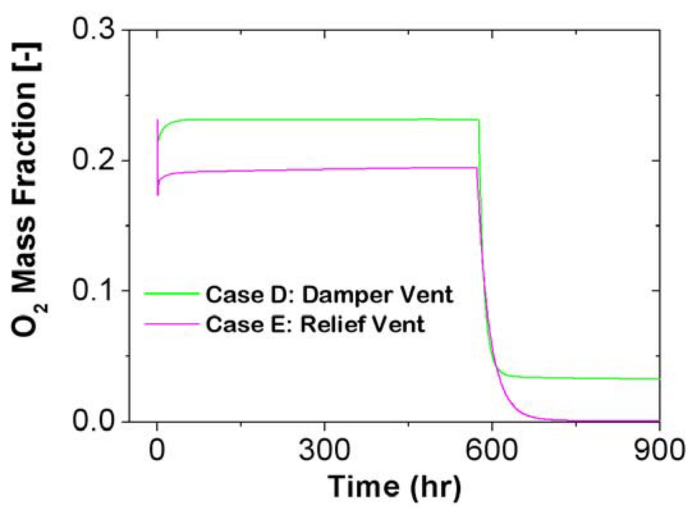

Fig. 4.2 Oxygen concentration in containment for containment vent methodology.

4.3 Temperature Behaviors

Figs. 4.3 and 4.4 show the temperature behviors during the air-ingress event. Before the natural convection, the temperature profiles for the five cases are nearly the same. The peak fuel temperature is lower than 1600 °C and is not affected by air-ingress. After the natural convection, massive chemical reaction occurs.

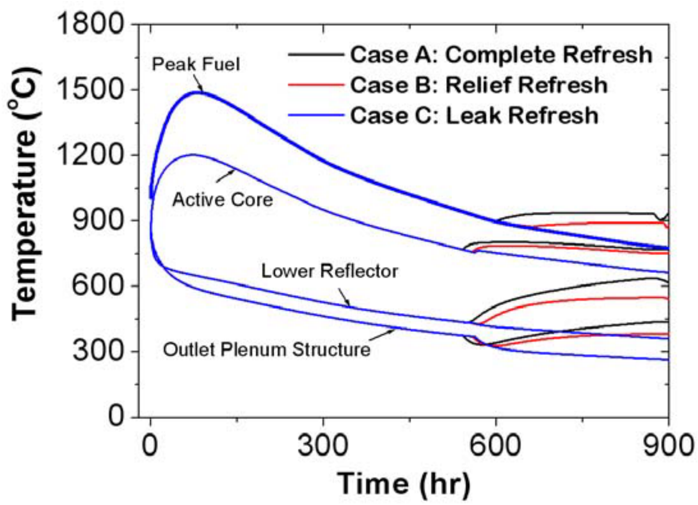

Fig. 4.4 Temperature behaviors for containment refreshment methodology.

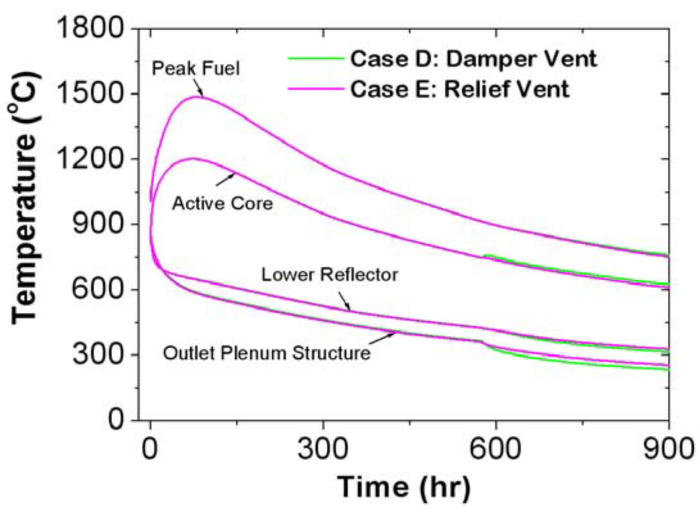

Fig. 4.5 Temperature behaviors for containment vent methodology.

The reaction heat causes the temperature rise at the outlet plenum structure, the lower reflector, the active core, and the fuel. Compared with the containment refreshment methodology (Fig. 4.4), the temperature increase induced by the chemical heat is smaller in the containment vent methodology (Fig. 4.5).

Fig. 4.6 shows the axial oxygen concentration at the core. It is shown that each case has different oxygen mass fraction at the outlet plenum. However, in all the cases, there is no oxygen at the exit of the active region. It clearly indicates that oxygen is depleted at the active core. Therefore, it can be expected that the graphite corrosion at the downstream of the active core (i.e., the upper reflector, the permanent side reflector) is negligibly small.

Fig. 4.7 shows the axial temperature profile at the core. The temperature peaks at the lower part of the active core are seen for Case A and Case B which have higher supply of oxygen. In the case of Case C, the chemical heat up is small due to smaller supply of oxygen, which leads lower temperature at the lower part. It can be expected that the other two cases (Case D and Case E) are similar with those of Case C. The chemical heatup of Case D and Case E are much smaller.

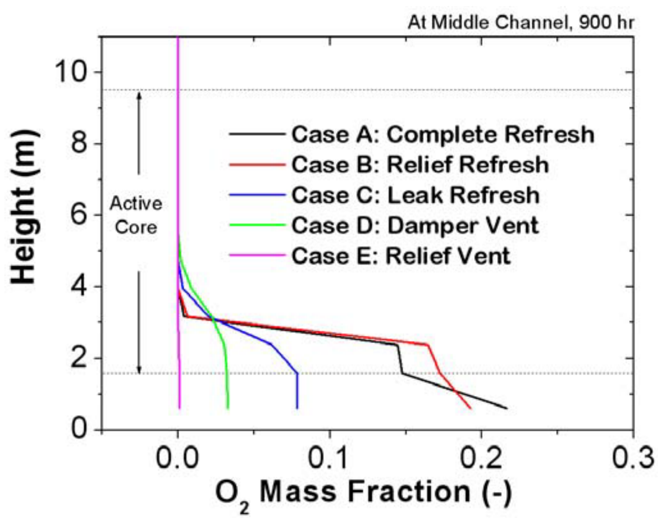

Fig. 4.6 Oxygen concentration profile at the core.

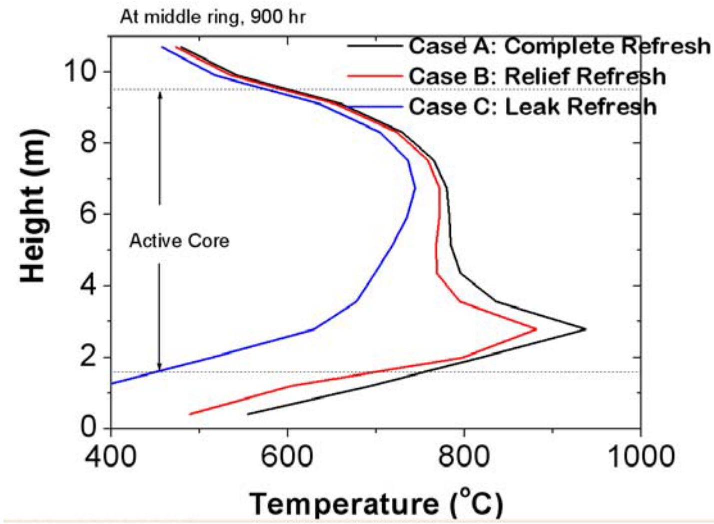

Fig. 4.7 Axial temperature profile at the core.

4.4 Graphite Corrosion

Fig. 4.8 shows the graphite corrosion versus time for Case A which represents the infinite air supply case. It is shown that massive graphite corrosion occurs at the active core and the lower reflector after the natural convection. In Case A, the corroded volume of graphite is increased with the time due to continuous air supply from the containment.

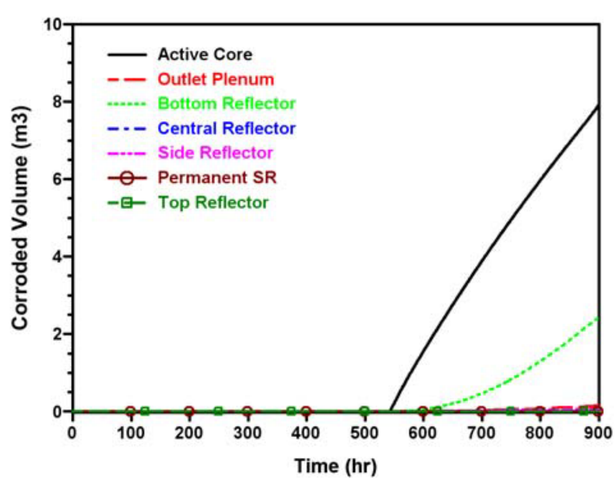

Fig. 4.8 Graphite corrosion behavior of Case A.

On the other hand, as shown in Fig. 4.9, the augment of the corroded volume of graphite is negligible after ~700 hr in Case E due to the lack of the air supply from the containment. Most of corrosions occur just after the natural convection starts in this case.

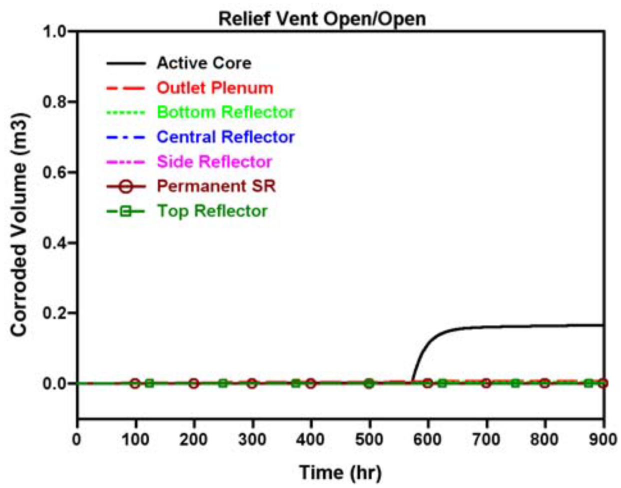

Fig. 4.9 Graphite corrosion behavior of Case E.

The corroded fractions of graphite for the considered cases are summarized at Table 4.1. The predicted corroded fraction of graphite is upto 3%. In particular, the corroded fractions of the active core and the bottom reflector are upto 11% and 19%, respectively. The calculated graphite corrosions by the containment vent methodology are significantly lower than those by the containment refreshment methodology. It should be noted here that the containment vent methodology is more realistic.

Fig. 4.10 shows the axial fractional burnoff profile at the core. It can be seen that the corrosion in the bottom layer of the fuel block is the most severe. Especially, complete burn off appears at the bottom layer of the fuel block for Case A and Case B. These occur at ~850 hr. However, one has to remind again that Case A and Case B are very conservative. In the case of Case D, the maximum burnoff is ~10% until 900 hr.

Table 4.1 The calculated graphite corrosion at 900 hr

	Initial graphite volume (m^3)	Corroded fraction (%)				
		Case A	Case B	Case C	Case D	Case E
Active core	72.3	1.10E+01	1.06E+01	4.90E+00	2.20E+00	2.30E-01
Outlet plenum	20.4	7.61E-01	2.79E-01	6.27E-02	5.59E-02	3.12E-02
Bottom reflector	12.9	1.88E+01	6.90E+00	2.43E-01	9.12E-02	3.16E-02
Central reflector	75.2	2.29E-02	2.17E-02	1.10E-02	5.66E-03	7.38E-04
Side reflector	110.4	6.53E-02	5.34E-02	1.66E-02	6.77E-03	6.79E-04
Permanent side refl.	43.3	2.68E-03	2.77E-03	2.83E-03	2.79E-03	1.19E-03
Top reflector	14.5	8.47E-05	8.72E-05	6.64E-05	6.63E-05	4.15E-05
Total	348.9	3.04E+00	2.49E+00	1.04E+00	4.66E-01	5.11E-02

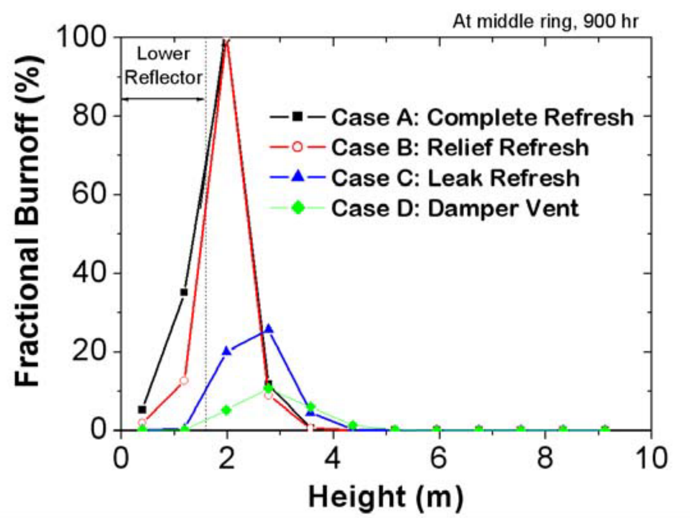

Fig. 4.10 Local burnoff of graphite at 900 hr.

5. SUMMARY

The purpose of the present analysis is to perform the system thermo-fluid and graphite oxidation calculations during an air-ingress event of the NGNP reactor. Two methodologies (i.e., containment refreshment and containment vent) were applied to model the air supply into the containment. The results of the system thermo-fluid calculations with the internal VCS pathway show that massive air-ingress is delayed until ~550 hr for all the considered cases. It is found that chemical reaction mainly attacks the lower part of the active core and the lower reflector due to the delayed air-ingress. The total corroded volume of graphite is predicted no more than 3% until 900 hr. On average, the predicted corroded fraction of the active core is upto 11% until 900 hr. The corrosion in the bottom layer of the fuel block is found to be the most severe.

6. REFERENCES

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