NGNP Conceptual Design Studies Reactor Building Design, Containment Issues, and Embedment Effects

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1.0 OBJECTIVE OF THE STUDY

The objective of this study is to develop technical and functional requirements for the NGNP and modular HTR containment and reactor building structure. The study also includes considerations of the NRC regulations and modular HTR design objectives regarding radionuclides release, regulations on physical security including design basis threats and external hazards protection, and whether there is a need for filtered reactor building and if needed, definition of filtration requirements. The study also evaluates the merits of locating the reactor building fully or partially underground.

This study is based on the current understanding of expected radionuclides release (source term) for Modular HTR accident scenarios specifically the AREVA prismatic HTR utilizing the indirect steam cycle. In addition the impact of a direct (conventional) steam cycle has been considered. As such, this is a scoping study helping to frame the issues associated with the development of technical and functional requirements (T&FR) for the reactor building in the containment of radionuclides. Accordingly, an objective of this study is to identify the issues and further R&D and engineering studies that are required to resolve these issues.

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

2.0 BACKGROUND

Traditionally the term containment and the reactor building were used synonymously. This is because in the current experience base of light water reactor technology the reactor building performs a key role in the radionuclides containment in the overall radionuclides retention function. Naturally, this has influenced the design and licensing of LWR containment/reactor building.

The NGNP design will use a helium cooled graphite moderated high temperature reactor technology specifically modular reactors with passive decay heat removal capability. This design performs radionuclide containment function with a variety of technical, operational and inherent design features that are 1) independent, 2) diverse, and 3) robust barriers between the source of radionuclides (inside each fuel particle) and the outside environment.

Therefore, it is noted that for modular HTR designs (including NGNP) reliance on the reactor building structure to perform radionuclide containment is not the same as that for the LWR designs. As such one of the design goals of the modular HTRs (to be demonstrated by NGNP) is that the traditional "source term" is expected to be low.

Traditionally, "source term" is defined as the amount and composition of radionuclides available for release into the reactor building and available for release into the environment if not for the capabilities of the LWR reactor building. In the past, for light water reactor licensing, the source term was defined generically and independent of reactor power level or type. More recently, the NRC has introduced the concept of "mechanistic" source term where the amount of radionuclides available for release is calculated specifically for the most severe accident scenario using mechanistic approach.

For modular HTRs and NGNP the use of mechanistic source term development methodology is recommended. This will support the design and implementation of the radionuclides containment and SSC functions.

One of the HTR design goals is to demonstrate that it is possible during a primary side depressurization accident to allow release of the inventory of radionuclides in the reactor building and the primary circulating activity into the atmosphere without violating the 10 CFR 100 site boundary dose limit. This design goal and the HTR radionuclides containment strategy drive the reactor operating characteristics and accident performance requirements.

3.0 TECHNICAL, SAFETY AND LICENSING ISSUES

In this section the technical, safety and licensing issues associated with the HTR radionuclides containment, reactor building are identified and discussed.

3.1 Technical Issues

The HTR radionuclides containment strategy as envisioned by AREVA is to maintain positive control and limit the release of radionuclides during all phases of reactor operations utilizing the following three physical barrier systems:

1. Ceramic coated fuel particles with high temperature capability dispersed in fuel compacts within the prismatic graphite fuel elements
2. A compact helium pressure boundary that completely surrounds the fuel barrier
3. A reactor building barrier that completely surrounds the helium pressure boundary barrier.

Each barrier functions independently from the others such that failure of one barrier does not result in failure of any other barrier.

Each barrier performs to achieve the plant radionuclide retention and containment function that meets the HTR design goal. Each barrier is expected to perform a separate and independent containment functions that will act to retain and limit the spread of radionuclides to the environment in normal and accident conditions.

The technical issues associated with each barrier are –

Ceramic coated particle fuel – The HTR fuel particle contains the fuel kernel coated with multiple layers of graphite and Silicone Carbide (SiC) coatings. Under power operating conditions the vast majority of radionuclides and fission products released by the fission reaction remain within the confines of the individual coated fuel particle. The radionuclides releases from failed particle fuel during normal and/or accident conditions fall in one of the following four categories:

Release from --

1. Manufacturing defects
2. Weak or weakened particles
3. Failed particles during normal operation
4. Failed particles during accident heatup conditions

The HTR fuel particles are designed and qualified to retain the majority of radionuclides within the particle coatings under all during normal and accident conditions. This is the fundamental technical and safety design basis of modular HTR fuel and reactor concept. The subsequent physical barriers and plant design and operating characteristics are chosen to function as defense-in-depth to maintain the radionuclide retention function.

Technical issues associated with the expected fuel particle performance are discussed in Reference (12-9077148-001, NGNP Fuel Design Special Study). In that the fuel particle and compact design envelope which provides adequate design flexibility for NGNP and future core designs were determined. The resulting design envelope considers the range of design parameters including enrichment, packing fraction, and use of burnable absorbers, which are required to support a design which allows operation of a nominal 18 month cycle and complies with fuel-related requirements of plant safety case. This information will be used to help define an ongoing NGNP fuel testing and qualification program through specification of a suggested testing envelope.

Primary helium pressure boundary – The primary helium pressure boundary contains the core. This includes the reactor pressure vessel, cross vessel, and the intermediate heat exchanger or the steam generator. Radionuclides that escape the coated particle fuel, fuel compact and the fuel block are contained in the primary system.

During normal operations, the source term consists of certain gaseous radionuclide species (Tritium – ^{3}H) that diffuse into the secondary side of the IHX or the SGs and others (e.g. Silver – ^{111}Ag) could plate out on certain surfaces of the primary circuit. Helium impurities such as graphite dust, nitrogen or oxygen could become activated and form the primary circuit circulating activity.

In an accident condition radionuclides available for release into the reactor building which constitutes the next barrier consist of the following:

- The initial circulation activity prior to the accident, and
- Accident source term consists of the radionuclides released from the fuel particles into the primary system as the result of the accident scenario, particle activation history, and post accident time – temperature dependencies.

The source term released into the next barrier (i.e. the reactor building) depends on the primary pressure boundary leakage or break status. The phenomena that are present for retention of radionuclides in the primary system pressure boundary are 1) the pressure boundary itself as a physical barrier, 2) plate-out of certain radionuclides on the inside surfaces of the pressure boundary, and 3) settlement of radionuclides on certain low turbulence areas of the primary system.

Reactor building – The reactor building structure surrounds the primary system pressure boundary. Any radionuclides released from the primary system are retained in the reactor building and is available for dispersion into the environment depending on the design choices and performance characteristics of the reactor building.

The reactor building design is the focus of this study and no specific design has been selected for this structure. In today's technical experience base with traditional Light Water Reactor (LWR) technology, the reactor building design is chosen to limit radionuclides release to the environment, i.e. retain radionuclides inside the building. This is considered the containment function of the reactor building. Additionally, the reactor building must function to protect systems, structures and components inside the building from external hazards and threats.

The technical bases for choosing an HTR reactor building design is not different from that in the current LWR technology. However, since the HTR and LWR technologies are fundamentally different, the reactor building designs should not be expected to be the same. HTR reactor building is defined as the structure that houses the reactor vessel (including the control rods), the IHXs, the circulators, the steam generators, the reactor cavity cooling system (panels and water supply tanks), and the interconnection piping.

The reactor building must maintain control of radionuclides and limit the spread of the radioactivity into the environment and protect the SSCs inside the building from external hazards and threats. These high level reactor building requirements are similar to those used in the LWR technology except for the degree of reliance on the radionuclide containment capability. Primarily due to fuel design differences the LWR technology relies heavily on the reactor building radionuclide containment function where HTR technology depends on the individual fuel particle coatings to perform this function.

Table 3-1 provides a listing of main technical issues considerations that must be addressed as part of the HTR reactor building design selection process.

Table 3-1: Reactor Building Technical Issues

	Technical Issues	**Key Analysis Factors**
1	Radionuclides retention capabilities	• Degrees of pressure retention - High to Low • Method of pressure retention • Need for filters • Plate-out and holdup considerations • Depressurization strategy
2	Degree of embedment	• Underground silo • Partially underground • Above ground
3	Ultimate heat sink	• Credit for beyond design basis accident UHS
4	External threat protection	• Airplane crash protection (including fuel fire) • Design bases threat protection design features
5	External hazard protection	• Seismic protection • External and internal fire and flood
6	Constructability	• Complexity of construction

3.2 Safety Issues

The HTR safety case relies heavily on the first barrier for the radionuclides containment, i.e. the coated fuel particle. In case the expected performance of the primary barrier is less than desirable and in the spirit of providing defense in depth for the critical safety functions two additional barriers are provided, i.e. the primary helium pressure boundary and the reactor building envelop.

Of particular interest in this study are the safety issues of the reactor building. However, the safety issues associated with all radionuclide barriers must be considered collectively when assessing the overall plant radionuclides retention function.

As the third barrier to radionuclides release, the reactor building must function to limit the release once the second barrier is compromised. The safety issues to be considered are as follows:

- Source Term evaluation assessment
    - The size of the Primary Helium Pressure Boundary (PHPB) opening.
        - A Leak
        - Small break
        - Medium break
        - Large break
    - Assessment of the reactor performance (accident initiation, progression and end state)
    - Assessment of fuel performance during each accident scenario
    - Reactor building radioactive “source term” versus time following the accident.
- Radionuclides containment functional response
    - Fuel performance requirements vs site boundary dose
- Reactor building barrier performance characteristics
    - Internal dose events

- Internal fire and floods
- Seismic performance assessment
- External hazards protection
  - Aircraft Crash Protection (ACP)
  - Design Basis Threat (DBT)

3.3 Licensing Issues

Licensing a new reactor concept such as the NGNP is highly unpredictable. The licensing rules and regulations that exist in the USA are primarily developed for the current light water reactor technology and any deviation from precedent requires solid justification and time consuming confirmatory reviews and analysis.

The overall licensing strategy for HTRs and NGNP is outside of the scope of this study and has been covered and reported previously within the pre-conceptual design phase of the AREVA team scope of work. Given the proposed strategy, the NGNP licensing activities inevitably will develop new regulations and criteria within the new Risk Informed and Performance Based (RI-PB) regulatory framework. Licensing issues associated with the containment of radionuclides and the role of the reactor building that might be considered is considered in this study.

The framework for advanced reactor licensing was developed by the US NRC and published in NUREG-1860. In this document the NRC proposed a framework for development of a risk informed and performance based criteria for licensing of power reactors as an alternative to the existing 10CFR 50 and 52. Under this framework early resolution to the NGNP reactor building licensing issues and the radionuclides containment methodology would be obtained. The licensing decision on acceptability of HTR radionuclides containment system and the role of the reactor building could play a critical parting determining the commercial viability of the modular HTRs.

Since the HTR technology is fundamentally different from LWR technology, the concept and the strategy for maintaining reactor safety is different from that used in LWRs. Therefore the LWR specific licensing rules do not apply. HTR safety and radionuclides containment strategy relies heavily on the particle fuel performance and the selection, configuration and operational parameters of the reactor.

The technical issues that are relevant to HTR licensing and have an impact on the reactor building and radionuclides containment discussions are as follows:

- No possibility of fuel melting
- Low circulating activity – normal operations,
- Low fuel failures – normal operation and core heat up accidents
- High negative reactivity coefficient – reactor shuts down without control rods
- High heat capacity and passive sensible and decay heat removal
- Low power density
- Slow transient evolution – large reaction time for most scenarios

Each technical issue must be broken down to a series of safety claims and translated into design decisions that must be fully proven and justified. The concept of margin to safety limits and defense-in-depth is a cornerstone of any licensing criteria. Therefore, margin to safety limits must be defined and developed, accepted, monitored and protected.

For radionuclides containment function of the reactor building the accident source term for Design Bases Accidents (DBA) and the Beyond Design Bases Accidents (BDBA) down to a plausible low mean frequency level shall not result in a plant site boundary dose of less than 1Rem. This is one of the key design goals of modular HTR technology that must be demonstrated by the NGNP.

The NRC has previously stated in policy and other public NRC documents that for advance reactors particularly non-LWR reactors, the need for conventional containment, i.e. high pressure, low leakage containment typical to LWR technologies, should not be a foregone conclusion. Instead, the regulations should be flexible for the designers to achieve, with high level of confidence, public health, safety and emergency preparedness by means that do not necessarily include a conventional LWR type containment. This vision is supported by the design work as the NRC crafts rules and regulations to assure unfettered advancements in reactor design, safety, and security.

HTR and the NGNP reactor safety are achieved in a fundamentally different way than LWR safety. Both current LWRs and the future HTR designs provide a very high level of protection to the public safety. The LWR radionuclide containment function is ultimately performed by a pressure retaining containment structure. The HTR radionuclides containment function is performed by multiple concentric and independent barriers between the nuclear fuel where the inventory of radionuclides and fission products reside and the general public that must be protected. Each barrier functions with active or passive safety features. The functions and performance requirements of each barrier would be selected based on the HTR modular technology specifics considering inherent properties of the core, the fuel, the moderator, and the Helium coolant. See Appendix A for a detail discussion of HTR radionuclides containment function.

Given acceptable fuel performance, the highest protection of the public is provides by early venting of Helium (primary heat transport media - coolant) in the event of major primary helium pressure boundary leak or rupture. Due to the basic nature of the HTR technology, the Helium released early in any event sequence contains known (i.e. measurable) normal circulating activity and some re-suspended dust from the system. This release has nominal, measurable, and acceptable consequences to the public. Once the reactor system and the reactor building are depressurized, the motive force for any subsequent fission product transport to the environment is removed. Thus, a vented reactor building, in general, benefits the HTR safety case. If the circulating activity in the primary system is such that it can not be readily vented into the atmosphere and still maintain site dose limits, a vented and filtered reactor building would be considered for the initial release.

Unlike LWRs the HTR designs including the NGNP accident scenarios are very slow, evolving over several days before peak fuel temperature is reached during scenarios without continued forced circulation cooling. During this period as core temperatures redistribute during a switchover to passive decay heat removal via conduction and radiation of decay and residual heat from the reactor vessel to the passive reactor building heat sinks, some delayed releases from a small number of initially defective fuel particles may occur, however such releases are minimized by minimizing the pressure gradient across the fission product barriers. Although public health and safety is assured by licensed passive means, the added time element and the vented and if necessary filtered reactor building design will provide substantial means for manual intervention for additional mitigation actions in the accident scenarios providing a truly defense-in-depth for accident management.

The HTR design and associated licensing strategy address the question of “containment” not as the name of a building or structure, but rather in terms of a “radionuclides containment system” that is designed, operated, and licensed within a risk-informed and performance-based with adequate defense-in-depth framework. The “radionuclides containment system” is comprised of multiple, independent, and concentric barriers to the transport of radioactive material. These barriers include highly reliable and robust TRISO fuel particles, nuclear grade graphite matrix, helium coolant and associated pressure boundary, and vented (albeit robust) reactor building structure. Fundamental to the defense-in-depth of the NGNP are the inherent safety features of the reactor technology as well as engineered passive and active safety systems that support the integrity of the barriers across a spectrum of licensing basis events. The inherent and passive engineered safety features of the reactor technology preclude the possibility of achieving fuel temperatures that would challenge the fuel particle integrity for a full range of licensing basis events including very low probability events beyond the design basis.

The safety functions of the HTR reactor building structure are different than those of LWR “containment building” as defined for conventional light-water reactors. Rather than serving as a physical barrier to the release of radioactive material from the fuel and reactor coolant system as it is with conventional LWRs, the NGNP reactor building structure provides important safety functions of maintaining core geometry and support structure for the passive heat removal capability, protecting against challenges to the reactor vessel and helium pressure boundary, and thereby controlling chemical attack of the fuel and graphite. The capability of the reactor building structure to provide a leak tight barrier for the initial phase of the depressurization event is neither necessary nor desirable in implementing the safety design philosophy of the NGNP and modular HTRs. The strategy to license this approach with the NRC is to work with them to fulfill their goal of developing a risk informed licensing process that is appropriate for the HTR technology and reactor design. To be successful with this approach it will be necessary to come to agreement on an appropriate set of top level regulatory criteria, licensing basis events, reactor specific safety functions, safety classification of SSCs required or supporting these safety functions, an appropriate set of regulatory requirements, and an approach to implementing the defense-in-depth philosophy that takes into account fundamental elements of the safety design philosophy of the NGNP and the modular HTR.

4.0 CROSS-CORRELATION OF DESIGN FEATURES VS ISSUES

This section provides an assessment of certain reactor building design feature on each technical, safety and licensing issues identified in the previous section. The goal of this exercise is to identify factors that have to be considered in order to "*minimize the extent of Exclusionary Area Boundary (EAB) and the need for a large Emergency Planning Zone (EPZ)".*

4.1 Reactor Building Design Features Considered

The NGNP reactor building is the structure that houses the reactor vessel, cross-vessel, steam generators (or IHXs), primary circulators, and the reactor cavity cooling system. Typically, this building is a vertical cylindrical structure built either partially or fully underground. See Figure 4-1 for a typical configuration.

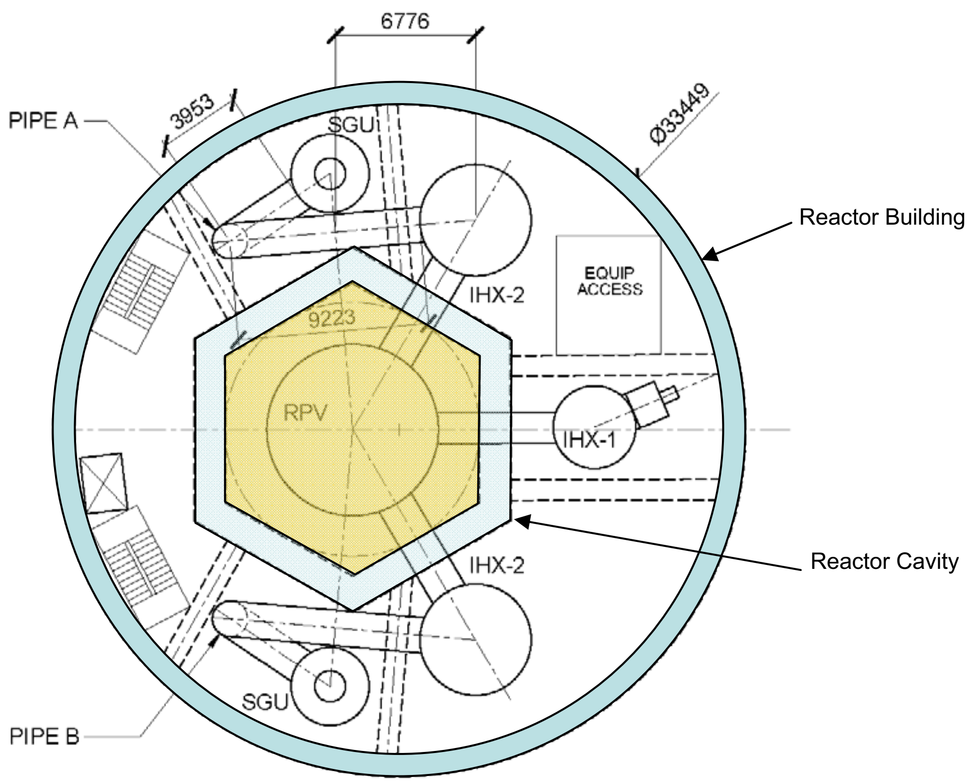


Figure 4-1: Plan view of a typical Reactor Building

The reactor cavity and building perimeter may be lined to minimize in-leakage or out-leakage. The temperature in the reactor cavity is controlled by the reactor cavity cooling system which is designed to passively maintain cavity walls temperature below maximum acceptable concrete temperature limit during normal and accident conditions. During normal operating conditions, access to reactor building internal spaces is determined by the maintenance needs and radiation level.

Other factors to be considered are discussed in the following sections.

4.1.1 Pressure Range

The pressure retention capability of the reactor building is one of the prime considerations associated with the HTR and the NGNP design. Typically it is desirable to maintain positive control of the reactor building pressure during normal operating conditions and maintain a slightly negative pressure in the reactor building using the normal reactor building heat and ventilations system. Additionally, air intake ports are provided to supply fresh air to the reactor building and the return air is monitored before release to the environment.

During accident conditions when there is a primary system depressurization the reactor building pressure will rise. In a sealed reactor building arrangement (pressure retaining), based on preliminary calculation the peak building pressure as the result of primary system break depends of the free volume of the reactor building and might be as high as 4 bars for a 25,000 cubic-meter building. In a vented arrangement the building pressure initially rises to activate the rupture discs or other vent mechanism. The time period to complete depressurization is dependant on the primary break vent size.

During a primary system depressurization accident the reactor building vent path is exposed to primary helium temperature before the helium is released to the filter system or the atmosphere. The reactor building surface temperatures rise could be close to core inlet temperature. The equilibrium building temperature depends on the mass and the initial temperature of the reactor building internal structures such calculations has not been performed.

The depressurization accidents are divided into six categories as follows:

1. Primary system leakage – gasket and fittings leakage. Course of action desired is to maintain control of the reactor building negative pressure using the reactor building HVAC system. Monitor and release building atmosphere through the exhaust stack. Identify the leakage and initiate repair procedure.

2. Small break – primary system break up to and including the size of the small bore attached piping (<15 millimeters) or instrument line failure to be verified during detail design. This scenario slow depressurization of the primary system. The helium released from the break will enter the reactor building and will be exhausted in to the outside environment through the stack.

Attempts will be made to identify and isolate the break. If the break is not isolatable within a certain period a reactor shutdown and normal depressurization and cooldown is initiated.

3. Medium break – up to and including the size of the largest attached piping (up to 80 millimeters) or stuck open safety relief valve. This scenario will cause a fast depressurization of the primary system and an automatic reactor trip on low reactor pressure or high reactor building temperature. Normal decay heat removal is commenced by secondary circuit, shutdown cooling system, or the reactor cavity cooling system. The reactor building pressure will tend to rise due to break, however, depending on the design of the reactor building, the primary helium is immediately release to the environment either through filters or unfiltered.

4. Large break – maximum depressurization of the primary system. This scenario will also cause an automatic reactor trip on low reactor pressure or high reactor building temperature. Normal decay heat removal is commenced through the secondary circuit, shutdown cooling system or the reactor cavity cooling system. The reactor building will respond to provide depressurization path for the primary helium to the environment through the plant stack. Such accident is considered beyond design basis.

5. Maximum energetic event – The maximum energetic event is the result of water ingress in a conventional steam cycle. During such event if the supply of water can not be terminated, the core water interaction will result in generation of combustible gases such as CO and Hydrogen. The increase in the primary system pressure opens the primary safety valves and leads to subsequent deflagration of the hydrogen and CO which results in a sudden increase in building pressure. Such accident is considered beyond design basis.

6. Steam Line Break – In a conventional steam cycle or an indirect steam cycle, a steam line break could also pressurize the reactor building with steam however without a concurrent or a subsequent primary side break no radionuclide release is expected. A concurrent steam and helium line break is considered beyond design basis.

4.1.2 Effect on Release Rate

The radionuclides release to the environment is expected to be low. The site boundary dose is limited to less than 1 rem per year during normal and accident conditions. If this can be accomplished then the EAB can be located at the site boundary and limit the EPZ.

Radionuclides release rate from the reactor building depend on the source term and the reactor building design. As discussed previously the reactor building source term is directly linked to 1) particle fuel performance during normal and accident conditions, 2) the fuel as manufactured quality, 3) plate-out and lift-off characteristics of the reactor pressure boundary, and 4) accident scenario. The reactor building source term release is linked to 1) the reactor building hold up and plate-out characteristics, and 2) the design features of the reactor building. Naturally if the source term is low (as expected) the release rate characteristics of the reactor building are not important. In fact, if the circulating activity forming the prompt source term is sufficiently low, it would be desirable to directly vent the building atmosphere without a need for passive filters and eliminate a potential driving force for the delayed source term which can be large. In a depressurization accident and following the initial release, the pressure in the building would be essentially atmospheric. Furthermore, it is much easier to isolate the reactor building with an internal pressure near atmospheric.

In an ideal case, the optimum HTR and the NGNP reactor building design are dictated by the particle fuel performance and manufacturing quality. If the fuel does not perform as expected and as a hedge for less than optimum expected fuel performance, the reactor building fission product retention capabilities can be tightened by adding certain design features such as filters. However, the choices are limited and constrained by increased cost and decrease in safety impacts which in turn influences the commercial viability of the reactor concept.

4.1.3 Air ingress (inert atmosphere)

Air ingress is the possibility of air infiltrating primary helium pressure boundary and causing oxidation of the core graphite structures and fuel blocks. This is a potential issue that has to be considered in the reactor building design evaluation.

In a depressurization accident scenario there is a potential for air ingress into the primary system. The source of air is that which resides in the reactor building during normal operation and in case of a vented reactor building is exhausted into the atmosphere as part of the initial depressurization of the primary system helium and is replace by outside air on cooldown due to pressure equalization effect. This leads to some amount of air that could infiltrate through the break into the primary system and come in contact with hot core graphite surfaces.

The chemical reaction that may ensue is the oxidation on hot surfaces of core graphite. Although the graphite oxidation process is exothermic, it is self limiting because of the defined amount of air available for oxidation. The number, the size and the location(s) of the primary break(s) can also play an important factor in how much air is available for ingress which dictates how long the oxidation can be sustained.

Air ingress can initiate graphite oxidation and if unmitigated a sustained graphite oxidation can weaken the core support graphite leading to possible compromised core geometry. If the graphite oxidation front reaches the fuel assembly graphite, the radionuclides trapped in the assembly structure graphite can be librated. This scenario is mitigated by adequate structural margins in the structural graphite design. In the unlikely case that the oxidation front reaches a large number of fuel particle, the oxidation process can not get past the SiC layer (high oxidation resistance) of the particle coating and expose the fuel. If the SiC layer of some fuel particles has already failed, additional incremental release would result.

The underground or partially embedded reactor building offers a limited supply of air for oxidation reaction. AREVA calculations have shown that there will not be enough supply of oxygen rich air in the containment space for a sustained a graphite oxidation leading to a failure of core support graphite. In an extreme case if a chimney effect is setup and a sustained supply of oxygen rich air is fed into the core, fuel block oxidation could occur in the flow passages of the lower fuel blocks and fuel particles could be exposed to the oxidation. In any case, the SiC layer of particle fuel coating will stop the oxidation front from reaching the fuel kernel.

To eliminate the possibility of graphite oxidation due to air ingress the reactor building atmosphere can be filled with an inert gas, however, consequences of such design on plant routine maintenance, personnel access, and life safety have to be considered.

4.1.4 Filtration

The concept of filtering the reactor building atmosphere before release into the environment would be necessary if in a post depressurization accident scenario, the radionuclides activity in the reactor building is high enough that release without filtering would violate the site boundary dose limit of 1 Rem per year.

Two types of filtering systems are possible, either a passive or an active system. In both cases the reactor building which is normally isolated from the outside environment is vented to the atmosphere following a rapid increase in building pressure, i.e. a large break. Depending on the volume of the reactor building and the primary system pressure (typically 5 to 7 MPa), the maximum pressure the reactor building could reach before venting to the atmosphere could be as high as 3 or 4 bars.

The reactor building surface temperature in the path of depressurization chutes could rise to core inlet temperature immediately following a depressurization accident, however little heat is transferred because of the short duration (seconds for large breaks) of the transient. The passive filter media, however, must be able to withstand the elevated temperature shock. The building temperature will begin to drop due to helium expansion as it cools and transfer energy to the building structures by convection. The building pressure will follow the same general trend, it rises to a maximum building pressure of possibly 3 to 4 bars and drops to the atmospheric pressure as it completes its vent cycle.

The reactor building depressurization and vent system reduces the building pressure by either directly venting into the atmosphere or through a passive sand filter system. Following the initial depressurization, the reactor building atmosphere is actively filtered and vented through the plant vent stack.

Following a break in the primary system an extended pressure holdup period in the reactor building is not desirable because the source term will continue to increase due to increasing core temperature as the result of the original accident. However, if the reactor building is depressurized, the potential driving force behind the increasing accident source term is removed and any further release can be monitored, filtered and possibly released utilizing nuclear grade HEPA filter system.

The decision to allow release of the initial circulating activity depends on the quality and performance of the TRISO particle fuel during normal reactor operation. If such quality can not be achieved, proven and licensed, the resultant consequence is the need for pressure retaining reactor building and radionuclides filtration equipment at high building pressure. This is because the reactor building pressure can not be reduced by condensation process, as is the case in LOCAs in LWRs. Thus building pressure must be relieved to the atmosphere, filtered or unfiltered, at some point after the initial depressurization accident.

The source term primarily includes:

- Metallic fission products such as Sr-90, Cs-137, and Ag-110
- Tritium
- Short lived fission product such as I and Xe, and
- Noble gases such as Xe and Kr-85
- Actinide such as Pu-241 and Am-241 (in case of TRISO coating failure)

Some of the activity could be mobile and circulating in the primary system with graphite dust. Others could be plated-out on the primary circuit surfaces or settled and lifted-off and mobilized due to the depressurization forces. The mostly graphite dust has been more of a phenomenon associated with the pebble bed reactor design because of constant fuel movement. No dust problems have been reported in the past HTRs with prismatic fuel designs.

4.2 DBT Other Hazards

The general, security requirements for HTRs and NGNP are expected to be similar to current requirements for current U. S. commercial nuclear plants. While some minimum characteristics are common to all facilities, the specifics are largely dependent on the location and design details of the facilities. A lower level of protection may be required for the NGNP if the actual (or perceived) risks are shown to be lower than those existing at today's commercial facilities. Unique design features of the NGNP, such as locating facilities below grade, will affect almost all aspects of the security program.

HTR security staffing and training requirements are expected to be similar to current U.S. commercial nuclear plants. However, integration of plant safety and security is encouraged by the NRC during the initial design. Novel security measures must be integrated into the NGNP design lowering the security risk profile of the plant.

A significant fraction of the lifetime costs of a security system is the cost of the guard force. Over the operating life of the project, total costs would be lower for a design that relies on hardware systems (i.e. more CCTVs, intrusion detection system, etc.). However, if the objective is to reduce capital costs, security could rely more heavily on a guard force. Conversely, a greater investment in hardware would reduce lifetime labor costs.

Without a security plan and complete security system design, only a coarse estimate of minimum security staff can be made. A reduced level of protection may be justified if the actual (or perceived) risks are less than those existing at today's commercial nuclear facilities. Note that post-9/11, commercial nuclear power plants were required to have amongst the most comprehensive security programs despite the fact that nuclear plants can be considered hardened targets with limits on the damage an adversarial force can cause. Social and political forces may require security requirements in excess of those strictly warranted by risk. As noted elsewhere in this report, security requirements can only be expected to escalate in the future.

While NGNP may include a number of unique features, (such as subterranean reactor containment and little or no offsite radiological consequences) their benefits on physical security have not been evaluated in depth. Given the limited consequences of an attack on any nuclear power reactor, a strong argument can be made that the reactor is of little strategic value from an actual (not perceived) risk viewpoint. Co-located hydrogen production, storage and transport would increase the probability of an attack as well the potential consequences.

Local terrain and features, (bodies of water, local terrain, nearby industrial or military facilities, size of site, etc.) will also affect the guard force size. One example of this is the use of elevated guard towers. Since 9/11, most commercial nuclear facilities in the US have been required to install elevated towers as observation platforms and strategic response. The number and location of these towers is governed to a large extent by their ability to give the guard a clear line of sight – or line of fire. Objects that block the line of sight (such as buildings, tanks, trees, blast shields, etc.) need to be considered in the placement of the towers. With minimal above-ground structures and flat terrain, four towers should be considered a practical minimum.

Minimum Security Staffing Requirements

The Table 4-1 provides an estimate of minimum staffing requirements for the reactor portion of the NGNP. Co-located hydrogen production and storage facilities may increase these minimum staffing by a factor of two. For the purposes of this estimate, we assume that hardware systems would be used to the maximum extent practical, reducing the size of the security force with no unique aspects that require extra guards. A detailed security analysis, including the development of target sets, adversary force size and attack and defense scenarios; considering the inherent safety characteristics of the HTR design which may significantly improve plant security, would be required before security system requirements could be finalized.

Table 4-1: Estimated Minimum Security Staffing for NGNP

Position/Function	On-Shift	Total for Five Shifts[3]
Manager[1]	0	1
Clerical Support[1]	0	1
Training Officer[1]	0	1
Shift Supervisor[5]	1	5
Central Alarm Station Officer	1	5
Secondary Alarm Station Officer[7]	1	5
Access Point[6]	2	10
Guard Tower[4,7]	4	20
Roving Patrol[2]	2	10
TOTAL	11	58

Notes:

1. Support staff is not required to be on-shift and are assumed to work traditional 5-day work week. Could be shared amongst multiple units.
2. Roving patrols can be on foot or in a vehicle.
3. Four shifts are required to provide 24/7 coverage without mandatory overtime. A fifth shift provides for training, re-qualification, vacations and sick time. As a power production facility, full time security coverage would be required 24/7.
4. Assumes a tower on each of the four corners of the site. The number, physical location, and height of guard towers must be adequate to ensure that the guards have an unobstructed view of the field of fire. Topographic features such as roads, rivers, drainage ditches, hills, etc. must be considered in the location and number of the towers.

5. Shift supervisor can be used as CAS officer, or to relieve security posts as required.
6. Assumes one officer at portal monitors, turnstiles and one in office. The on-duty officer in access area office may be able to perform SAS duties.
7. May not be required.

In general, the size and layout of the NGNP facility will have a greater effect on the security requirements than the technology employed. However, because of the inherent safety of this technology if the NGNP does not present unmitigated risks unique to HTR technology, lower security measures would be expected, with lower attendant costs. In today's environment, however, security requirements of any nuclear installation are expected to escalate in the near term. Detailed assessments of design basis threat and security requirements are discussed in Appendix B.

4.3 Embedment Effects

The degree of embedment is evaluated in this section against a series of attributes associated with the reactor design, operations, safety and security, construction, cost and seismic. There are benefits and consequences associated with underground or partially underground construction that must be considered primarily for the commercial HTR application that must be demonstrated in the NGNP.

This section of the report deals with the effect of the Reactor Building embedment on the following:

- Operations (reactor protection, refueling etc.)
- Reactor design
- Ultimate heat sink
- Site location (geo-technical constraints, water table)
- Natural phenomena hazards
- Physical security including Design Basis Threat
- Construction complexity
- Vulnerability to external hazards from other nearby industrial facilities
- Cost

4.3.1 Operations (Reactor Protection, Refueling, Etc.)

The reactor operation in either normal or accident conditions has limited impact on the degrees of embedment. However, for refueling operations, the full embedded core is preferred. This is because the refueling equipment would best operate at ground level. This is to minimize the refueling time for the fuel movement to and from the reactor.

The water reservoir source for the reactor cavity cooling system (RCCS) must be located above the reactor cavity cooling panels. The RCCS water storage tank would then be located at ground level simplifying seismic support and post accident refill.

Maintenance activities that require heavy cranes prefer embedded reactor building design. Such maintenance activities would include primary circulator replacement or repair, IHX replacements, and the RCCS panel inspections and maintenance.

Operational activities such as refueling would also prefer embedded reactor building design. The refueling machine is expected to be a large heavily shielded mechanism that must operate above the reactor vessel and be shared among the modules in a multi module plant configuration. This does not affect the design of the refueling equipment directly, but it does affect the structural and seismic requirements on an above grade building which support the refueling equipment.

4.3.2 Reactor Design

The prismatic reactor design is not influenced by the reactor embedment or installation depth. However, the refueling system and location and elevation of the near reactor fuel storage space are impacted by the embedment depth. These issues were addressed in the previous section hence they will not be repeated here.

4.3.3 Ultimate Heat Sink

The NGNP ultimate heat sink is provided by the reactor cavity cooling system (RCCS). The RCCS performs two functions:

- Protect the reactor cavity concrete temperatures (and also thereby reactor vessel and fuel temperatures) from exceeding the code permissible limits under normal, off-normal and accident conditions; and
- Provide an alternate means of core decay heat removal

Thus, the RCCS is required to operate continuously under all modes of plant operation.

Decay heat removal under accident conditions (while protecting reactor cavity concrete)

Under accident conditions, the inventory of water in the passive cooling system closed loop constitutes the ultimate heat sink. The inventory of the water in this system is relied upon to remove decay heat through sensible heat and subsequent boil-off. The passive system is maintained at a predetermined pressure level so that decay heat can be removed by sensible heat transfer for low end of the severe accident spectrum. The duration and severity of the accident would be analyzed to determine:

- Inventory of water needed
- System pressure setting
- Concrete temperature transients following various accidents
- Reactor vessel and fuel temperature transients following various accidents

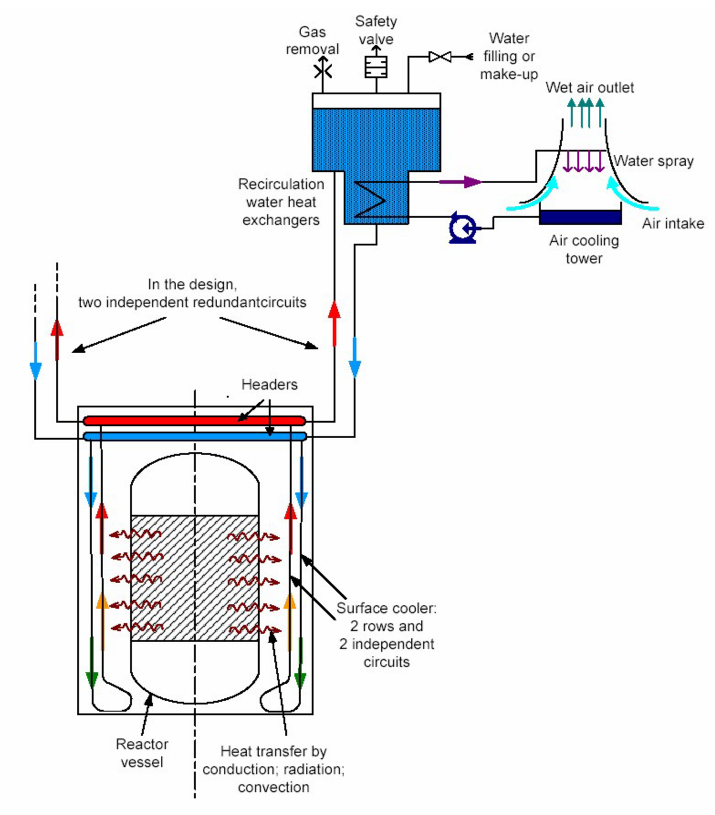


Figure 4-2: Reactor Cavity Cooling System

Interrelationship of the Ultimate Heat Sink design with Technical Issues

Degree of Embedment

The embedment depth of the Reactor Building (fully or partially embedded) does not influence the ultimate heat sink. The vapors formed in the RCCS tanks as part of the boil-off must be vented to the plant stack through separate penetrations in the reactor building since the reactor building HVAC system penetrations could be isolated under the accident conditions. Refilling of these tanks during an accident are less complicated if the tanks are at the ground level (fully embedded option).

External Threat Protection

The passive mode of the RCCS uses components located in the reactor cavity and the Reactor Building only. Thus, the RCCS passive operation mode is well protected. Hence, external hazards such as an airplane crash, a blast or explosion (as part of the design basis threat) is expected to have minimal impact as long as the vent lines open to vent-off the vapors from the boiling pool to the outside.

Beyond Design Basis Accidents

The HTR must contend with Beyond Design Basis Accidents:

- Airplane crash
- Core disruption on loss of core cooling function
- Depressurized conduction cooldown with or without a reactor scram

Maintaining core geometry is one of the top level required safety functions partially provided by the Reactor Building and the vessel support system. Therefore, a very large seismic event that results in the core geometry disruption leading to fuel blocks geometry disruption and relocation in the bottom of the reactor pit is excluded from consideration. Such accidents are well below the minimum accident cut-off frequency and thus unmitigated.

Furthermore, in contrast to LWR cores, HTR cores have no heatup accident scenarios that could cause fuel melt leading to separation of fuel from the fuel coating and recollection of melted core to the bottom of the reactor or the reactor cavity pit.

4.3.4 Site location (Geo-Technical Constraints, Water Table)

The effect of the site location on the Reactor Building embedment as well as other parameters is discussed qualitatively in Table 4-2. A detailed discussion follows.

Site Location

The site location should be based on geological, hydrological and seismological considerations. The foundation mat must be located on stable soil strata, preferably on or near bedrock. Expansive clays are subject to large volumetric changes under wet/dry conditions and should be avoided; saturated loose sands/silts must be avoided to preclude the possibility of soil liquefaction during a seismic event. Although rock foundations are highly desirable from a structural standpoint, deep excavations in bedrock may increase construction complexity and cost. The presence of a high water table causes hydraulic pressures on an embedded structure and necessitates additional design considerations; it also adds to construction costs, as dewatering may be required as discussed in section 4.3.7. Since soil conditions generally improve with increasing depth of excavation, an embedded structure is likely to encounter better foundation strata than an above grade structure.

Seismic Performance

Recent developments in soil-structure interaction have generally confirmed that the deeper a structure is embedded into the ground the smaller the seismically induced accelerations and stresses it has to endure. Soil-structure interaction occurs in two ways: as kinematic interaction, which modifies the seismic wave front due to incoherency effects caused by waves propagating through and reflecting off the embedded structural medium, and as inertial interaction, which combines stiffness and inertia (mass) properties of soil and structure into one coupled dynamic system.

In general, foundations constructed on soft soils have a tendency to amplify the seismic motion, whereas those on bedrock do not. If seismic effects are to be minimized, it is advantageous to embed the structure sufficiently and place the foundation on firm soil or bedrock.

Seismic accelerations are generally lower for embedded structures than for above grade structures. In-structure-response-spectra (ISRS) that are used for the design of systems and components are also expected to be lower for an embedded structure. The relative reduction in ISRS depends upon the vibration characteristics of the foundation system, the structure, systems and the components. Generally, it is expected to be significant, possibly 50% or more.

Table 4-2: Effect of Various Parameters on the Embedded and Above-grade Reactor Building[1]

SUBJECT	EMBEDDED STRUCTURE	ABOVE GRADE STRUCTURE
Site Location		
High water table	Heavier structure due to hydrostatic pressure. Added costs for dewatering and maintaining water tightness of the structure. Higher construction cost.	Reduced impact
Geology	Bedrock formations result in higher excavation costs.	Excavation needed to reach bedrock or stable soil strata.
Seismic activity	Bedrock formation minimizes seismic amplification and will result in lower seismic response. Soft soil amplifies seismic response. Embedded structures will result in lower seismic response compared to the above grade structures for the same soil conditions.	
Natural Phenomena		
High winds, tornado, tornado missiles	No impact on the embedded structure.	Structure must be designed for these loads and impacts.
External Hazards		
Aircraft impact Chemical storage, processing, or transport near the site	Advantageous. Added costs due to the required heavier slab should the embedment be too deep. Aviation fuel fires need to be evaluated. Chemical release/explosions must be considered.	Structures must be designed for aircraft impact and the facility for the resulting fires. Chemical release/explosions must be considered.
Physical Security including DBT	In general advantageous compared to above grade structure. However, it is detail dependent.	
Construction Complexity	Construction is more complex than for non-embedded structure	
Cost	Costs are determined by sum total of costs due to each of the factors considered in this table.	

[1] Note: Other safety class structures (other than the Reactor Building) have not been evaluated.

4.3.5 Natural Phenomena Hazards

Natural phenomena hazards comprise floods, high winds, tornados, tornado missiles, hurricanes and earthquakes. With possible exception of the external floods, the effects of other natural hazards are considerably reduced for an embedded structure, whereas above grade structures must be designed to withstand these events.

4.3.6 Physical Security Including Design Basis Threat

The embedded Reactor Building structure reduces the physical security requirements. However, the detail of the layout would have to be known to make such a determination.

Additionally other safety grade buildings must also be evaluated from the consideration of impact based on the degree of embedment.

Security requirements for a NGNP are expected to be similar to current requirements for current U. S. commercial nuclear plants. While some minimum characteristics are common to all facilities, the specifics are largely dependent on the location and design details of the facilities. In general, the size and layout of the NGNP facility will have a greater effect on the security requirements than the technology employed. Unique design features of the NGNP, such as locating facilities below grade, or nearby chemical plants will affect almost all aspects of the security program. A reduced level of protection may be justified for the NGNP if the actual (or perceived) risks are less than those existing at today's commercial facilities.

Many aspects of the physical security program can be affected by design or operational changes. These aspects include: stand-off distance; number and locations of defensive positions; use of delay tactics; number and frequency of security patrols; number and location of close-circuit television camera; location of central alarm station; location and design of backup central alarm station; alarm response procedures; guard qualifications; guard training; state and federal weapons authorization; fire protection; response time of local law enforcement, access to airports and special equipment;

emergency plans, resources of state and local governments, resources of local industrial facilities; availability of water resources for fire fighting; and vehicle barrier system capabilities.

Physical security programs and strategic security plans at commercial nuclear power plants are highly integrated. Changes in one aspect of the plan can indirectly affect other elements – intentionally or otherwise. Because the principles of security are similar for most industrial faculties, we can expect similar degree of integration at NGNP. While the nuclear technology ultimately employed at the NGNP can be expected to influence plant security, other aspects will be at least as important.

If the NGNP does not present any unmitigated risks, costs will be comparable. Passive security features (such as a body of water, trees), incorporated at the conceptual design could reduce costs. Security requirements are expected to escalate for the foreseeable future.

4.3.7 Construction Complexity

The hexagonal reactor cavity and the containment and the interior walls are good candidates for sandwich wall modular construction using top loading. A typical sandwich wall construction is comprised of a steel sandwich formwork with built-in stiffeners. Alternately, reinforcement cages could be used with or without stiffeners.

The complexity associated with the construction of the partially or fully embedded silo is discussed in this section.

The silo measures approximately 35 meters in diameter and the basemat is located about 50 meters below grade as shown in Figure 4-3.

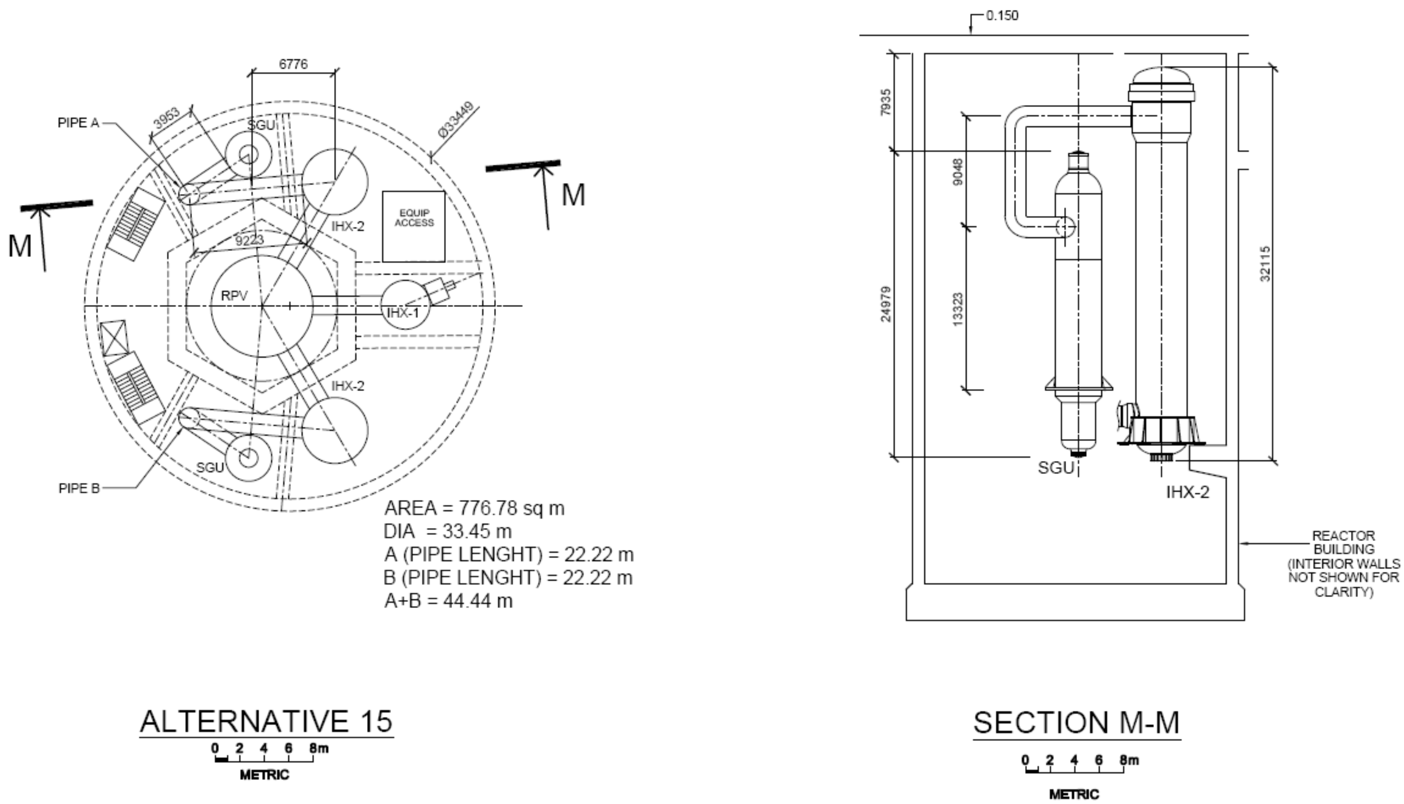


Figure 4-3: The Selected Reactor Silo Layout for this Study

The inside surface of the silo will be lined with a steel liner. Both the liner and the structure will be designed to ASME Section III, Division 2 requirements.

Fully Embedded Silo Construction

INL Site

Excavation

At the INL site the silo would be excavated using blasting in full benches. Monitoring of blasting would be required and the necessary control programs will need to be put in place. Thus, it would not be cost effective to place concrete for the permanent structures such as the Reactor Building prior to completion of the blasting.

Basemat Construction

Immediately following the excavation construction will commence on the silo basemat. The basemat is a reinforced concrete structure containing bridging bar assemblies to transfer interior wall reinforcing through the containment liner plate into the mat.

Reactor Building Liner

The installation of the liner will begin upon completion of the silo basemat. The floor and exterior wall liners will be installed in parallel. Prefabricated section sandwich rings will be used to minimize work effort in the deep silos as much as possible. All modular sandwich rings will be large enough to require hauling by a heavy carrier in order to minimize labor.

Reactor Cavity Construction and Interior walls

The construction of the hexagonal reactor cavity and the interior walls will proceed in parallel with the Reactor Building using prefabricated sandwich modular construction.

<u>Other Sites</u>

At other sites the excavation technique will depend upon the site soil and geological conditions as described below:

Savannah River Site (SRS)

SRS is a soil site with a groundwater table about 13 m below grade. One of the methods for groundwater control is by placement of concrete cutoff walls by the slurry trench method. Other method employs several sump pump pit dewatering systems. These methods are discussed in section 4.3.8

Dewatering would be maintained until the containment silo is loaded with its heavy components to avoid the water floating/lifting the containment structure upward. Backfilling operations will follow at that stage.

While the subsequent subsurface excavation time will be shorter than for the INL site, the total construction period would probably be longer compared to the INL site.

<u>Above Grade Construction</u>

The above grade design must be protect the Reactor Building from external threats and hazards by designing a nuclear grade enclosure around it. Additionally, the structure is to be designed to:

- Higher seismic loads
- Natural phenomena: high winds, and considerations of tornado and tornado missiles

4.3.8 Suitability of Above and Below Grade Construction at various sites

Three sites were considered for a little more specific determination of the influence of site parameters on the embedded and above grade construction concepts:

- Representative coastal site: South Texas Project (STP) site
- Representative site with high water table: Savannah River Site
- Rocky site: NGNP site at INL

South Texas Project (STP) Site

The Mitsui – South Texas Project (STP) site, is approximately 49 km^2 (12200 acre) site located in a rural area of south central Matagorda County. Matagorda County lies in the Coastal Prairie region in the southeastern part of Texas, along the Gulf of Mexico. The STP site is located approximately 143 km (89 mi) southwest of Houston, Texas, and 322 km (200 mi) southeast of Austin, Texas.

The STP site lies within the Coastal Prairie subsection of the Gulf Coastal Plain Physiographic Province. It begins at the edge of the Gulf of Mexico and extends to the northwest for approximately 80 to 120 km (50 to 75 mi). The land surface has an almost negligible slope to the southeast. The sediments are composed of young (Pleistocene and Holocene) unconsolidated deltaic sands, silts, and clays incised by meandering streams that discharge into the Gulf of Mexico. Approximately 8 km (26000 ft) of unconsolidated Cenozoic sediments underlie the surface of this area. The elevation ranges from sea level to approximately 90 m (300 ft), and is approximately 9 m (30 ft) MSL at the STP site.

Savannah River Site

The Savannah River site (SRS), is a circular tract of land occupying approximately 803 km^2 (310 mi^2) within Aiken, Barnwell, and Allendale Counties in southwestern South Carolina. The center of SRS is approximately 40 km (25 mi) southeast of the city limits of Augusta, Georgia; 161 km (100 mi) from the Atlantic Ocean Coast; and about 177 km (110 mi) south-southeast of the North Carolina border. SRS is bounded along 27 km (17 mi) of its southwest border by the Savannah River. The elevation of SRS varies from 25 to 122 meters (80 to 400 feet) above mean sea level (MSL).

The topography at SRS varies from gently sloping to moderately steep. Some upland areas are nearly level, and those on bottomland along the major streams are level. The slopes in small, narrow areas adjacent to drainage-ways are steep. Most of the soils are sandy over a loamy or clayey subsoil. The well-drained soils have a sandy surface layer underlain by a loamy subsoil. The somewhat excessively drained soils have a sandy surface layer that extends to a depth of 2 meters (80 inches) or more in some areas. The soils on bottomland range from well drained to very poorly drained. In the Sand Hills area, some soils on the abrupt slope breaks have dense, brittle subsoil. Commonly referred to as "Carolina bays", numerous upland depressions are found on SRS. These range in size from less than one acre to many acres. Water stands in most of these depressions for long periods in most years.

INL

The Idaho National Laboratory (INL), is approximately 2300 km^2 (890 mi^2) of the upper Snake River Plain in southeastern Idaho. It is located 328 km (203 mi) from Salt Lake City, Utah; 380 km (236 mi) from Butte, Montana; and 450 km (280 mi) from Boise, Idaho. INL is located mostly within Butte County, but with portions in Bingham, Bonneville, Jefferson, and Clark Counties.

The INL is located in a large, relatively undisturbed expanse of sagebrush steppe habitat. Approximately 94 percent of the land on the INL is open and undeveloped. The site has an average elevation of 1500 m (4900 ft) above mean sea level (MSL), and it is bordered on the north and west by mountain ranges and on the south by volcanic buttes and open plain. Lands immediately adjacent to the INL are open rangeland, foothills, or agricultural fields.

The Eastern Snake River Plain, where the INL is located, can be summarized as a broad northeast-trending basin that began filling with volcanic deposits about 6 million years ago. Most of the Plain that is visible today was shaped by volcanic eruptions of lava flows and domes during the last 1.2 million years. Overlying the lavas are thin, discontinuous deposits of wind-blown sand and loess, floodplain, riverbed and lake sediments, and landslope debris. These sedimentary deposits are often found between the lava flows, showing that a quiet period occurred between past volcanic eruptions. To the northeast, the Plain merges with the Yellowstone Plateau. Higher elevation mountains and valleys of the Basin and Range Province bound the Plain to the north and south. These mountains are formed by rocks more than 70 million years old, which have been folded and faulted. This Basin and Range deformation, which began 20 to 30 million years ago, affects some ongoing volcanic and tectonic processes in the INL area.

Surficial sediments at the site consist mostly of gravel, gravelly sands, and sands, and vegetative cover is only about 5 percent. Soils have been characterized and consist of 1.5 m (5 ft) of uncontrolled fill, or loose silt, overlaying about 7.6 m (25 ft) of dense sand and gravel. The silty soils are of loose to medium-dense consistency and have aeolian and fluvial origins.

Annual precipitation at INL averaged 22 cm/yr (8.7 in/yr) from 1951 through 1994. Annual net evaporation from large water surfaces in the Eastern Snake River Plain is 84 cm/yr (33 in/yr) (Rodriguez, et al., 1997).

The INL overlies the Snake River Plain Aquifer, the largest aquifer in Idaho. This aquifer is the major source of drinking water for southeast Idaho and has been designated a sole-source aquifer by EPA. This aquifer underlies the Eastern Snake River Plain and covers an area of about 24900 km^2 (9611 mi^2). The aquifer flows to the south and southwest. Depth to the top of the aquifer ranges from 61 m (200 ft) in the northern part of INL to about 274 m (900 ft) in the southern part (Rodriquez, et al., 1997). The aquifer, with estimates of thickness ranging from 76 m (250 ft) to more than 914 m (3000 ft), consists of thin basaltic flows, interspersed with sedimentary layers.

Geotechnical Stability Considerations

The following issues emanate from the geotechnical stability considerations:

- Requirements and specifications for excavation and backfill at a given site.
- Liquefaction susceptibility evaluation at a given site

- Foundation settlement dependent upon soil conditions, structure, geometry and loading, and the magnitude of settlement a structure can tolerate without adversely affecting performance.
- Existence of soft zones and the potential for settlement at a given site.

Natural Phenomena Hazards

This section identifies and describes natural phenomena events considered potential accident initiators. Natural phenomena hazards are comprised of floods, high winds, tornadoes, tornado missiles, hurricanes and earthquakes. With possible exception of the external floods, the effects of the natural phenomena are considerably reduced for an embedded structure. Above grade structures must be designed to withstand these events.

Floods

The probable maximum flood (PMF) water level for the Mitsui – South Texas site is 30.5 cm (1ft) below grade, resulting in a design impact to the embedded structures at this site. For the Savannah River Site, the PMF is 36 m (118 ft) above mean sea level (MSL) with wave run-up, and as high as 50 m (165 ft) with wave run-up. The minimum plant grade at this site is approximately 76 m (250 ft) above MSL, so most of the embedded structures would be above the flood stage, resulting in minimal impact on an embedded structure and no impact on above grade structure from this event. The PMF water level for the Idaho National Laboratory facility is about 1500 m (4921 ft), with water velocities of about 0.3 to 1 m/s (1 to 3 ft/s), where the final graded ground surface elevation for the site is 1498.7m (4917 ft), so both embedded and above grade structures would be designed for this event.

The information related to floods and other natural phenomena parameters pertaining to the three sites considered is compiled in Table 4-3.

Earthquake

As stated before, recent developments in soil-structure interaction have generally confirmed that the deeper a structure is embedded into the ground, the smaller the seismically induced accelerations and stresses it has to endure. In general, foundations constructed on soft soils have a tendency to amplify the seismic motion, whereas those on bedrock do not. If seismic effects are to be minimized, it is advantageous to embed the structure sufficiently and place the foundation on firm soil or bedrock.

SRS

The Savannah River site is located with the Coastal Plain physiographical province of South Carolina. However, seismic activity associated with the SRS area displays characteristics more closely associated with the Piedmont province, with a marked lack of clustering seismic zones. The activity is more characteristic of the occasional energy strain release occurring through a broad area of central Piedmont. Two earthquakes of MMI III or less have occurred with epicentral locations within the boundaries of SRS. On June 9, 1985, an intensity III earthquake with a local duration magnitude of 2.6 occurred at SRS. Another event occurred onsite August 5, 1988, with an MMI I-II and a local duration magnitude of 2.0. Neither of these earthquakes triggered the seismic alarms (setpoint 0.002g) at SRS facilities. Another event with MMI estimated at IV-V with a duration magnitude of 3.2 occurred on August 8, 1993. No alarms were triggered during this event either.

SRS is located on soils (sedimentary strata) ranging in thickness from 600 to 1500 ft overlying crystalline or Triassic basement. Deep stiff soils, such as those present at SRS, severely condition bedrock spectra by frequency-dependent amplification or de-amplification. Depending upon the frequency and amplitude of bedrock motion, the key soil properties controlling the soil spectrum are the soil column thickness, the dynamic properties (strain dependant shear-modulus ratio and damping), low-strain soil shear wave velocity structure, and impedance contrast with the basemat.

INL

Earthquake histories and seismic characteristics for the Idaho National Laboratory (INL) site located in the Eastern Snake River Plain and the adjacent Basin and Range Province are different. The Plain historically has produced only infrequent, small-magnitude earthquakes (King, et al., 1987; Pelton, et al., 1990; Woodward-Clyde Consultants, 1992; Jackson, et al., 1993). Larger historical earthquakes and active faulting are associated with tectonic activity in the Basin and Range Province. For example, the 1959 Hebgen Lake Earthquake (moment magnitude 7.5) occurred about 150 km (93 mi) from the INL. The October 28, 1983, Borah Peak earthquake (moment magnitude 6.9, Richter magnitude 7.3) occurred along the Lost River fault about 100 km (62 mi) from the INL site. Although the Borah Peak earthquake produced peak ground accelerations of 0.022 g to 0.078 g at INL (Jackson, 1985), INL facilities were not damaged significantly (Guenzler and Gorman, 1985).

Major seismic hazards include the effects from ground shaking and surface deformation (faulting, tilting). Other potential seismic hazards (e.g., avalanches, landslides, mudslides, soil settlement, and soil liquefaction) are not likely to occur at INL because the local geologic conditions are not conducive. Based on the seismic history and the geologic conditions, earthquakes greater than moment magnitude 5.5 (and associated strong ground shaking and surface fault rupture) are not likely to occur in the Plain. However, moderate to strong ground shaking from earthquakes in the Basin and Range Province can affect INL. Researchers use patterns of seismicity and locations of mapped faults to assess potential sources of future earthquakes and to estimate levels of ground motion at the site. The sources and maximum magnitudes of earthquakes that could produce the maximum levels of ground motions at INL include the following (Woodward-Clyde Consultants, 1990, 1992):

- A moment magnitude 7.0 earthquake at the southern end of the Lemhi fault along the Howe and Fallert Springs segments;
- A moment magnitude 7.0 earthquake at the southern end of the Lost River fault along the Arco segment;
- A moment magnitude 5.5 earthquake associated with dike injection in either the Arco or Lava Ridge–Hell's Half Acre Volcanic Rift Zone and the Axial Volcanic Zone; and
- A random moment magnitude 5.5 earthquake in the Eastern Snake River Plain.

South Texas Project (STP) site
Seismic activity at the STP site (Matagorda County, Texas) is extremely low. The peak ground acceleration in the area is less than 0.02g for a seismic event with a 2500 year return period (2% probability of exceedance in 50 years). Hence, seismic loading does not govern the design of either embedded or above grade structures.

External Hazards

Man-made external hazards (such as air plane crash) have also been considered in Table 4-3. Physical security including design basis threats has been discussed separately in this section of the report.

Table 4-3: Effect of Site Parameters on Embedded /Above Grade Construction at Three Sites

Subject	Mitsui – South Texas		Savannah River Site		Idaho National Laboratory	
	Embedded	**Grade Structure**	**Embedded**	**Grade Structure**	**Embedded**	**Grade Structure**
High water table	>61cm (2ft) below grade: Heavier structure design due to hydrostatic pressures/ uplift Higher construction costs due to dewatering	Construction of basement at a depth of 10 m requires de-watering	About 13m below grade. Control of groundwater requires sump pumps or slurry walls. High construction cost.	Construction of basement at a depth of 10 m requires de-watering	No effect	No effect
Seismic Liquefaction Potential	Possible Additional analyses and structural considerations required	Possible Additional analyses required	Not likely	Not Likely	None	None
Extreme wind	No Effect	215 km/h (140 mph) for 3- second wind gust, 100 yr recurrence interval. Structures designed for high wind loads.	No Effect	140 km/h (87 mph) fastest mile wind speed, 100 yr recurrence interval. Structures designed for moderate wind loads.	No Effect	274 km/h (171 mph) with a probability of 1×10^{-7} /yr Structures designed for highest wind loads.
Probable Maximum Flood	30.5 cm below grade. Plant level 9 m above MSL. Consider design impact.	No design impact.	36 m above MSL. Plant level 76 m above MSL. Minimal impact to the design.	No design impact	1500 m above MSL. Plant level 1498.7 m. Design for the water head and leakage.	Minimal design impact.

Subject	Mitsui – South Texas		Savannah River Site		Idaho National Laboratory	
	Embedded	**Grade Structure**	**Embedded**	**Grade Structure**	**Embedded**	**Grade Structure**
Maximum flood	414.5 cm (13.6ft) above nominal plant grade. Structures to be designed for hydraulic head and leakage potential.	Must be addressed in structural design. Safety systems to be located in flood-protected enclosures.	2590 cm (85ft) below grade. Structures to be designed for hydraulic head and leakage potential.	No design impact	130 cm (4.3ft) above grade. Structures to be designed for hydraulic head and leakage potential.	Minimal design impact. Safety systems to be located in flood-protected enclosures.
Maximum tornado wind speed	No Effect	322 km/h (200 mph) Structures designed for moderate tornado loads and tornado missiles	No Effect	467 km/h (290 mph). Structures designed for high tornado loads and tornado missiles	No Effect	459 km/h (285 mph). Structures designed for high tornado loads and tornado missiles
Site proximity missiles/aircraft	No Effect	Missiles: None. Aircraft: 1.16×10^{-7} per year. No structural considerations.	No Effect	Missiles: None Negligible air traffic hazard (4000 flights per year) No structural considerations	No Effect	10^{-7} per year No structural considerations

4.3.9 Cost

4.3.9.1 General

The purpose of this cost estimate is to evaluate rough order of magnitude construction cost estimates for the Reactor Building. Additional concrete and structures needed to protect the Reactor Building against natural phenomena and external hazards for the above ground construction option are also included.

The estimates address two construction scenarios, one below grade, and the other above grade for the INL and the Savannah River sites.

This estimate is based on pre-conceptual designs of the Reactor Building and structures needed to protect the Reactor Building described in section 6.0 of this report.

The estimate uses the cost model from previous BREI NGNP Reactor Building estimates adjusted for scale and volume. The estimates for all scenarios assume direct costs limited to earthwork, cast in place concrete and structural steel. Local productivity rates have been factored into the direct labor rates.

4.3.9.2 Estimate Clarifications

Both Savannah River and INL Sites

- The above ground scenarios include additional protective measures for the Reactor Building.
- All building concrete is estimated using the NQA-1 specification
- Burns and Roe Construction Management was consulted for the dewatering systems applied mainly to the Savannah River Site (SRS).

INL Site

- Excavation costs were modeled based on previous Burns and Roe NGNP estimates for the INL site. The estimate is based on excavating common earth, bed rock, compaction, and hauling utilizing subcontractor labor. The estimate utilizes information from an experienced subcontractor with familiarity working with the DOE / INL site specific procedures and excavation and dewatering techniques.

- The above grade estimate assumes bedrock at 5 m below grade with rock excavation down to the base mat level.

4.3.9.3 Below Grade Construction at Savannah River Site

The estimates address two methods for dewatering: Slurry Wall and Well Dewatering.

- The estimate assumes that the excavations will be sloped at 1:4 ratio to provide the site-specific safety requirements to ensure a safe working environment.
- Assumes groundwater table at 13 meters.
- Assumes mechanically stable earth.

Slurry Wall System

A slurry wall system will be installed for the below ground scenario consisting of W36x245 beams @ ~3 m on center with attached concrete panels. This estimate utilizes clamshell excavation, bentonite slurry with a density of 150 lb/cy with 4000 lb concrete.

- Assumes slurry wall excavation is ~ 1 meter wide.

- Assumes slurry wall will be installed 60 m below grade to bedrock. Note that the slurry wall dewatering method will not be practical if the bedrock is not present at 60 m. The slurry wall system requires bonding to the bedrock to prevent wash out.

Well Dewater System

This method is recommended by a specialty contractor New England Dewatering, LLC and consists of the following steps:

- Drill for wells, 30 cm bore holes, 60 m deep
- Install 20 m high stainless steel screen at the lower end of these holes
- Install 40 m high casings on the top of the stainless screens

- Fill annular space between casing and sides of hole to within 6 m of surface with grits or well gravel. Place concrete cap above packing to 6 m down preventing ground water from leaching into dewatering well.
- Install appropriate dewatering pumps in each well
- Discharge water at least 300 m from the dewatering site. Collect sediment from dewatering wells in large capacity FRAK tanks.

4.3.9.4 Construction Cost Estimate

Rough order of magnitude estimates for the above ground and embedded construction for the INL and SRS are summarized below.

Site	**Rough Cost Estimate, Millions of 2008 Dollars**			
	Embedded			**Above Grade**
	Slurry Wall System	Well Dewater System	Rock Excavation	
INL			58	39
SRS	122	102		38

The below grade construction is 50% more costly than the above grade construction at the INL site. Below grade construction is very costly compared to the above grade construction at the SRS mainly due to large volume of soil to be disposed and dewatering needs for about 3 years of site preparation and construction.

5.0 PRELIMINARY TECHNICAL AND FUNCTIONAL REQUIREMENTS

The Reactor Building consists of a cylindrical reinforced concrete design, containing the reactor vessel, the Intermediate Heat Exchanger vessel (IHX), Steam Generator, the process heat IHX, the primary side Cross Vessels and Circulators, and the secondary high temperature piping, circulators and supporting equipment. The Reactor Building contains all nuclear safety related equipment for the plant including the RCCS tanks.

The reactor and IHX vessels are enclosed in cavities of vertical cylinders of reinforced concrete. The reactor vessel and IHX vessels are completely separated, with the exception of required openings and movement space needed for proper operation. The Reactor Building internals will be designed to provide biological shielding, seismic integrity, external missile protection, internal missile protection, airplane crash and resultant fire protection, and to sustain all internal and external loading conditions which may be reasonably expected to occur during the life of the plant.

Using the geometry described for the power generation system, a typical Reactor Building outside diameter for the cylindrical portion of the structure of 34.00 meters was derived. Also, based on preliminary refueling machine and control rod drive mechanism design information, the reactor head to refueling floor dimension of 14 meters was established. This sets the centerline of the cross vessel at 32.25 meters below the refueling floor elevation. These dimensions are used in the reactor building embedment evaluation.

The reactor cavity is equipped with cooling panels to protect the surrounding concrete from the high temperatures generated by the un-insulated reactor vessel. This cooling is provided by a water filled panels connected to the reactor cavity cooling system (RCCS) tanks, located above the reactor vessel. The RCCS provides passive cooling to the reactor cavity via conduction and natural circulation. The RCCS also surrounds and cools the reactor vessel support system located in the lower levels of this area.

The following assumptions on the performance and quality of particle fuel are necessary for the selected preliminary technical and functional requirements of the reactor building.

- Low as manufactured defect (fuel quality)

- Low in-service failure rate (fuel quality and reactor operating parameters)
- Low failure during heat-up accidents (fuel quality and reactor operating parameters)

Given the fuel performance and quality assumption above, the following requirements are applicable to the reactor building or containment structure in which the reactor is located.

1. The reactor building shall be designed as a vented building with isolation capability after release of initial circulating inventory.
2. The reactor building as an element of the NGNP radionuclides containment system shall provide sufficient reactivity retention to ensure PAG radiological dose limits at plant EAB are not exceeded.
3. The allowable reactor building leak rate shall be such that PAG limits at the plant boundary do not exceed 1 rem / yr for duration of any postulated accident
4. The reactor building shall be capable of withstanding challenges to its structural integrity from external and internal events
5. Following a depressurization accident, the reactor building design shall vent through plant stack to near-atmospheric pressure
6. Following depressurization reactor building release streams shall be filtered sufficiently to ensure PAG limits at the EAB are not exceeded
7. The reactor building internal structures shall provide suitable shielding to permit personnel access for maintenance activities to any required equipment during refueling or schedule reactor shutdown periods
8. The reactor building design shall not negatively impact refueling operations

The above preliminary T&F requirements are specific to the Reactor Building. The T&F requirements for the Auxiliary Building and Fuel Building where new and spent HTR fuel are located are out side of the scope of this task.

A review of current NRC regulation was performed to identify regulatory exposure limits. The regulatory requirements for occupational (onsite) and environmental (off-site) exposure limits are listed in Table 5-1.

Table 5-1: Regulatory Requirements which control Radionuclide Release

Regulation	Control Function	Offsite Dose Limits
10CFR20	Occupational dose limits	• Whole body ≤ 5 rem/plant yr • Thyroid ≤ 15 rem/plant yr
10CFR50, Appendix I	Plant effluents during normal operation and anticipated operational occurrences (AOOs)	• Whole body ≤ 5 mrem/plant yr • Thyroid ≤ 15 mrem/yr at the EAB
10CFR50.34	Offsite dose limits during design basis events (DBEs)	• Total Effective Dose Equivalent (TEDE) ≤ 25 rem/event at the EAB
EPA-520 Protective Action Guidelines (PAGs)	Offsite dose limits during DBEs and beyond design basis events (BDBEs) at which sheltering is considered	• Whole body < 1 rem/plant yr • Thyroid < 5 rem/plant yr • TEDE ≤ 1 rem/plant yr at the emergency planning zone (EPZ)

Two user requirements which are used in prior designs of the U.S. modular gas cooled reactor plants have been adopted. These are:

1. Occupational exposures shall be ≤ 10% of 10CFR20, and
2. Off-normal events shall not exceed the sheltering PAGs offsite for a 425 meter EAB

Taken together, the controlling requirements from the regulator and the utility/user are 10CFR50, Appendix I for AOO and the PAGs for the DBE and BDBE, both at the site boundary.

6.0 POTENTIAL ALTERNATIVE DESIGNS

The reactor vessel, IHX vessels, and SGUs are located in enclosures of vertical structures of reinforced concrete within the Reactor Building. The reactor vessel, IHX vessels, and SGUs are completely separated, with the exception of required openings and space needed for proper installation, inspection and maintenance. The Reactor Building and its internal structures are designed to provide biological shielding, seismic integrity, external missile protection, internal missile protection, airplane crash and resultant aviation fuel fire protection, and to sustain all internal and external loading conditions which may be reasonably expected to occur during the life of the plant.

The primary helium gas is contained by the reactor vessel, the crossover vessel between the reactor vessel and the IHX, and the IHX vessel. The Reactor Building is vented directly to the atmosphere in the event of a major gas leak. The Reactor Building is designed to collect escaping primary and secondary gases for minor leaks up to 65 mm diameter but does not function to contain the pressure resulting from a major gas leak, greater than 65 mm diameter. The collected minor leakage will be processed through the plant waste gas treatment and HVAC filtering systems for eventual release to the atmosphere through the plant vent. A rupture panel or similar passive device will be designed and provided to maintain the normal environmental boundary, but open in the event of the major leak. Isolation mechanisms will be used to close the vent path thereafter.

The Reactor Building outside diameter for the cylindrical portion of the structure would be approximately 33.5 meters. Based on preliminary refueling machine and control rod drive mechanism design options, the reactor head to refueling floor dimension was established. This sets the centerline of the cross vessel at approximately 32 meters below the refueling floor elevation. These dimensions establish the resulting building elevations.

The reactor cavity is provided with a cooling system to protect the surrounding concrete from the high temperatures generated from the un-insulated reactor vessel wall. This cooling is performed by a "water wall" reactor cavity cooling system (RCCS) connected to the cooling system supply tanks located above the reactor vessel. The RCCS provides passive cooling to the reactor cavity via conduction and natural circulation. The RCCS also surrounds and protects the reactor vessel supports located in the lower levels of this area.

A section view of the Reactor Building is provided in Figure 6-3 to Figure 6-5.

Refueling Floor

The refueling floor is located on top of the Reactor Building; it contains an access hatch for fit up to the refueling machine, within a removable hatch for removal of the reactor head or access to the internals. A large portion of the Reactor Building ends at floor elevation above the IHX.

All hatches are accessible by the Maintenance Enclosure crane above the Reactor Building. The reactor vessel hatch is also accessible by the refueling machine positioner. Both the reactor vessel and the module fuel storage are accessible by the fuel server. Embedded rails are provided for both the refueling machine positioner and the fuel transfer system.

Below the refueling floor, the Reactor Building is sub-divided into compartments for specific equipment and functions. Chases are provided for routing supporting cables, piping, and HVAC, as well as chases for venting the reactor and IHX compartments for major leaks and breaks through a discharge path to the filter area. Stairs and elevators are provided external to the Reactor Building for access to all floors.

Vessel Head Elevation

The Reactor Building at and above the reactor vessel head elevation consists only of the reactor cavity, which houses the control rod drive (CRD) mechanisms and RCCS headers. The height of the CRDs depends on the CRD mechanism design which has not been defined yet. A traditional bayonet design requires more head room than a motor and pulley type design driving a chain link CRD mechanism.

Reactor Cavity

The reactor cavity houses the reactor vessel, RCCS cooling elements and the vessel support systems. The spaces out side of the reactor cavity are occupied by the IHX and SGU compartments. Maintenance access for inspection and removal of equipment through elevators doorways, stairs and hatches are provided throughout the Reactor building compartments.

Reactor Building Floor Elevation

The lowest floor in the Reactor Building is the basement of the building. This floor provides access to the shutdown cooling system (SCS) circulator, access to the bottom of the IHX and SGU vessels, and contains sumps and sump pumps for the respective cavities. The floor elevation will be established to provide sufficient space below the reactor vessel for removal of the shutdown cooling circulator and heat exchanger into a shielded container for removal.

An access door is provided into the reactor cavity. The door is sized to allow passage of the shielded cart used for removal of the SCS circulator. A door is also provided through the Reactor Building wall to allow passage of this cart into the Auxiliary Building, for subsequent removal through the Auxiliary Building equipment access shaft.

Access is also provided by a large door into the IHX cavity for inspection and maintenance. In addition, access is provided to the lower (hot pipe) isolation valves for maintenance. The isolation valve auxiliary equipment will be located on this floor level inside the Reactor Building.

The doors to the IHX and reactor cavities also serve as pressure relief doors to vent major primary and secondary coolant releases from these cavities. The released coolant escapes through the doors into the large area outside the reactor and IHX cavities for further expansion and subsequent routing into the filter area. Failures at the secondary coolant piping in the Reactor Building will also vent into these areas.

Once the pressure from the coolant release in the large area reaches a pre-determined point, a series of pressure-relief dampers or valves will allow the coolant to escape to vent shaft, where it will be directed to release points above the operating floor, safely releasing the excess Reactor Building pressure to the atmosphere or the filter area. Once building pressure drops to acceptable levels in this vent path, the reactor building is isolated. Long term reactor building atmosphere cleanup and filtrating is performed through the active filter system.

6.1 Fully Embedded Reactor Building with and without Filtration – Indirect Power Cycle

Under a separate task AREVA provided several plant layout alternatives for the NGNP. The indirect power cycle layout with both IHX and SGU located in the reactor building was selected for embedment evaluation, see Figure 6-3. Also added to this layout is a concept of building filtration system in case of a depressurization accident would filter the primary system helium circulating radioactivity before it is vented to the atmosphere. The need for such filter system is discussed in Section 6.4.

Filtrations System

Background

The approach for the selection of the NGNP radionuclide source term for accidents and for the design of containment concept is derived from the slow response of the reactor to core heatup events which results in a prompt and delayed portion of the source term. The prompt source term involves radionuclide releases which occur early as a result of depressurization of the primary system. The prompt source term contains radionuclides circulating with the helium, but is dominated by liftoff of radionuclides previously deposited on surfaces within the primary system. Although uncertainties exist, it is expected that the prompt source term is small enough that the radionuclides could be vented from the Reactor Building unfiltered through a stack.

The delayed source term develops from releases from fuel during the subsequent lengthy core heatup phase of the accident, possibly requiring filtration through HEPA filters. Because the delayed release occurs at atmospheric pressure, a conventional high pressure, low leakage containment is not necessary as it serves no identifiable function.

Filtration System Design and Arrangement

The filtration system is capable of directing coolant leaking from the primary coolant boundary in the reactor cavity to a set of high temperature and pressure filters during the initial transient until the pressures equalize. Subsequently, the high temperature filters are isolated and forced venting of the reactor cavity takes place through a HEPA filtration system to maintain negative pressure in the cavity. See flow diagram - Figure 6-1.

When the primary coolant pressure boundary is ruptured (i.e. a depressurization accident), the coolant is released within the cavity, pressurizing the cavity rapidly to the point of rupturing rupture panels located at the bottom of each of the six vertical chases (starting immediately above the reactor ledge support) in each corner of the hexagonal reactor cavity. The vertical chases connect to a hexagonal horizontal ring header at the very top in the reactor cavity as shown on the drawings.

The high temperature filters comprise of sand filter media. Expansion of the primary coolant at an initial pressure of 5 MPa would result in choked flow conditions either at the point of leak or downstream along its path of discharge (such as at the orifice). Under choked flow conditions the flow is calculated to be about 12 kg/sec for a flow area of 0.33 sq-ft (300 sq-cm). Typically, sand filters are comprised of several layers (7 to 10), with a total bed depth of ~2 meters. The inlet temperature could be significantly lower that core inlet temperature because the helium is passively cooled by expansion and natural convection with the building structures before it reaches the filter media. Because of the low permissible face velocities through the sand filters, the filter size would be very large (~7000 sq m) as shown on the drawings. A schematic diagram of a typical nuclear sand filter is shown on Figure 6-2.

The reactor cavity volume excluding the reactor volume is approximately 6,000 cu m. To maintain a minimal negative pressure in the lined reactor cavity, under equilibrium venting conditions a HEPA filtration capacity of about 4 cu m per minute is calculated to be adequate to account for leakages through the penetrations and hatches.

Cooling of the Reactor Cavity

Cooling of the reactor cavity during the entire transient is accomplished by the redundant safety grade RCCS. This system also functions during normal reactor operation therefore the operability of such system is continuously checked.

General Arrangement

The redundant sand filters are located in hardened enclosures underground at two separate locations as shown on the drawings.

The RCCS water storage tanks and the HEPA filters for reactor cavity venting are located in the Maintenance Enclosure above the Reactor Building in hardened enclosures as shown on the drawings.

Efficient Construction

The Reactor Building is about 50 m deep. Below grade construction reduces construction efficiency in general and especially at high water table sites.

Modular sandwich wall construction reduces construction duration and cost. It also allows parallel construction activities on critical path to be accomplished. Prefabricated section sandwich rings will be used to minimize work effort in the deep silos as much as possible. All modular sandwich rings will be large enough to require hauling by a heavy carrier in order to minimize labor. These sandwich plates form the liners as well as the formwork. Concrete would be directly poured into these sandwich elements.

Use of outside liners for embedded design reduces water migration/leakages into the Reactor Building in high water table areas. However, epoxy coatings applied on the outside surfaces of a below grade Reactor Building would be a second choice.

Epoxy coatings are recommended for the outside surface of a non-embedded Reactor Building construction if conventional construction techniques are used.

Other Considerations

Reactor Cavity

Reactor cavity liner serves several purposes. (However, prevention of activation of concrete is not one of it.) Under a pipe break scenario, helium is first vented from the cavity to outside before the cavity and the Reactor Building are isolated. Following the vent period, releases from the reactor (to the reactor cavity) are filtered through a HEPA filtration system to maintain negative pressure in the cavity. The size of the filtration system is determined by the leakages into the reactor cavity through various penetrations. The liners limit leakage into the cavity.

Another important reason to put a stainless liner in the reactor cavity is to facilitate decontamination during operations (such as post primary coolant release) and future decommissioning of the reactor facility at the end of its life.

Reactor cavity liner would also serve as a shield for jet impingement in case of a pipe break.

Note that the RCCS panel offers possibilities to double as cavity liner.

Reactor Building

As noted above the lined Reactor Building may serve as the containment. The new GE Economic Simplified Boiling water Reactor (ESBWR) does use steel lined reactor building containment.

As noted above liners facilitate decontamination during operations and future decommissioning and could serve as a shield for jet impingement in case of a pipe break.

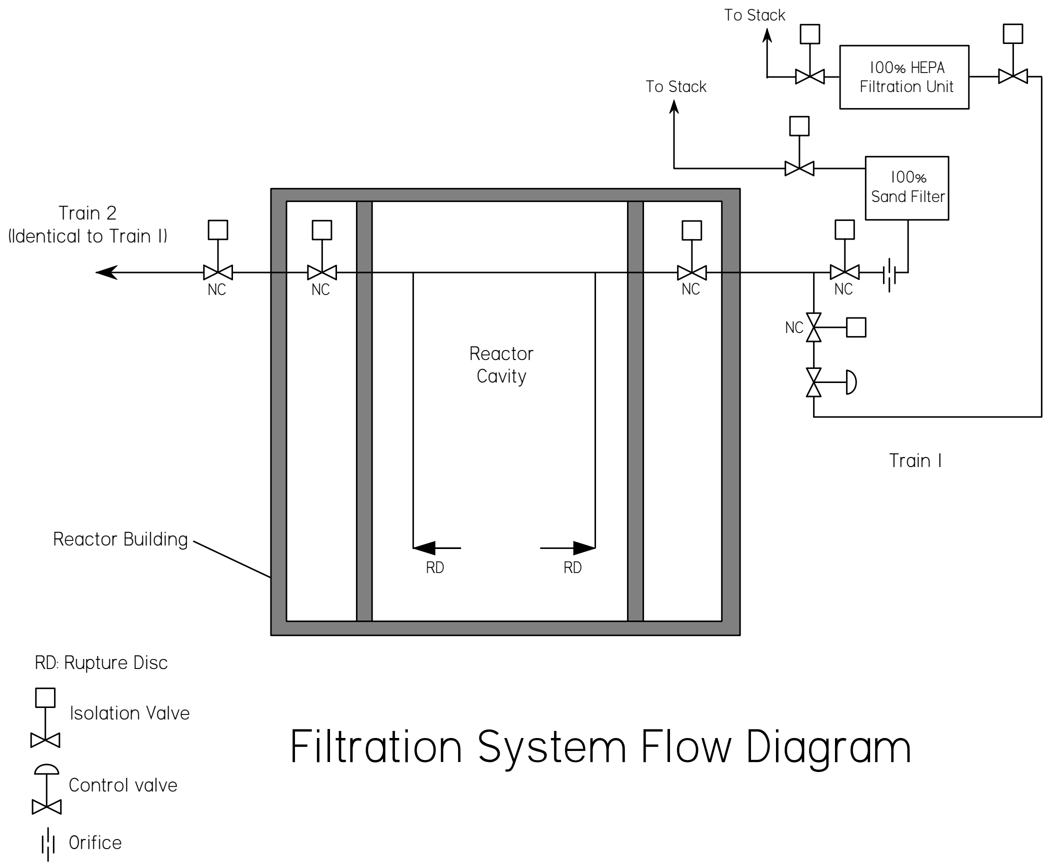


Figure 6-1: Filter System Flow Diagram

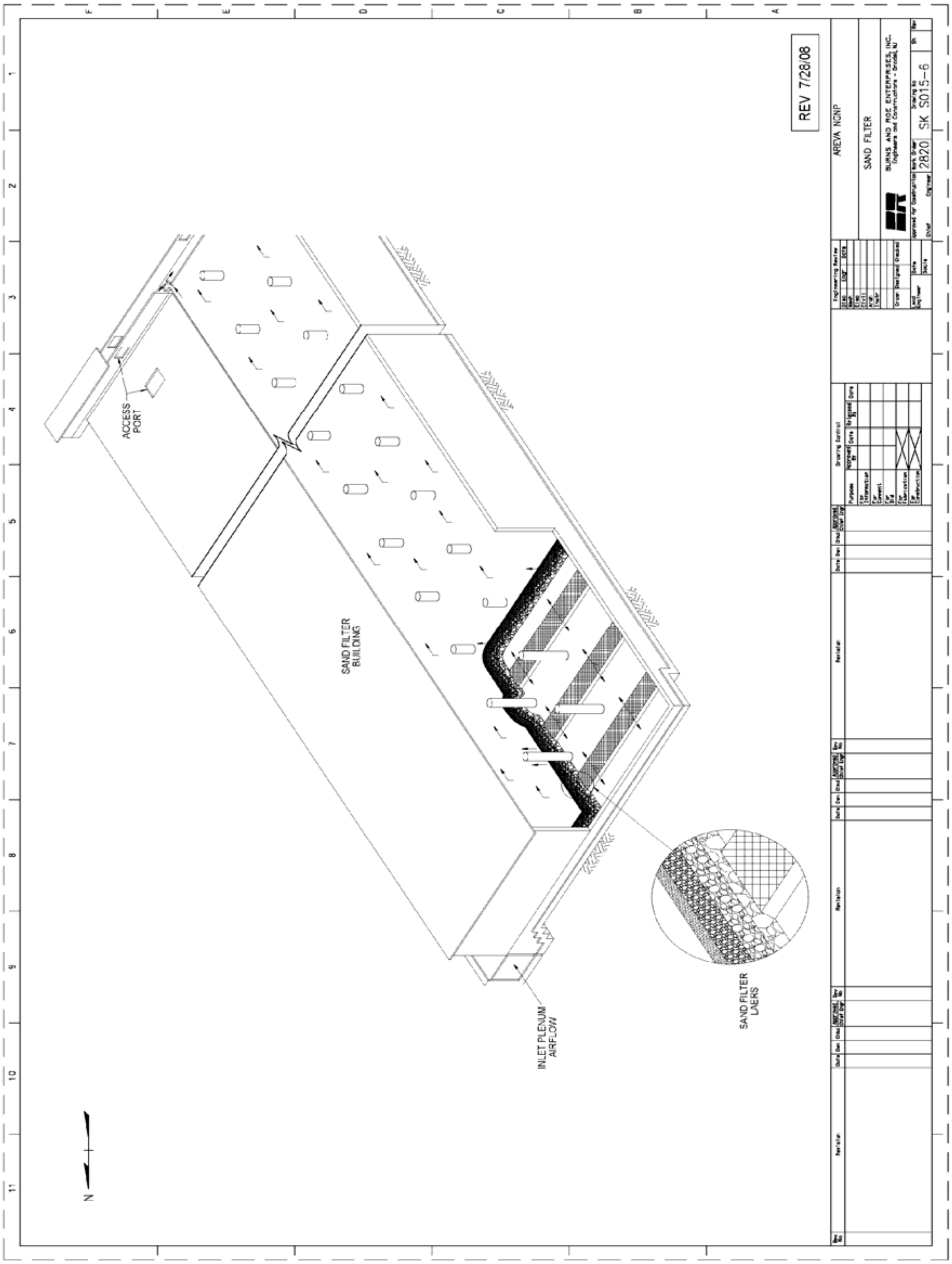


Figure 6-2: Schematic of a Typical Sand Filter - ORNL

Figure 6-3: NGNP Fully Embedded General Arrangement Layout – Plan View

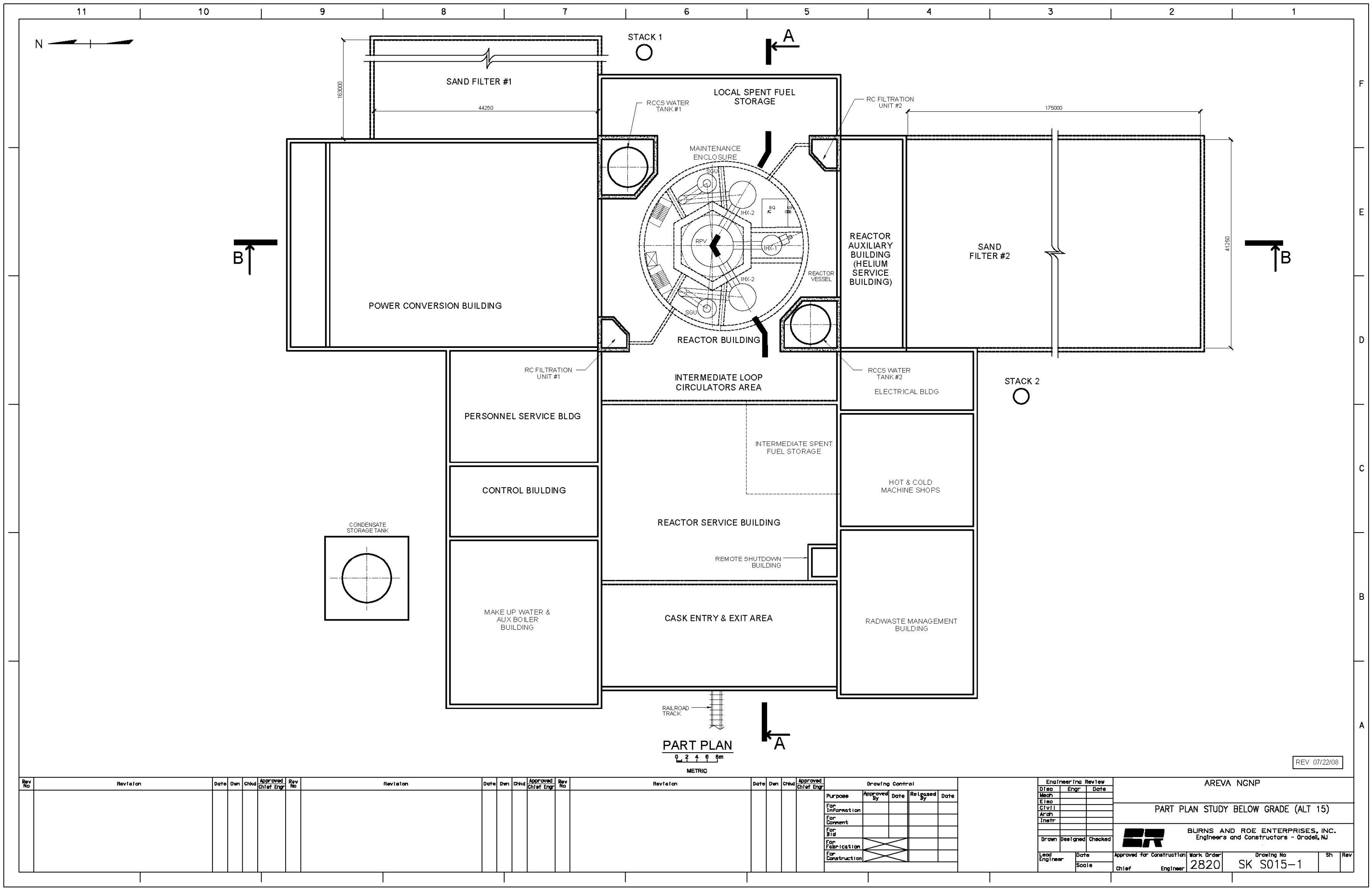


Figure 6-4: NGNP Fully Embedded General Arrangement Layout – Section A-A

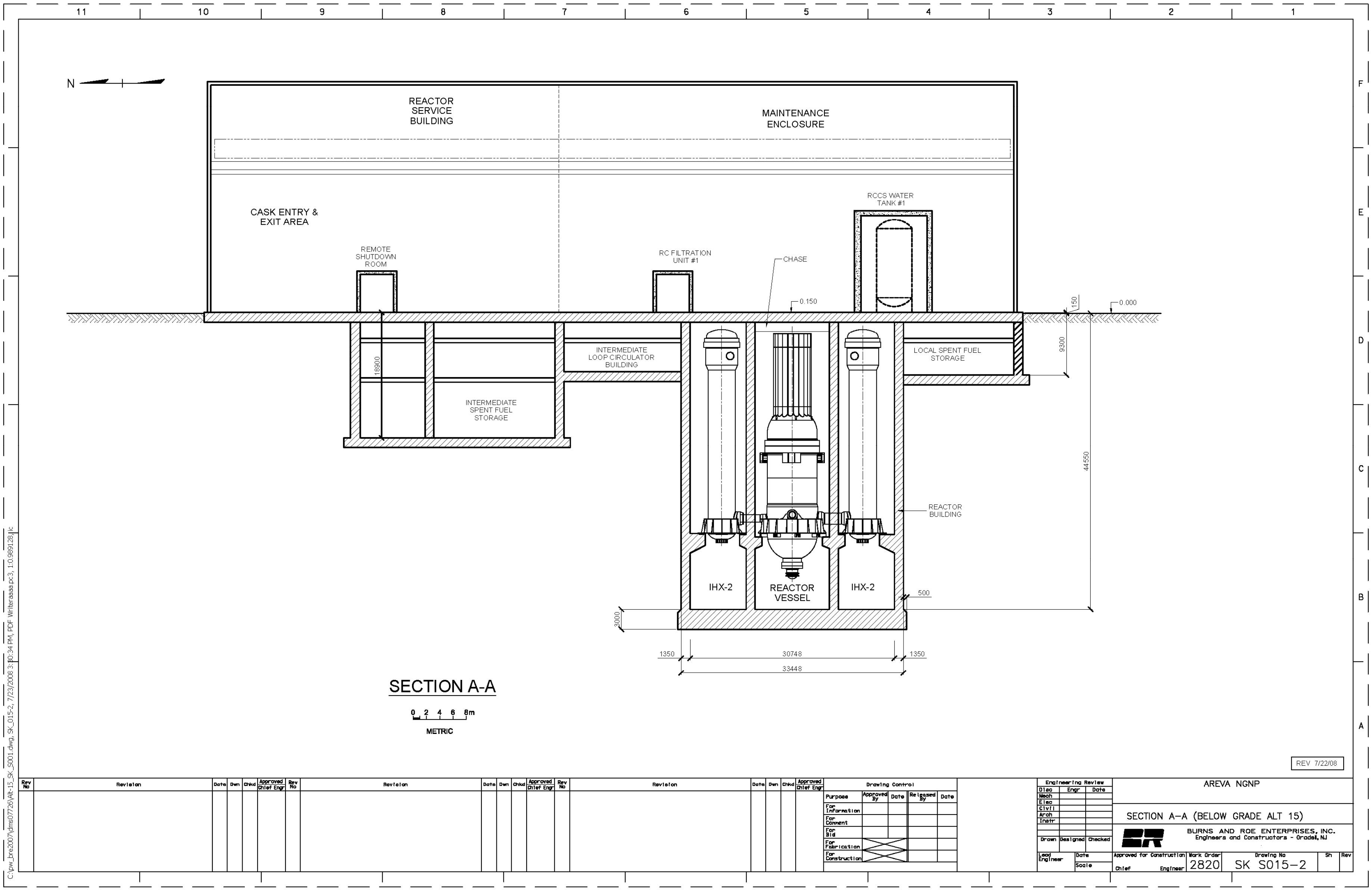
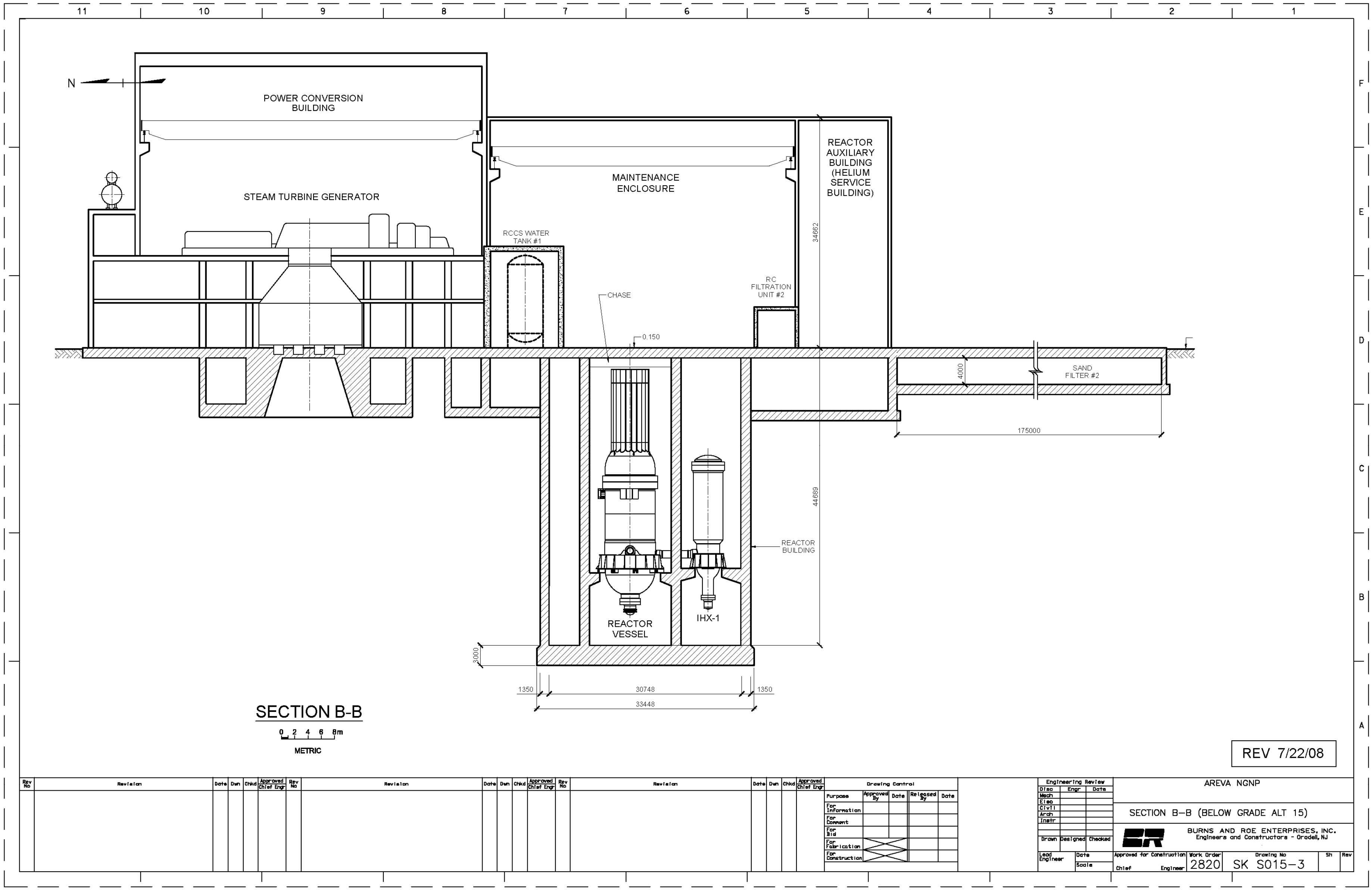


Figure 6-5: NGNP Fully Embedded General Arrangement Layout – Section B-B



6.2 Embedded Reactor Building Configuration for Conventional Steam Cycle

Several layout options were evaluated for the embedded Reactor Building (RB) concepts for the indirect power conversion concepts (and a limited degree of concepts for the direct power conversion) To get a better estimate of the size of the embedded RB employing direct power conversion cycle it was decided to evaluate the layout of the 200 MWe embedded Chinese High Temperature Gas-cooled Reactor-Pebble Bed Modular Project (HTR-PM) direct power conversion cycle. The HTR-PM is a pebble bed reactor. Its additional features are as follows:

Unlike the NGNP designs that show significant space (occupied by the control rod mechanisms) above the reactor vessel, the HTR-PM uses much less space above the reactor head.

The steam generators for the HTR-PM facility are located relatively at a lower elevation than the IHXs/SGs in the NGNP indirect designs. Consequently, the HTR-PM Reactor Building has a stepped design for the reactor cavity and the steam generator areas as shown on Figure 6-6.

Reactor Building Sizes

Based on the NGNP RB layout studies performed to date it is estimated that the NGNP direct power conversion system can be accommodated in a Reactor Building that measures approximately 39 m in diameter and 47 m deep.

The Chinese HTR-PM Reactor Building measures approximately 22m in diameter and 39 m deep and the distance from the centerline of the reactor nozzles to the bottom of the basemat is 16 m (Figure 6-6). These dimensions are simply scaled directly from the Figure 6-6 based on the knowledge that the Chinese reactor measures ~6.0 m in diameter.

While the diameters of the NGNP and HTR-PM Reactor Buildings vary a great deal (39 versus 22 m), the depth of the Reactor Building from the reactor nozzles to the bottom of the basemat for the two designs is nearly the same (16 m). For the purpose of this evaluation the depth of the RB was of great interest. Based on the identical depth of the RB basemat below the reactor nozzle, it is assumed that the RB height should stay as 47 m.

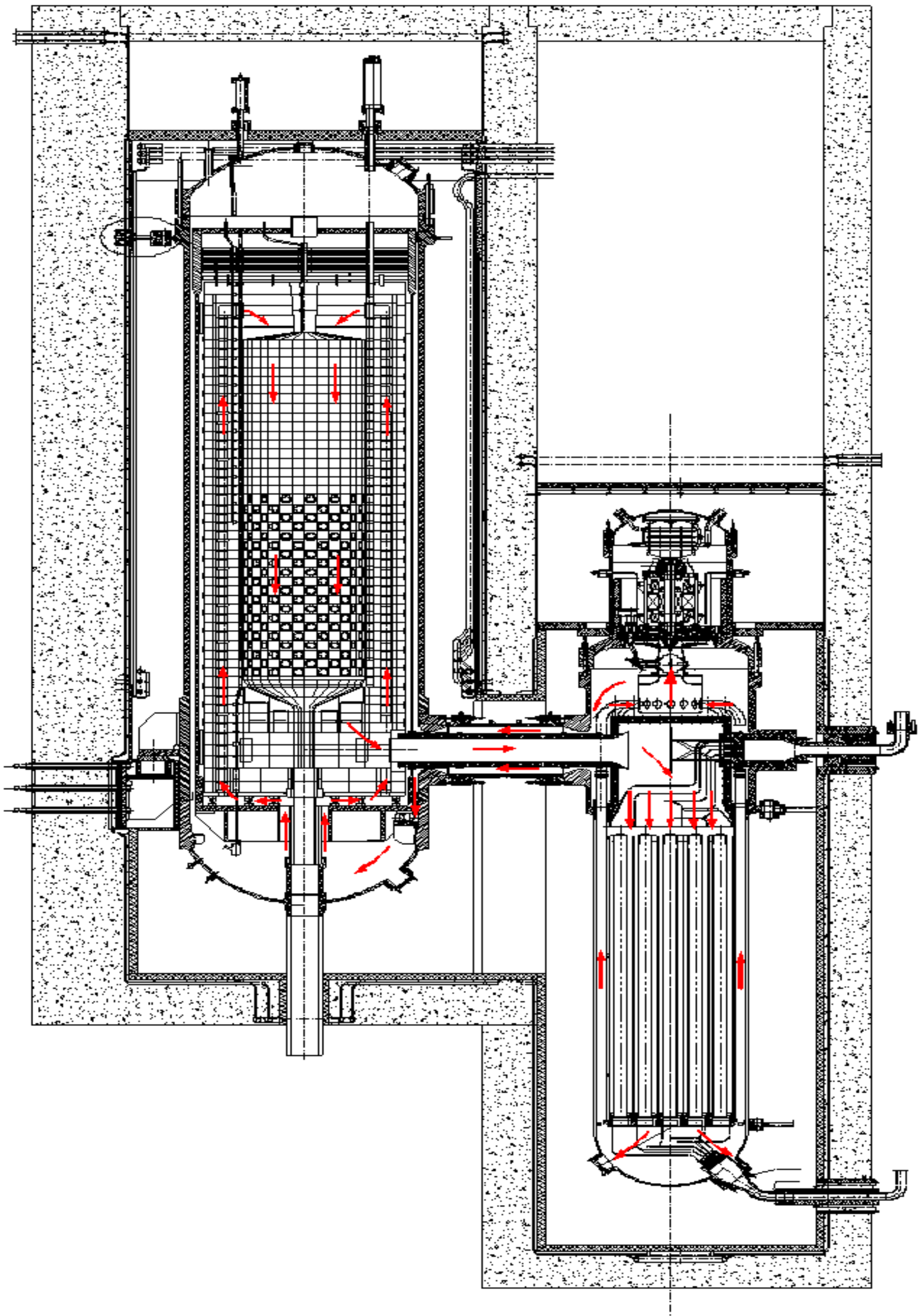
Considerations unique to the conventional steam cycle concepts

For a conventional steam cycle NGNP the steam generator would be located in the reactor building because of shorter piping runs between the reactor and the steam generators. The centerline of the steam generator vessel will be located lower than the reactor vessel for peak cycle operation which also limits liquid transport into the core by gravity drain.

Water ingress accident is the consequence of a tube failure in the steam generator that could result in water leaking into the reactor. A leak is important for several reasons including the concern that the water will react with the hotter portions of the graphite core structure to form carbon monoxide and hydrogen. The protection system would function to isolate the water supply and the water dump system would dump steam generator water inventory so that it is not available for sustained water ingress into the core. Large water entry accident would cause some core damage[2]. The large dump tanks may be located within the RB or in a neighboring building. Locating the Steam generator at a lower elevation of the reactor vessel will naturally reduce the potential water ingress due to gravity. Such layout arrangement has been selected for the HTR-PM and would be recommended for the NGNP.

[2] J.M. Hendrie, Safety of Nuclear Power Plants, **Annual Review of Energy**, Vol. 1: 663-683 (Volume publication date November 1976) .

Figure 6-6: Cross-section of the Chinese HTR-PM Reactor



6.3 Above Ground Reactor Building

The indirect power cycle layout with both IHX and SGU located in the reactor building was selected for the above ground evaluation, see Figure 6-7 to Figure 6-9. Also included in this layout is a building filtration system for primary coolant release to the surroundings.

The discussions under the embedded concept discussed earlier also apply to the partially embedded concept except for the General Arrangements.

General Arrangement

One of the considerations in the layout is to protect the Reactor Building from man made hazards (airplane crash and aviation fuel fire, explosions etc) in addition to the natural phenomena (seismic, flood, high winds, tornado and tornado missiles).

The layout drawings show a hardened structure all around the Reactor Building. Reactor Auxiliary Building (Helium Service Building) is located in a hardened enclosure surrounding the Reactor Building.

The redundant sand filters (if required) are located in hardened enclosures underground at two separate locations as shown on the drawings.

The RCCS water storage tanks and the HEPA filters for reactor cavity venting are located in the Maintenance Enclosure above the Reactor Building in hardened enclosures as shown on the drawings.

Figure 6-7: NGNP Above Grade General Arrangement Layout with Optional Sand Filters– Plan View

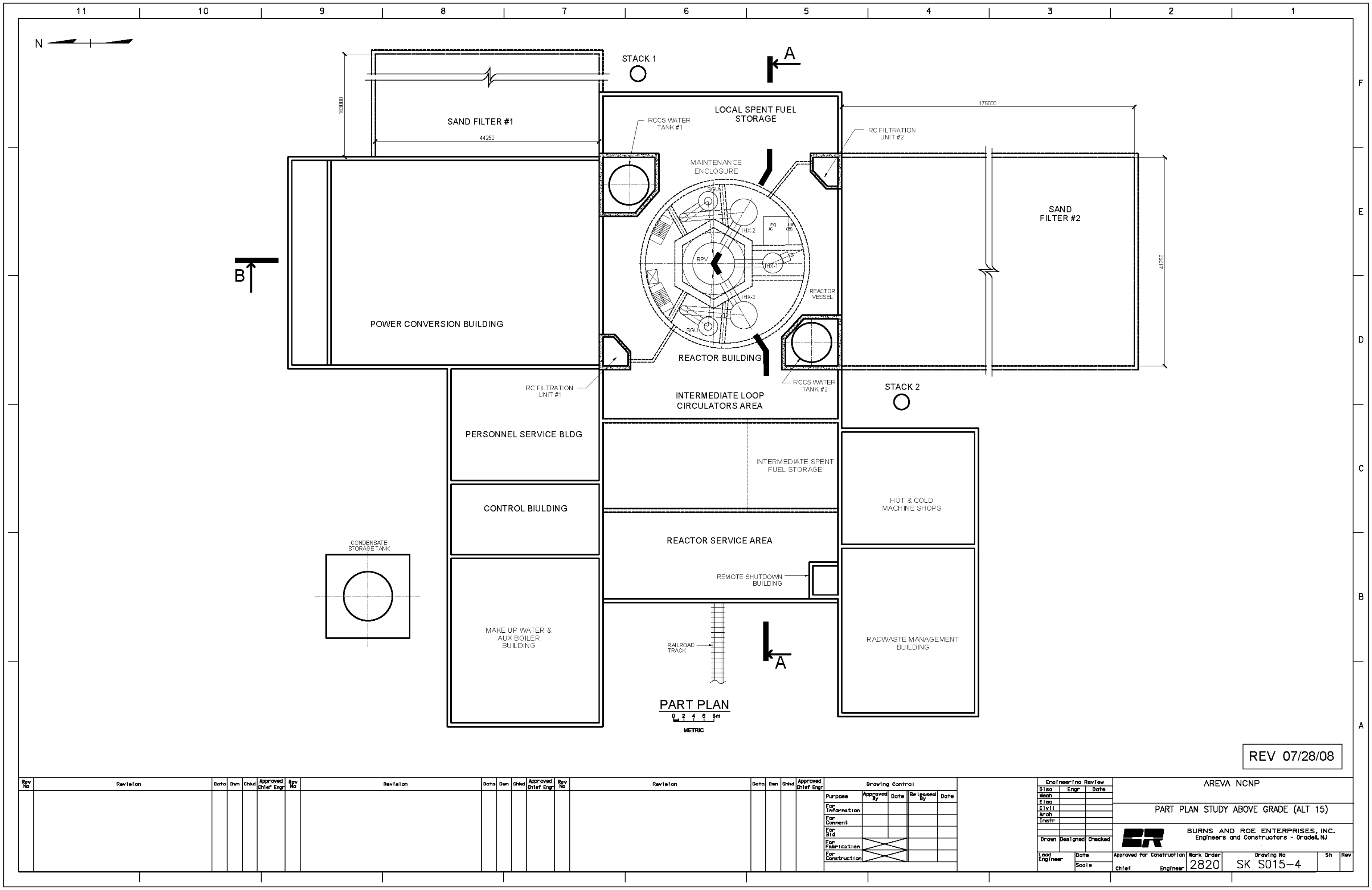


Figure 6-8: NGNP Above Grade General Arrangement Layout – Section A-A

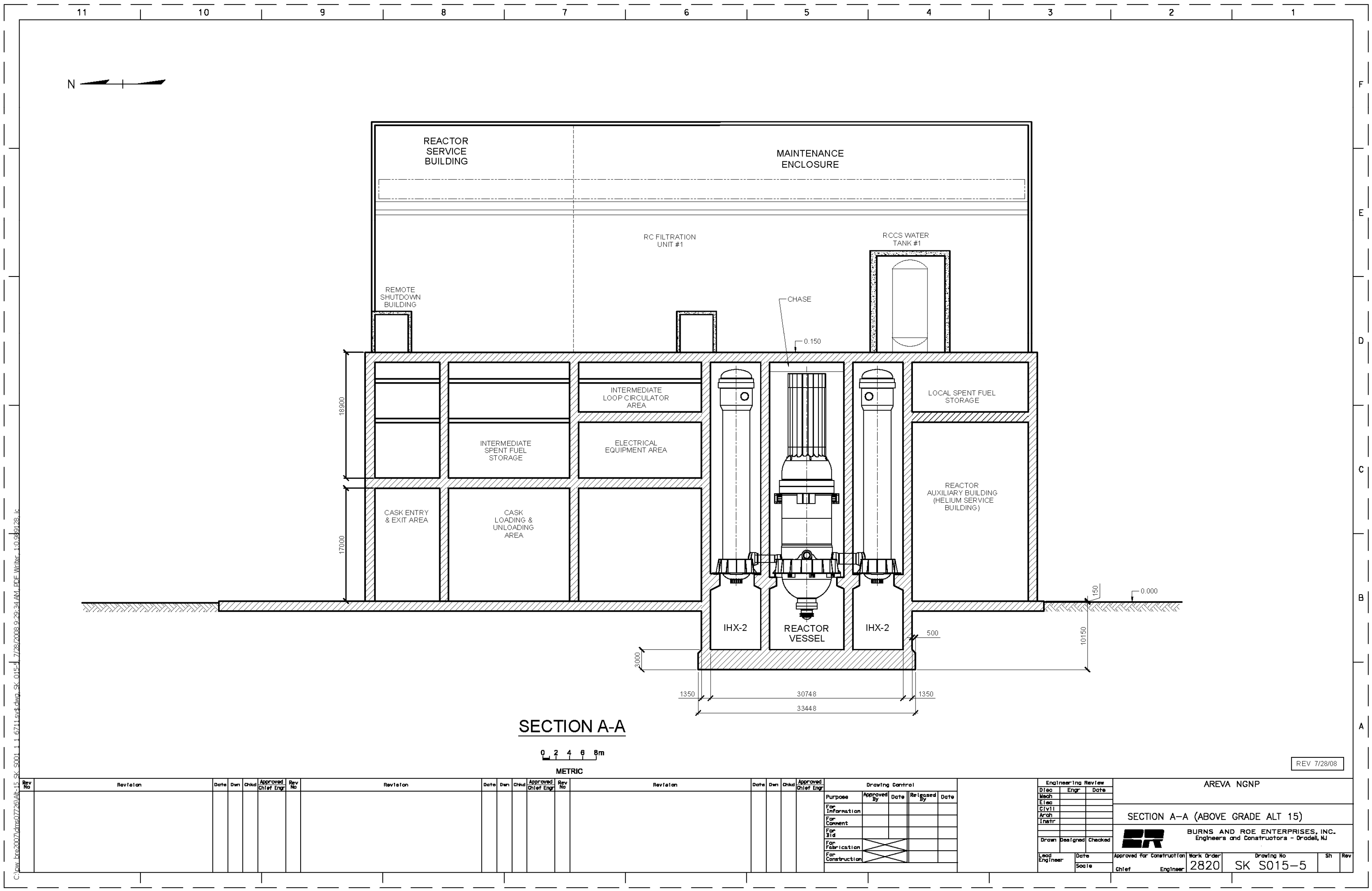
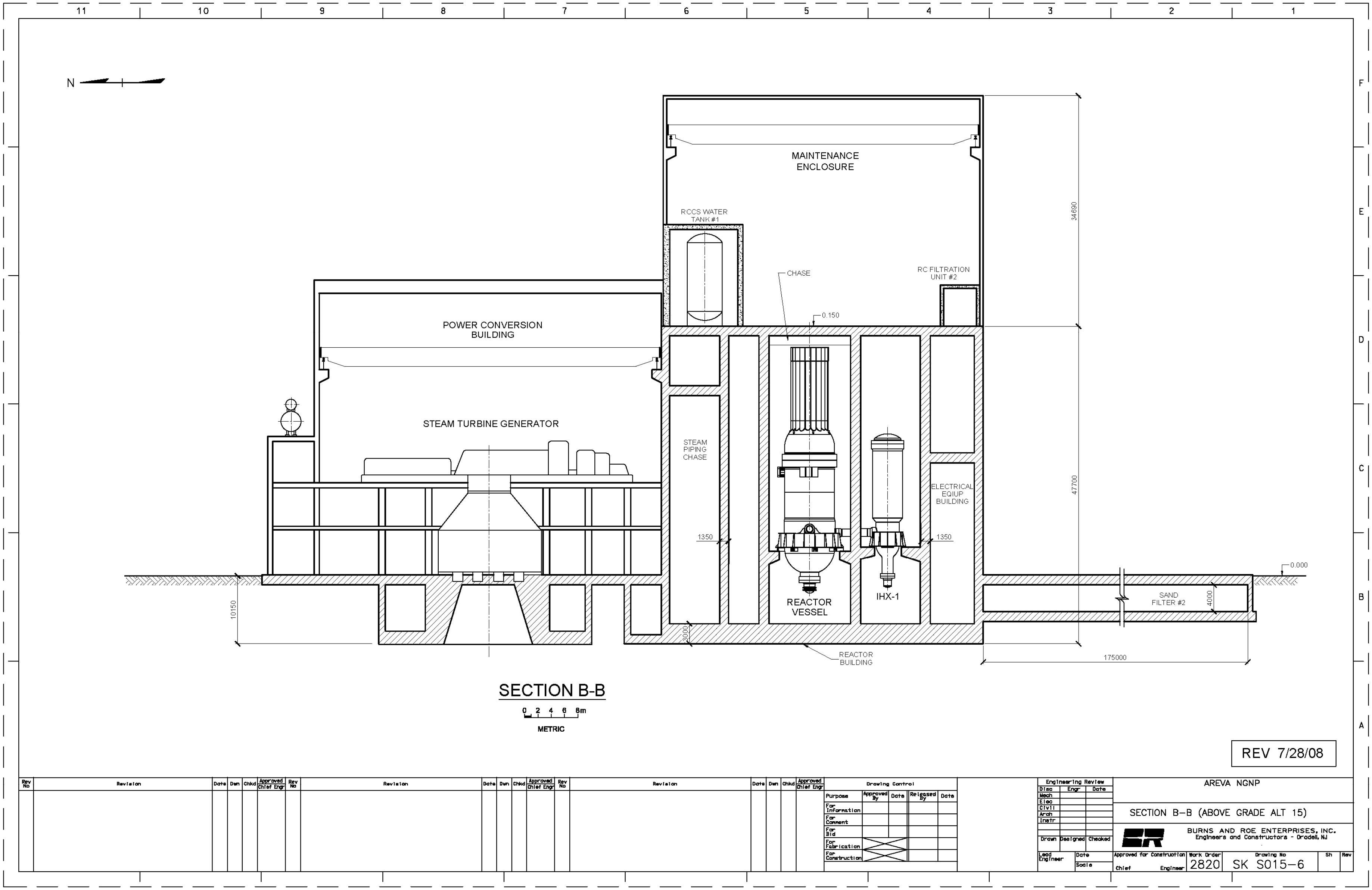


Figure 6-9: NGNP Above Grade General Arrangement Layout – Section B-B



6.4 Partially Embedded Reactor Building

The indirect power cycle layout with both IHX and SGU located in the reactor building was selected for the partially embedded reactor building design evaluation, see Figure 6-10 to Figure 6-12. Also included in this layout is a building filtration system before the primary coolant is released to the surrounding environment.

The discussions under the embedded concept in 6.2 also apply to the partially embedded concept except for the General Arrangements.

<u>General Arrangement</u>

The discussions under the above ground concept also apply to the partially embedded design. The considerations in the layout include protection of the Reactor Building from man made external hazards (airplane crash and aviation fuel fire, explosions etc.) in addition to the natural phenomena (seismic, flood, high winds, tornado and tornado missiles). However, only the structures above grade would need to be designed for these considerations.

The layout drawings show a hardened structure all around the Reactor Building. Reactor Auxiliary Building (Helium Service Building) is located in the hardened enclosure surrounding the Reactor Building.

The redundant sand filters are located in hardened enclosures underground at two separate locations as shown on the drawings.

The RCCS water storage tanks and the HEPA filters for reactor cavity venting are located in the Maintenance Enclosure above the Reactor Building in hardened enclosures as shown on the drawings.

Figure 6-10: NGNP Partially Embedded Reactor Building Design with Optional Sand Filters– Plan View

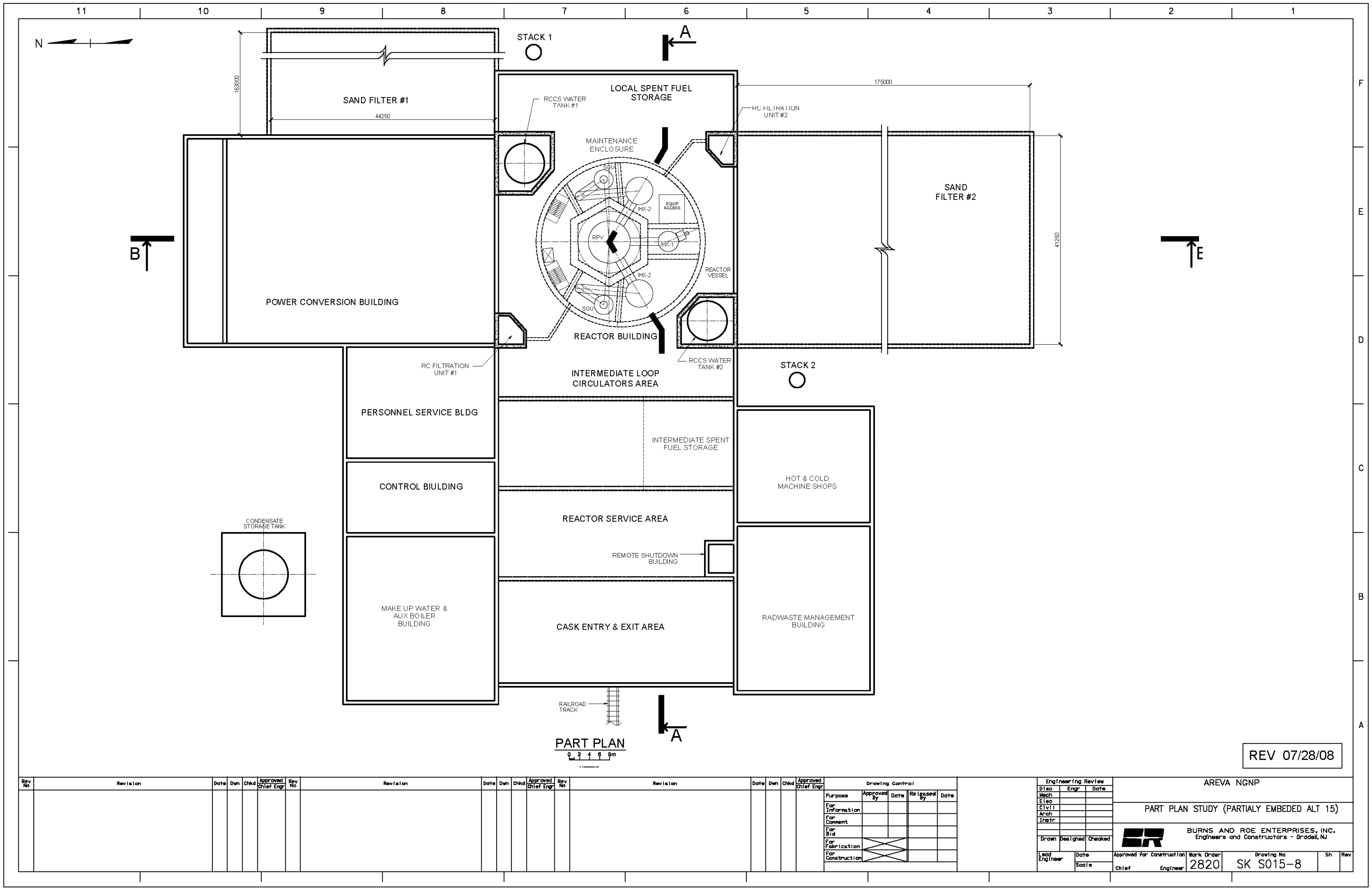


Figure 6-11: NGNP Partially Embedded Reactor Building – Section A-A

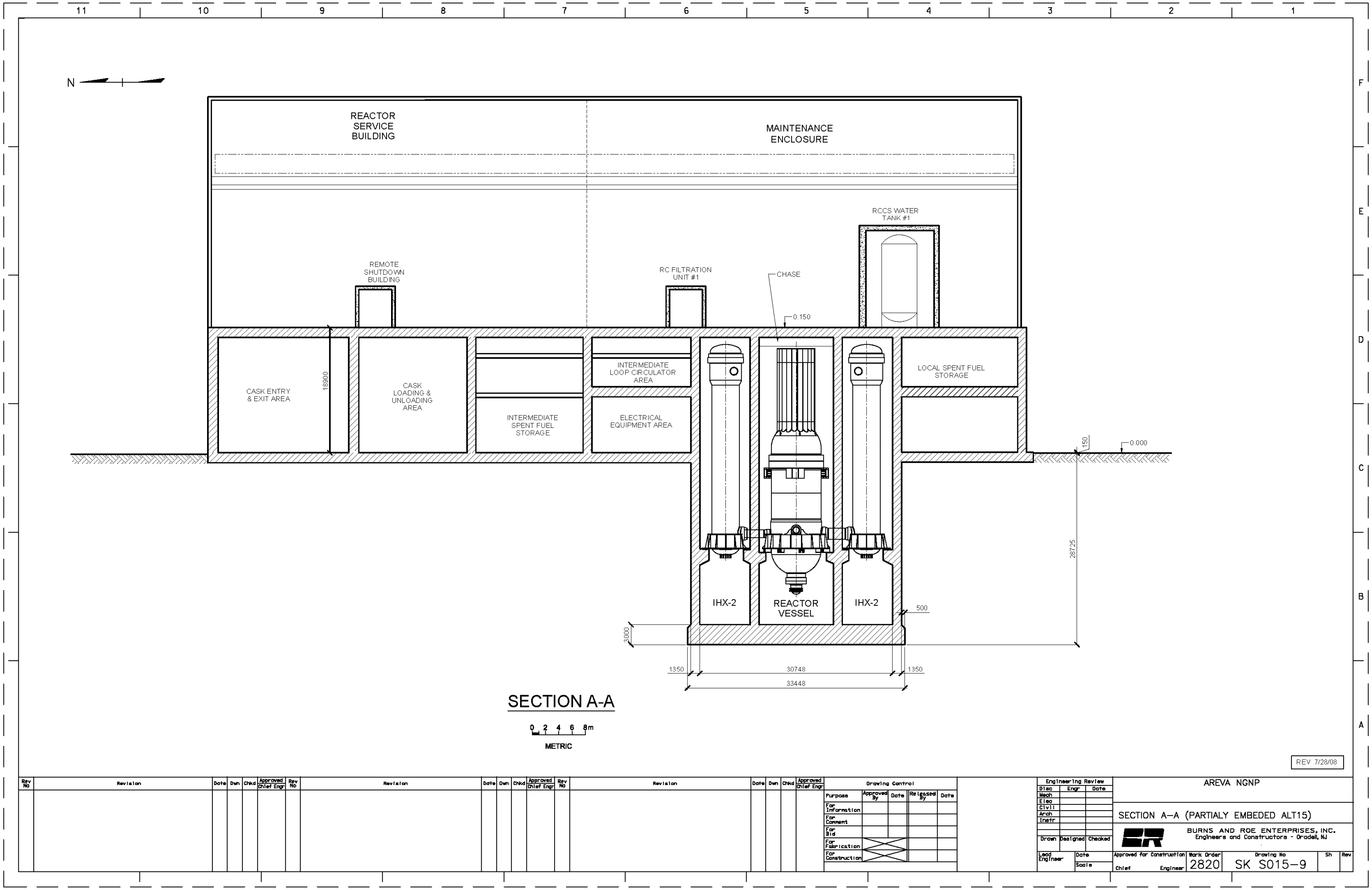
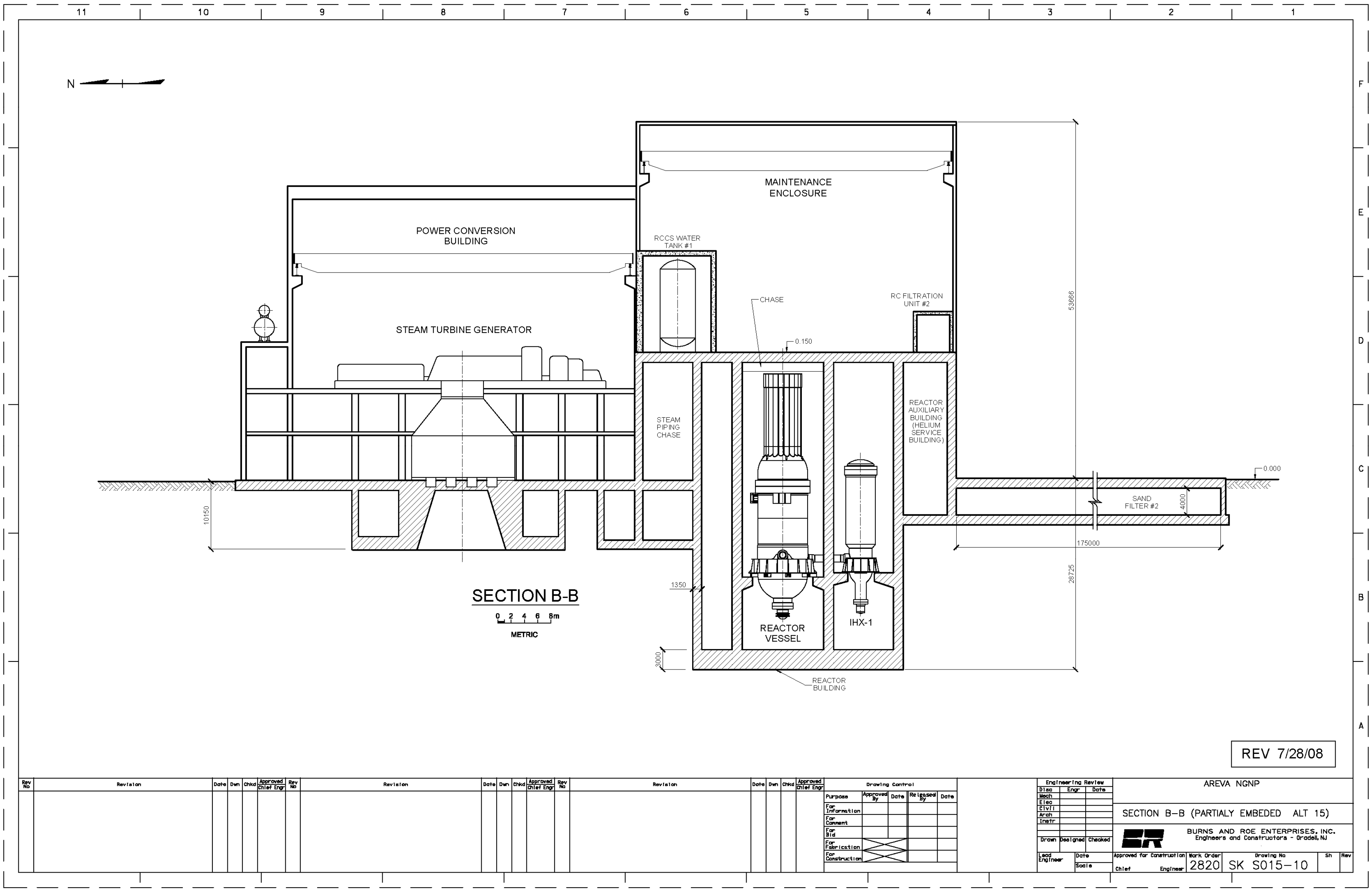


Figure 6-12: NGNP Partially Embedded Reactor Building Section B-B



6.5 Passive Filter System

This study also evaluated the design of a passive filter system that filters the prompt release following the primary system depressurization accident. The passive filter system requires a large area for the sand filter media. The use of such filter system is not expected or recommended by AREVA.

The passive filter system is normally in standby. Upon a large primary system depressurization accident the reactor building is rapidly pressurized causing the blow-out panels or the rupture discs to activate channeling the pressurized building atmosphere flow into the filter system before it is discharged into the atmosphere. The composite of the reactor building atmosphere at this stage includes pre depressurization air (or an inert gas – if chosen), primary circulating helium, dust particles dislodged due to depressurization, and a variety of radionuclides species that form the prompt source term. The reactor building atmosphere is channeled through a system of passive filter system driven by the internal building pressure and released from the exhaust stack until the reactor building is fully depressurized.

The most appropriate high temperature passive filter system for this application is a sand filter system with multiple layers coarse and fine sand trapping suspended radionuclides including any dust particle before it is released to the atmosphere. Figure 6-2 shows an artist rendition of such filter system whereas Figure 6-1 provides a flow diagram. The filter media will potentially be exposed to helium with temperature significantly below the primary gas temperature as a result of expansion to nearly atmospheric pressure.

The need for filtering the reactor building atmosphere following the initial depressurization is governed by the prompt source term discussed in Appendix A. If it can be shown that such release dose not violate site boundary dose limits under the limiting accident scenarios, such filter system would not be required. Therefore, the requirements for high temperature passive filters directly linked to the prompt source term and fuel quality and performance characteristics.

In case passive filters are not required the same flow chase would be used to depressurize the primary system and the reactor building and discharging directly into the atmosphere.

Once the initial depressurization of the reactor building concluded and the accident progresses, the sand filter system is isolated and further reactor building atmospheric releases are through the HEPA filter system.

The building initial depressurization channels and the sand filter bed area must be designed for the maximum expected pressure transient and may not be occupied when the reactor systems are pressurized. The initial estimate of such filter system leads to a large filter area as depicted in the plan view layout, Figure 6-3.

6.5.1 Piping Layout for the Passive Filter

The passive filtration system serves both the reactor cavity and the areas housing the IHXs and the steam generators in the Reactor Building. Each vent pipe measures 1 m in diameter and serves one of the two filters. An attempt was made to route the piping to the sand filters keeping into consideration the following:

- Protection against neutron streaming through the vent piping by adequate routing and shielding
- Non-interference with component access and removal paths
- Minimal interference with RCCS panels in the reactor cavity
- Provide access for maintenance of valves in the vent path

The schematic routing of vent piping for the embedded Reactor Building is shown on Figures 6-13 and 6-14. This routing exhibits ducts located in each corner of the hexagonal reactor cavity. The RCCS panels would be located on the entire inside surface of the reactor cavity and over the six ducts in the corners. Each triangular duct is provided with a rupture panel at the lowest end immediately above the reactor ledge. Reliability of rupture panel rupture under certain pressure differentials is not very high. Therefore multiple partial capacity rupture panels need to be provided. These multiple collection points lead to a common ring header at the top from where connections to the redundant filters are made.

Interference with RCCS can be reduced. Interference with reactor support at the ledge would need to be examined once the support system is defined. Four vent pipes (two per filter), 1m in diameter, may be needed. Four large vent pipes penetrating the reactor cavity wall would result in neutron streaming into the adjoining cells and should be minimized by concrete shielding. Because of the large height allocated to the control rods, the neutron flux at the top of the reactor cavity will be lower thus aiding shielding concerns. Additionally, routing of the vent pipes (towards the sand filters) need to be 2-3 m below grade level. This routing neatly fits the structural layouts of the embedded reactor building concept.

Note - AREVA is not emphasizing passive filter systems but simply making a statement that we realize the issues involved and have demonstrated one possible way at this pre-conceptual design stage.

Figure 6-13: Plan View of Conceptual Routing of the Vented Pipe to Sand Filter

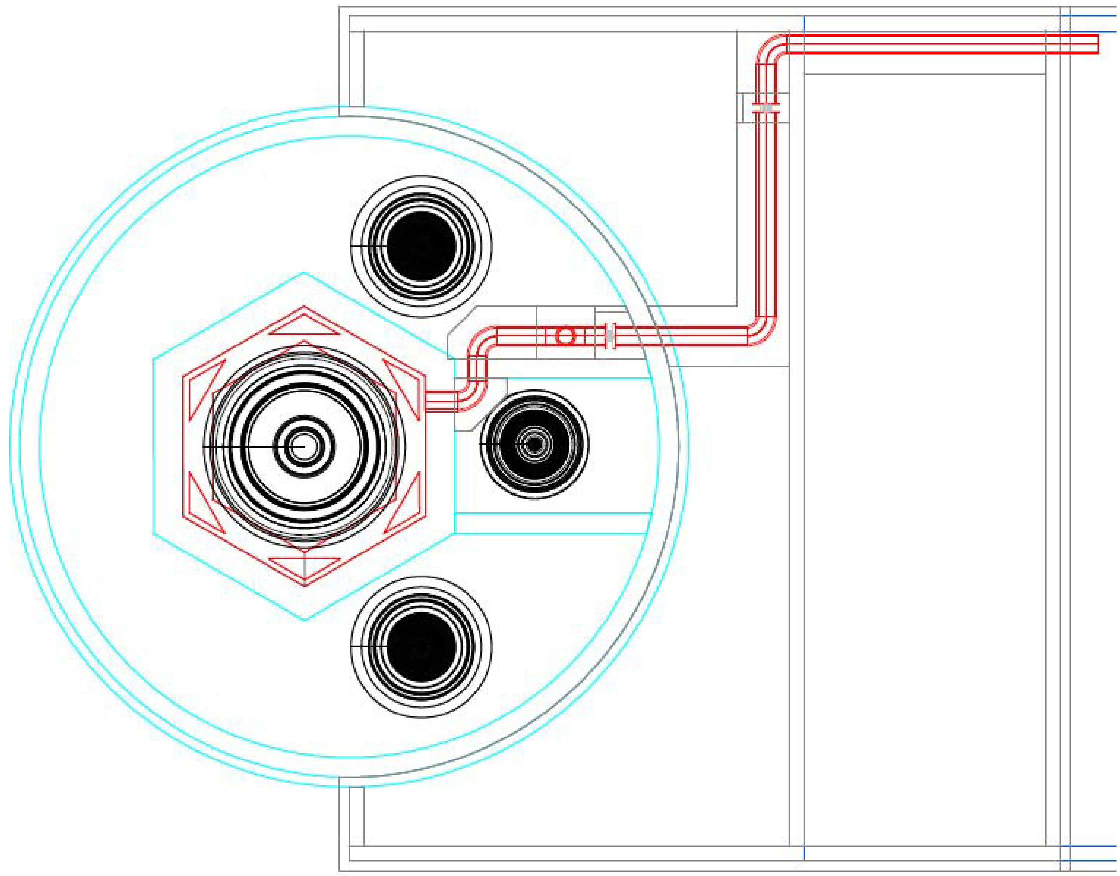
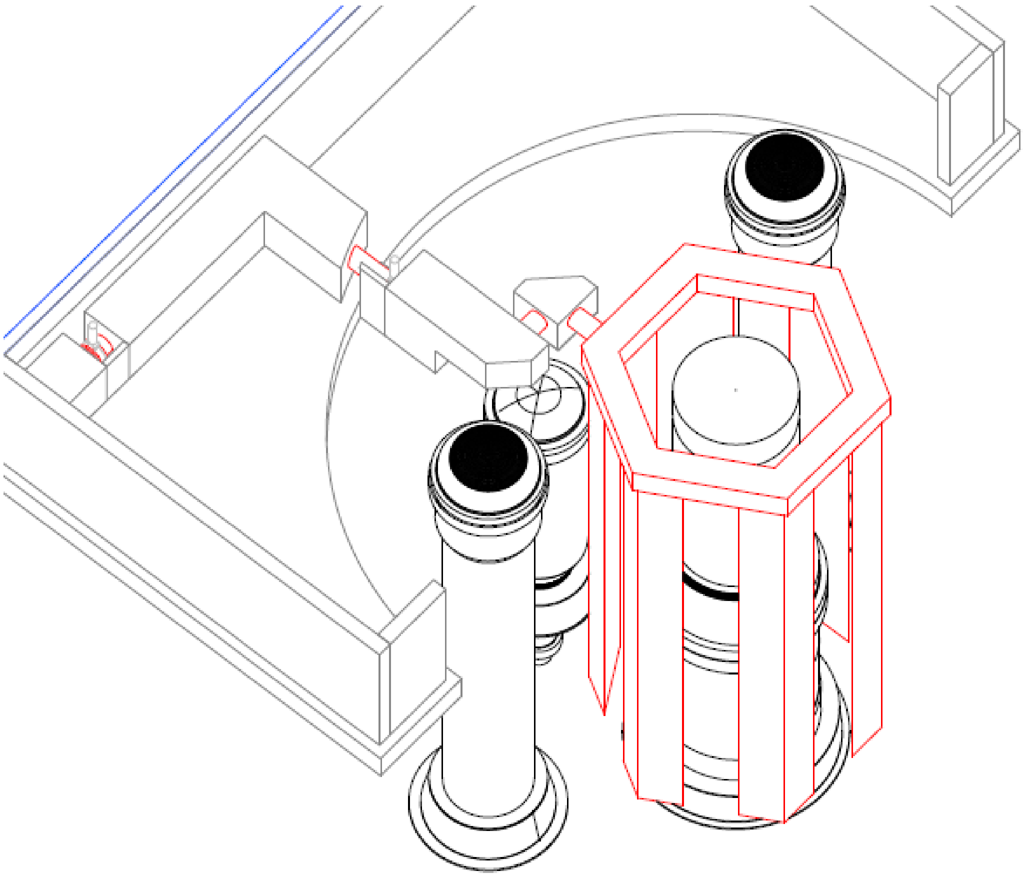


Figure 6-14: 3-D View of Conceptual Routing of the Vent Pipe to Sand Filter



6.6 Active Filter System

Active filters are used to remove radionuclides in reactor building caused by the delayed source term before release into the atmosphere. Following a primary system depressurization accident and subsequent pressurization and depressurization of the reactor building, the reactor building would now be in a depressurized status (slightly negative pressure) and sealed. Any further release to the reactor building atmosphere will be filtered and monitored through the HEPA filter system before release to the outside environment - Figure 6-1. During this phase of accident progression the core will continue to heatup as the core decay and sensible heat is removed by the shutdown cooling system (SCS) or the reactor cavity cooling system (RCCS). As the core heats up and cooldown progresses additional fuel failure may occur. This additional release forms the delayed source term. The aerosol part of the delayed source term could find its way in to the reactor building, once there it will be filtered, monitored and would be released to the outside environment while maintaining site boundary dose limits. Without a driving force the balance of radionuclides are immobile and remain in the core.

6.7 Reactor Building Design Implications on Commercialization

Design and licensing of the HTR reactor building evokes strong technical and emotional responses from technical community, academic community, national regulators and the general public. The main challenge for the HTRs reactor technology is that the safety design philosophy is fundamentally different from those used in the current conventional light water reactor technology. Those differences that impact the radionuclides containment function and the reactor building design are discussed in this study.

From the regulator and the public perspective, the natural tendency is to expect similar safety design features as used in the current nuclear reactors; the fact that the technology is different is somewhat lost in the initial discussions. It is therefore incumbent upon the proponents of this technology to properly introduce the regulators and the general public to this technology to make sure the technology differences do not lead to regulatory disadvantages that would make the design less safe and potentially commercially unattractive.

This study presumes the use of a vented reactor building with active filter option for the delayed source term.

Three reactor building design were evaluated in this study. The HTR reactor building choices are primarily depend on the fuel performance and site characteristics:

> *Fuel Performance* –
> The fuel performance requirements and the reactor operating characteristics result in a source prompt source term level that may require addition of a prompt source term filter system. An initial conceptual design for a high temperature sand filter was evaluated. The preliminary results indicate that site footprint will increase by a multiple of 4 to 5 due to a large cross sectional area necessary for the sand filter media. This in turn will increase cost of the plant and reduce the esthetic appeal of the small modular design. However, in a multi-module commercial application the redundant sand filters may be share between the modules.
>
> *Site Characteristics* –
> The specific site geological characteristic limits the reactor building design choices. A site with high (close to surface) water table limits the building options available to partially or fully above ground. Such building design has greater challenge to external hazards including airplane crash and internal design challenge of refueling difficulties. The cost impact of the fully embedded option is large and particularly so for a coastal or a high water table site.

For commercialization it is vital that the HTR specific reactor building design be demonstrated from a technical and licensing perspective by the NGNP. Such demonstration must include a lack of need for sand or some other high temperature prompt source term filter system.

7.0 SUMMARY OF THE STUDY

This study identified and assessed the technical, safety, and licensing issues associated with the NGNP reactor building and radionuclides containment function. It also evaluated the merits of reactor building embedment depth on the plant construction. Issues such as cost, security and external hazards were evaluated. Building design features such as filtrating systems were considered.

The objective of this study was to define the preliminary technical and functional requirements (T&FRs) that are listed in Section 5.0. An ancillary objective of this study was to identify any open issues and recommendations for further design studies and R&D needs to address these issues.

A summary comparison of fission product filtration and release options is provided Table 7-1.

Table 7-1 – Fission Product Filter and Release Options

	ANTARES (Preferred)	ANTARES (Backup)	LWR	CANDU	Sand & HEPA Filter Option	HEPA (only) Filter Option	MHTGR	PBMR (US)
High Pressure Building	N	N	Y	Y	N	N	N	N
Filter Initial Release	N	N	N/A	N/A	Y	N	N	N
Filter Delayed Release	N	Y	N/A	N/A	Y	Y	N	?
Delayed Vent and Filter	NA	NA	Y	Y	NA	NA	NA	?

7.1 Open Issues

This section provides a list of technical, safety and licensing issues that must be addressed and resolved to complete the definition of the T&F requirements.

Technical

T1. Hydrogen generation process and capacity

The hydrogen generation process to be employed for the NGNP would be determined by the temperatures of the primary coolant. The hydrogen generation process, capacity, storage (if any) and transport logistics constitute hazards to the co-located power generation unit.

T2. Reactor fabrication and transportation

The size of the reactor vessel (over 8 m in diameter and 24 m long) imposes certain limitations such as large forgings and fabrication of large sections of the reactor for which currently there are no qualified foundries and fabrication shops in the USA. Transport of the large reactor vessel sections for final fabrication (and inspections) to complete the reactor assembly at the site shop poses additional challenges.

- Evaluate on-site reactor vessel fabrication and installation options

T3. Reactor vessel and concrete temperature under accident conditions

The ASME and the ACI codes specify (for the reactor vessel and the concrete respectively) permissible temperature limits and allowable duration under those conditions, following off-normal/accident plant conditions.

- Develop requirements for the reactor cavity concrete, stability and strength
- Determine maximum time at temperature dependencies and requirements
- Evaluate reactor vessel, cross vessel, steam generator and IHX support system design options

T4. Spent fuel storage/transportation/processing

Spent fuel storage, transportation and processing was not in the scope of this study. The near reactor storage of new and used fuel will impact the design of the reactor building.

- Establish requirements and evaluate design options for the near reactor spent and new fuel storage options and its impact on the reactor building design.

T5. Refueling machine design

A detailed comparison and selection of the NGNP refueling strategy and equipment should be performed. This study would use time and motion studies to evaluate different refueling equipment designs, including the “industry default” cask system and AREVA’s fuel server system, as well as various reflector element management scenarios.

- Evaluate refueling processes

- Establish design and interface requirements for the refueling machine

T6. Control Rod Drive System

The design of the control rods and their drive mechanisms influence the height above the reactor vessel head requirements.

- Evaluate CRD design options and interfaces with the reactor building and reactor vessel

T7. Reactor Building pressure, temperature, and Source Term

Post depressurization accident building pressure, temperature and radioactive source must be estimated.

- Develop methodology for calculating building pressure, temperature and source term.

Safety

S1. Containment characteristics

Containment pressure retention and leakage characteristics are governed by the material at risk, i.e. the internal energy and the source term associated with the primary coolant. The source term is determined by the fuel characteristics.

- Develop NGNP event specific source term calculation methods
- Fully characterize radionuclides containment function for the HTRs and NGNP

S2. Hydrogen storage quantities/distances and transportation

The hydrogen generation capacity, storage and transport logistics constitute ‘external’ hazards to the co-located power generation unit. The frequency of transport and the sizes/quantities of the hydrogen containers or transport piping systems must be considered in the hazard analysis.

S3. Effectiveness of sand filters under accident conditions

The sand filters may be considered to reduce the radiological activity of prompt source term before release to the environment during the initial phase of a depressurization event.

- Evaluate sand filter designs and their effectiveness to remove graphite dust and other particulates.

S4. Fuel particle as individual containments/prismatic fuel design

Coated particle fuel is expected to play a major role in the radionuclides containment function of the NGNP reactor technology.

- Develop prompt and delayed source term concept and calculating methodology
- Review existing methodologies and assess capabilities for calculating fuel particle source term release, and radionuclides transport through fuel, reactor pressure vessel and reactor building
- Examine plate-out and lift-off phenomenon and identify required release correlations

S5. Quality of fuel/TRISO coating under off-normal/accident conditions

- Develop HTR safety case and examine the correlation between fuel particle coating quality and performance requirements and the HTR safety
- Develop HTR fuel acquisition and qualification program

S6. Source term of the primary coolant including radionuclide absorption on graphite dust

- Review existing methodologies and assess capabilities for calculating primary coolant source term and the associated radionuclides absorption in graphite, dust generation and transport.

S7. Steam leakage (conventional steam cycle) and graphite / graphite dust interactions

- Review existing methodologies and assess capabilities for calculating water and/or air ingress

S8. Plate-out and lift-off of radionuclides (under depressurization events)

- Review existing methodologies and assess capabilities for calculating or predicting plate-out and lift-off phenomena

S9. Effectiveness of the RCCS under low emissivity (of reactor vessel/RCCS panel) conditions

- Evaluate and effectiveness of RCCS and characterize passive heat transfer performance requirements

S10. Building Construction

- Assess modular construction techniques and its applicability to NGNP and the commercial plant construction

- Develop design guidelines relevant systems and site conditions

Licensing

A detailed NGNP licensing strategy topical report should be prepared for submittal and review by the NRC. This report should provide a roadmap for both licensing of the NGNP prototype plant and design certification of one or more subsequent commercial plant designs, including the strong interrelationships between these two activities. The need for supporting technology must be addressed including the required timing of this data. Direct vendor involvement in this activity is essential if the resulting plan is to support the projects commercialization objective.

L1. NRC regulations governing HTGRs

- Develop NGNP licensing strategy and begin preapplication interactions with the NRC

- Develop technology familiarization plan

- Identify required topical reports

- Identify required codes and standards

- Identify policy and regulatory needs

L2. Define approach to criticality and full power operation process for NGNP

- Develop a stepwise licensing process where HTR safety concepts will be tested during NGNP commissioning

L3. Definition of applicable design basis and severe (beyond design basis) accidents

- Develop licensing basis events and select design basis accidents and identify applicable beyond design basis accidents

L4. Develop mechanistic/deterministic licensing basis events selection criteria

- Develop the NGNP risk informed methodology for licensing basis event selection and safety classification criteria of SSCs

L5. Definition of required physical security including design basis threats

Unique design features of the NGNP, such as co-located hydrogen production and storage facility, will affect the physical security program. The design basis threat is part of the physical and strategic security; it would be defined by the NRC specifically for the NGNP project

7.2 Additional R&D and Studies

The section provides the R&D and engineering studies needed to resolve the issues identified in 7.1

R 1. Fuel particle as individual containments

Particle fuel manufacturing quality, operating performance characteristics and understanding of fission product release mechanisms during normal and accident conditions is central establishing the HTR radionuclides containment requirements.

- Interact and interface with going fuel R&D and Irradiation
    1. Metallic fission product retention characteristics
    2. Particle fuel manufacturing quality controls
    3. Heavy metal contamination control
- Develop and benchmark computer codes and methodologies for radionuclides generation, release and transport.
- Develop and benchmark computer codes for core thermo-fluidic performance characterization, i.e. core flow and temperature prediction.

- Incore temperature instrumentation needs
- Develop core radionuclides plate-out and lift-off models

R 2. Vessel fabrication techniques using high temperature materials and the impact on the reactor building design

- Specifications of ASME qualified materials for high temperature duty need development followed by development of fabrication techniques and inspections.
- Availability of global foundries and fabrication shops for fabricating reactor vessel segments using high temperature materials that could lead to final fabrication of the reactor vessel in a site shop needs development.

R 3. Water and Air ingress computer modeling and benchmark experiments

- Examine core graphite oxidation characteristics
- Volatile gas generation models

R 4. Instrumentation

- Incore temperature instrumentation

8.0 REFERENCES

1. NGNP Fuel Design Special Study, AREVA Document No. 12-9077148-001,

2. Fleming, Karl N. and Fred A. Silady, “A Risk-Informed Framework for Defense-in-Depth for Advanced and Existing Reactors,”, Reliability Engineering and System Safety, Elsevier Publishing Company, 78 (2002) pp. 205–225

APPENDIX A: RADIONUCLIDES CONTAINMENT FUNCTION

A.1 Introduction

The overall objective of the HTR and the NGNP is the production of safe economic power and process heat for hydrogen production that meets regulatory and user/utility requirements. This design objective is broken down to four supporting goals, as follows:

Goal 1. Produce Electric Power and/or Process Heat
Goal 2. Maintain Plant Protection
Goal 3. Maintain Control of Radiation and Radionuclide Release
Goal 4. Provide for Emergency Preparedness

These goals must be supported during each of the four states of plant operation: 1) energy production, 2) shutdown, 3) refueling, and 4) transition states (i.e. starting up/shutting down).

For convenience, the four goals will be discussed separately in the order listed above, but the design process in achieving these goals is integrated and iterative so that, for example, the choice of the fuel, moderator, and coolant for maintaining plant operation significantly influences maintaining plant protection and control of radionuclide release. The first two goals are primarily of an economic nature, whereas the second two are driven by safety concerns. As design decisions impact on all four goals, it is necessary to address all of them in explaining the safety design philosophy for the plant.

The safety design philosophy for any nuclear power plant must address the selection of the inherent features of the reactor, i.e. the choice of a fuel, moderator, and coolant and the basic materials and design characteristics of these, the design of the barriers to contain the inventory of radioactive material, and the safety functions that are provided to maintain and protect these barriers. In addition to the inherent reactor features, the safety design philosophy includes the design of systems, structures, and components (SSCs) that provide the safety functions and the approaches that are taken to support the capability and reliability of these SSCs. These approaches include the use of passive SSCs, and appropriate application of redundancy and diversity of any active SSCs.

Modular HTGRs have benefited from the experience of the prototype, testing, and demonstration HTGRs designed, built, and operated to date with their ceramic-coated fuel particles, graphite moderator, and inert helium coolant, which comprise the inherent characteristics of all HTGRs. Further, modular HTGRs have optimized those fundamental core features by specifying and configuring them to emphasize public safety to the extent that the public's normal day-to-day activities are not disturbed over a wide spectrum of normal and off-normal events. In essence, stringent requirements for the third goal are set to minimize the design requirements from the fourth goal. To accomplish this with high assurance, the design of the modular HTRs and the NGNP have been guided by a safety design philosophy with emphasis on radionuclide retention at the source within the fuel particles with minimal reliance on active design features or operator actions.

This philosophy has had a profound impact on the design of the HTRs in two important ways. First, the philosophy requires control of radionuclide releases primarily by retention within the coated fuel particles and with decreasing reliance on secondary barriers (such as the helium pressure boundary or the reactor building). This leads to important design selections: the type of fuel, the specification of its as-manufactured quality, and its required in-service performance. Proof of radionuclide retention is dramatically simplified if arguments can center on issues associated with fuel particle integrity alone.

Second, the philosophy requires that control of radionuclides be accomplished with judicious reliance on passive systems and minimal reliance on prompt operator actions. This leads to fundamental design selections such as the core size and geometry, the power density and vessel type. By minimizing the need to rely on active systems or operator actions, the safety case centers on the behavior of the laws of physics and on the integrity of passive design features. Arguments need not center on an assessment of the reliability of active pumps, valves, and their associated services or on the probability of an operator taking various actions, given the associated uncertainties involved in such assessments. However, although not required, an explicit objective of the safety design approach is to provide long response times for operator actions.

An important element of the safety design philosophy of the HTR is to demonstrate that the principles of defense-in-depth have been effectively applied. These include implementation of barrier defense-in-depth, process defense-in-depth, and scenario defense-in-depth principles (Reference 2). The NGNP uses inherent and passive capabilities to prevent the release of radioactive material from the fuel during both design basis and beyond design basis accident conditions. Because large releases are precluded by the inherent and passive features of the reactor, there is a greater balance between prevention and mitigation across the full range of design basis and beyond design basis accidents.

The overall intent is to provide a simple safety case that provides high confidence that the safety requirements are met.

A.2 Radionuclides Retention

The safety design for radionuclide retention follows the overall process discussed above. Design selections for the four plant states of energy production, shutdown, refueling, and starting up / shutting down lead to the need for multiple barriers for radionuclide retention:

- Ceramic coated fuel particles with high temperature capability dispersed in fuel compacts within the prismatic graphite fuel elements
- A compact helium pressure boundary that completely surrounds the fuel barrier
- A below-grade reactor building barrier that completely surrounds the helium pressure boundary barrier.

The selection of multiple, concentric barriers is then reviewed relative to the requirements of the second goal to maintain plant protection. Oftentimes, the design selections for the first normal operation goal are sufficient to meet the second goal's requirements, for example, the choice of reactor materials that have high temperature capability and that are chemically compatible avoids plant protection concerns related to reactor internal core damage or the production of high pressure gases with the potential to transport radionuclides from the fuel and helium pressure boundary barriers. In other cases, the plant protection requirements lead to new design selections, for example, an independent means of forced core cooling to allow maintenance of the main heat transport system. A third case is when the plant protection requirements lead to modification or adjustment of the design selections chosen for normal operation, for example, the high reliability of the helium pressure boundary barrier so that any accidental releases of primary circuit activity are low and worker exposures and cleanup are minimized.

The third goal of maintaining control of radiation and radionuclide retention during normal operation and during a comprehensive set of accident conditions is at the center of the NGNP and HTR safety design. The top requirement for Goal 3 is to not disturb the normal day-to-day activities of the public. To accomplish this, there must be high assurance that over a wide spectrum of events the offsite doses are not only within regulatory limits but are within the more limiting levels that trigger the need for the public to be sheltered or evacuated. The spectrum of events must include a full range of potential accidental releases that involve: 1) the normal operation circulating and plate-out activity in the helium and/or 2) release from fuel particles that do not have intact coatings as a result of the fuel manufacturing process, as a result of normal operation service, and as a result of the event itself.

With the above emphasis on radionuclide retention, the emergency preparedness provisions of the fourth goal are limited to onsite actions and offsite notifications.

The design features for meeting the safety requirements are selected in parallel with and within the constraints imposed by other performance and economic requirements. Retention of radionuclides at the source within the fuel assists in meeting not only public safety but personnel safety requirements, in greater access and reduced maintenance times, and potentially to a more straightforward and transparent licensing process. In addition, retention within the fuel is also supportive of the investment protection goal by minimizing the risk of costly radioactive material clean-up.

With this background on the context of the selected barriers within the design process, the following bases for the safety design are made relative to the radionuclide retention function.

A.3 Bases for Safety Design of Fuel Barrier

The fuel compacts shall be manufactured to high quality standards, that is, low levels of defective coated fuel particles and heavy metal contamination.

Fuel performance during normal operation shall not lead to unacceptable failure of intact particles and/or unacceptable radionuclide release from as-manufactured defective fuel particles.

Circulating and plate-out activity within the helium pressure boundary during normal operation shall be acceptably low.

Fuel performance during off normal events and accidents shall not lead to unacceptable failure of intact particles and/or unacceptable radionuclide release from in-service and as-manufactured particle failures.

A.4 Bases for Safety Design of Helium Pressure Boundary Barrier

The helium pressure boundary shall be manufactured to high quality standards.

The helium pressure boundary shall encompass the fuel barrier during power operation.

The vessels and associated piping comprising the helium pressure boundary shall be equipped with safety/relief system to protect against over-pressurization.

Interfacing fluid systems shall be at pressures lower than the helium pressure boundary relief set point to preclude over-pressurization from external sources.

A single phase coolant gas shall be utilized to preclude the potential for internal over-pressurization of the helium pressure boundary.

Compatible fuel, core, moderator, and coolant materials shall be utilized to preclude the potential for chemical reactions leading to internal over-pressurization of the helium pressure boundary.

Accidental leaks or breaks in the helium pressure boundary shall be limited in frequency and size to acceptable levels.

During accidents the helium pressure boundary shall provide radionuclide retention, additional to that provided by the fuel, by means of natural passive mechanisms including plate-out, deposition, and radioactivity decay.

A.5 Bases for Safety Design of Reactor Building Barrier

The reactor building shall encompass as much as practical the helium pressure boundary.

In the event of leaks or breaks in the helium pressure boundary, the reactor building shall have vented dampers to allow helium blowdown to minimize subsequent radionuclide transport mechanisms.

During accidents the reactor building shall provide radionuclide retention, additional to that provided by the fuel and helium pressure boundary barriers, by means of natural passive mechanisms including deposition, agglomeration, and radioactivity decay.

A.6 Validation of Safety Design Bases for Radionuclide Retention

In general the means to demonstrate the safety design bases include:

- Reactor experience
- Analyses
- Component tests
- Separate effects tests
- Critical test facility
- Simulator or mock-up tests
- Irradiation Testing
- Manufacturing quality control testing
- Prototype plant confirmation testing
- Extended prototype operation
- Commercial plant start-up testing
- Commercial plant operation monitoring

One or more of these means may be sufficient depending on the basis for the safety design. In addition, prototype plant confirmation testing and/or commercial plant monitoring may be employed to provide greater levels of assurance. In general, because of its importance to the ANTARES safety case, the bases for the safety design of the fuel barrier will employ practically all of the above means. In contrast, the degree of radionuclide retention required in the other barriers will lead to less extensive validation and will rely more on reactor experience.

The next four sections discuss the required sub-functions to achieve radionuclide retention in the fuel. Once again the design process is integrated and iterative so the order of discussion is selected for ease of discussion.

APPENDIX B: SECURITY /DBT CONSIDERATIONS FOR NGNP

B.1 Introduction

For the purposes of this work, it was assumed that the NGNP would employ typical layouts, construction methods, designs and security programs – similar to commercial U.S. nuclear power plants - with the provision that considerations should be given to constructing risk-significant structures below grade.

U.S. commercial nuclear power plants are heavily fortified with well-trained and armed guards. Immediately after the 9/11 terrorist attacks, the NRC advised nuclear facilities to go to the highest level of security. After that, a series of orders were issued to further strengthen security. They have layered physical security measures, such as access controls, water barriers, intrusion detection and strategically placed guard towers. Together, these make up the plants' response to the Design Basis Threat (DBT). The DBT is developed from real-world intelligence information and describes the adversary force – coming from both ground and water – the plants must defend against. DBT specifics are classified as SGI (Safeguards information) withheld from public disclosure in order to protect sensitive information that could aid terrorists. The NRC regularly reviews the DBT and adds new requirements when necessary. In addition, there are two Category I Fuel-Cycle Facilities in the U.S. that make reactor fuel for nuclear plants. Since these plants handle nuclear material that could be targeted by adversaries, they also must defend against a DBT similar to that for nuclear power plants. Additional security requirements apply to the transportation of radioactive material including new and spent fuel.

B.2 Physical Security Program

Common physical security components of a security program include:

- Personnel Access Control System
- Intrusion Detection System and/or Fences
- Guard Force and Patrols
- Security Plan
- Guard Towers

Considerations on these components for a NGNP are discussed below.

Personnel access control systems – A NGNP would employ a system essentially that same as that in at commercial Nuclear Power Plants (NPPs). The NGNP site would be divided into three (or more) concentric areas (e.g., owner-controlled, protected and vital) with the most-restricted areas in the center. (See Figure B-1). The physical arrangement of the facility will govern the number and location of access points into and out-of these areas. Little or no controls may be required for administration building of no safety or strategic significance.

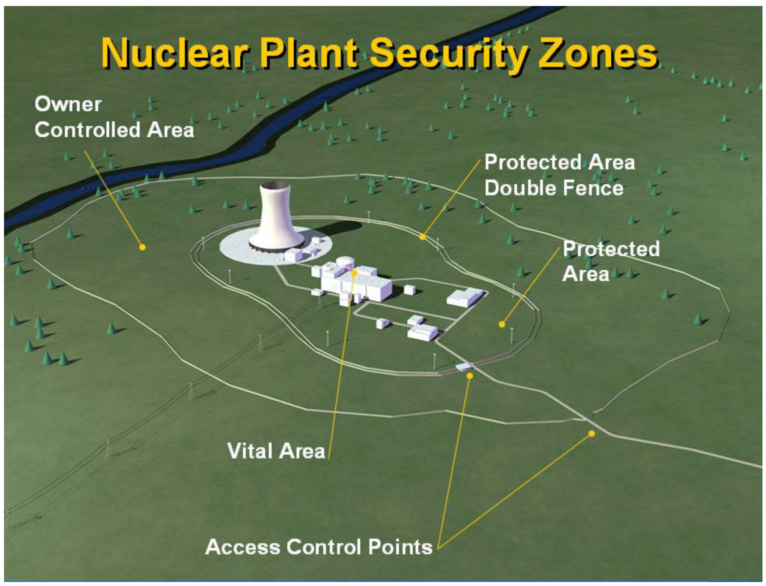


Figure B-1 (Graphic NEI)

Intrusion Detection Systems and/or Fences – A larger protected area would increase the costs associated with intrusions systems, fences, CCTV and routine patrols. Depending on site terrain and features, more guard towers may be required. However, the costs associated with hardening critical, safety-related structures against the effects of a vehicle bomb would decrease as the size of the protected area increases. Studies have shown that the cost differences are only a small fraction of the total costs.

Guard Force – The size of the guard force is directly influenced by the size, terrain and arrangement of the site. A larger site would require a larger force.

Security Plan – To the extent that this plan only reflects the physical protection features, training and qualification requirements and administrative controls already in place, the security plan is not expected to be significantly different at a NGNP than at a current-day commercial NPP.

Guard Towers – The number, physical location, and height must be adequate to ensure that the guards have an unobstructed view of the field of fire. A large site would require more towers. Topographic features such as roads, rivers, drainage ditches, hills, etc. must be considered in the location and number of the towers.

B.3 Design Basis Threat

10 CFR 73.1 "prescribes requirements for the establishment and maintenance of a physical protection system which will have capabilities for the protection of special nuclear material at fixed sites and in transit and of plants in which special nuclear material is used." It requires facilities and licensees to address the following:

- Radiological Sabotage
- Internal Threat
- Land Vehicle Bomb
- Water Vehicle Bomb
- Cyber attack

(Note that aircraft hazards are not currently part of 10 CFR 73.1.) While the implementation of these rules may be different at an NGNP, a review of the rules themselves did not identify any significant impediments to their implementation.

B.4 Sub Terrain Containment

While it would seem obvious that a below-grade containment structure would be easier to defend, quantitative evidence to confirm this would require modeling and comparison of comparable above and below grade containment designs. This is beyond the scope of this work.

A sub terrain containment may exhibit reduced post-accident leakage, potentially reducing post-accident on and offsite radiological doses. However, a ground-level release (compared to an elevated stack release) would have a significant affect on the plume and on-site accident doses. These aspects warrant further study to quantify any potential benefits.

In addition, limited accessibility to outer containment surfaces may complicate in-service inspection, maintenance and repair of sub terrain containments.

B.5 Energy Policy Act of 2005

In this legislation, Congress outlined 12 factors that were considered in the new DBT rule. Among those factors were:

- An assessment of physical, cyber, biochemical, and other terrorist threats;
- The potential for attack on facilities by multiple coordinated teams of a large number of individuals and several insiders;
- The potential for suicide attacks;
- The potential for water-based and air-based threats;
- The potential use of explosive devices of considerable size and other modern weaponry;
- The potential for attacks by persons with a sophisticated knowledge of facility operations;
- The potential for possibly long-lived fires.

B.6 Current Regulations

10 CFR Part 73 “Physical Protection” details existing requirements for security at nuclear power plants in the U.S.

- 73.55 “Requirements for physical protection of licensed activities in nuclear power reactors against radiological sabotage”
- 73.56 “Personnel access authorization requirements for nuclear power plants”
- 73.71 “Reporting of safeguards events”
- Part 73 Appendix B “General criteria for security personnel”
- Part 73 Appendix C “Licensee safeguards contingency plans”
- Part 73 Appendix G “Reportable safeguards events”

Most recently, three new rulemakings provide additional security enhancements.

- One rule, issued by the NRC in March 2007 after extensive public comment, modifies and enhances the Design Basis Threat. The rule describes general adversary characteristics that nuclear power plants must defend against. All existing nuclear power plants and Category I Fuel Cycle Facilities – and any built in the future – must adhere to this rule. The new rule also reflects insights gained by the NRC since 9/11, the latest threat information and a strengthened cyber threat component, as suggested by Congress and the public. In all, the NRC received and considered over 900 public comments on the rule.
- A second rule, which was issued for public comment in 2006, proposes enhancements to the physical security at nuclear power plants. Among other things, the proposed rule addresses access controls, event reporting, security personnel training, safety and security activity coordination, contingency planning, cyber and radiological sabotage protection.
- A third rule, still in the early stages, will propose additional aircraft impact assessments for new power reactor designs.

B.7 Protecting Against Aircraft

Classified studies indicate that there is a low likelihood that an airplane attack on a nuclear power plant would affect public health and safety, thanks in part to the inherent robustness of the structures. If the NGNP is constructed with a containment structure similar to those used by U.S. nuclear plants, a combination of factors could be credited, including the fact they are robust structures of steel and concrete, and relatively small targets. Operating plants have already implemented steps to minimize damage and risk to the public in the event of any kind of large fire or explosion. Cooperation with other federal agencies (FAA, TSA, etc.) also reduces the risk of an aircraft attack.

Studies have shown that NPP containments constructed below grade could be designed to be less vulnerable to aircraft impacts. That study concluded that building costs “will be driven primarily by the operational and functional requirements of its configuration, not by the thickness or quantity of its structural elements.”

B.8 Force-on-Force Exercises

Commercial NPP licensees are required to routinely test the security at nuclear facilities with realistic exercises using a well-trained mock adversary force. Similar exercises could be required at NGNPs. These force-on-force exercises are designed to test a security force’s ability to defend against the DBT.

These exercises typically span several days. During the attack, the mock adversary force tries to reach and damage key safety systems. Every plant with a force-on-force exercises a minimum of every three years in addition to yearly exercises.

B.9 Information Security

Three types of security information include:

Classified Information
Safeguards Information (SGI)
Sensitive Unclassified Non-Safeguards Information (SUNSI)

These security information classifications would be applicable to a NGNP.

B.10 Other Aspects

Physical security programs and strategic security plans at commercial NPPs are highly integrated. Changes in one aspect of the plan can indirectly affect other elements – intentionally or otherwise. Because the principles of security are similar for most industrial faculties, we can expect similar degree of integration at NGNP. While the nuclear technology ultimately employed at the NGNP can be expected to influence plant security, other aspects will be at least as important.

Many aspects of the security program that can be affected by design or operational changes include: stand-off distance; number and locations of defensive positions; use of delay tactics; number and frequency of security patrols; number and location of close-circuit television camera; location of central alarm station; location and design of backup central alarm station; alarm response procedures; guard qualifications; guard training; state and federal weapons authorization; fire protection; response time of local law enforcement, access to airports and special equipment; emergency plans, resources of state and local governments, resources of local industrial facilities; availability of water resources for fire fighting; and vehicle barrier system capabilities.

B.11 Conclusions

In general, security requirements for a NGNP are expected to be similar to current requirements for current U. S. commercial nuclear plants. While some minimum characteristics are common to all facilities the specifics are largely dependent on the location and design details of the facilities. A higher level of protection may be required for the NGNP if the actual (or perceived) risks are greater than those existing at today's commercial facilities. Unique design features of the NGNP, such as locating facilities below grade, will affect almost all aspects of the security program.

In general, the size and layout of the NGNP facility will have a greater effect on the security requirements than the technology employed. However, if the NGNP presents unmitigated risks unique to the technology employed, additional security measures will be necessary, with attendant costs. Security requirements are expected to escalate in the near term.

APPENDIX C: APPLICATION OF UTILITY REQUIREMENTS DOCUMENT

C.1 Introduction

The purpose of this Appendix is to determine a subset of the EPRI Utility Requirements that could be considered for application to NGNP and HTRs.

The URD was developed to facilitate incorporation of the utilities' five decades long operation and maintenance experience in operating nuclear power plants in the US for application to the design of the new generation of the Light Water Reactors (LWRs) called Advanced Light Water Reactors (ALWRS). The Requirements Document applies to ALWR plants of both kinds- Boiling Water Reactors (BWRs) and Pressurized Water Reactors (PWRs).

Volume I of the document summarizes program policy statements and top-tier design requirements and covers all ALWRs. Volumes II and III of the Requirements Document contain the complete set (top-tier and detailed) design requirements for the Evolutionary and Passive ALWRs, respectively.

Though the URD is for the ALWRs certain requirements could be applied to the NGNP. We have scanned through all the top tier of requirements in Volumes I and II and drawn policy statements and requirements that in our opinion are relevant to NGNP. In this process we have eliminated the following categories of policy statements and requirements to limit the content:

- Federal and NRC Requirements
- Code-related requirements
- Requirements that represent good engineering practice

The result of the screening for the NGNP resulted in the following policies and requirements for the NGNP:

C.2 Policy Statements

<u>Safety Policy</u>

The Safety Policy describes the integrated approach to safety with three overlapping levels of safety protection:

1. The first level of protection and the cornerstone of safety is accident resistance. Accident resistance is designed in order to minimize the frequency and severity of initiating events which could challenge safety.

2. The second level of protection is provided by core damage prevention. Core damage prevention includes systems and features which provide high confidence that initiating events which do occur or will not progress to the point of core damage. It is to be noted that "core damage" as specified for LWRs do not apply to HTRs. Therefore, a set of HTR specific metrics for accident prevention and mitigation has to be defined.

3. Policy on accident mitigation is to establish a challenging, quantitative requirement on mitigation (whole body dose less than 25 rem at the site boundary [about 0.5 miles from the reactor] for accident sequences with cumulative frequency greater than 10^{-6} per year) and to provide conservative, rugged containment systems to meet this requirement and regulatory requirements. This 10^{-6}, 25 rem requirement provides considerable margin to the NRC safety goal. The PRA is also used to confirm that the 10^{-6}, 25 rem requirement is met.

<u>Standardization Policy</u>

Standardization policy recognizes the importance of standard designs and the historic problems associated with customized designs. Accordingly, the program developed design requirements intended to form the technical foundation which will lead the way to one or more standardized detailed designs. Key plant features are specified in sufficient detail in the Requirements Document to permit meaningful standardization.

<u>Proven Technology Policy</u>

Successful, proven technology be employed throughout the plant, including system and component designs, maintainability and operability features, and construction techniques. The intent is to utilize the large experience base from existing HTGRs and LWRs as appropriate, in order to minimize the risk to the plant owner, assure credibility and control of schedules and costs, and ensure that a power plant prototype is not required.

Policy on Plant Maintainability

The Plant Maintainability policy is to design from the outset to make the plants readily maintainable over their life. This includes providing standardization of components, designing equipment to minimize maintenance needs and designing to facilitate those maintenance needs to reduce occupational exposure. Such needs include activities to support inspection, test, repair, and replacement of equipment and systems over the plant life and assuring that adequate access, laydown space, tooling, and services are provided as part of the basic plant design.

Constructability Policy

The constructability policy is to achieve a substantially improved construction schedule compared to experience with existing nuclear power plants. Specifically, the Evolutionary plants are designed to be constructed in no more than 54 months from the start of structural concrete placement through completion of the full power warranty run.

Program Economic Goals

It is the policy of the Utility Requirements Document that power plants will have projected busbar costs that:

- are as low as practicable while conforming to the operational and safety policies
- provide a sufficient cost advantage over the competing baseload electricity generation technologies to offset the higher capital investment
- risk associated with nuclear plant deployment have quantified uncertainties (and therefore quantified cost risks) that provide a similar cost advantage over the competing baseload electricity generation technologies

C.3 Overall Requirements

Safety Design/ Investment Protection Requirements

The following requirements are over and above the licensing design basis requirements and emergency planning requirements that are covered by Federal and NRC regulations:

Simplicity

One of the main thrusts of the requirements is that simplicity of design leads to reliability of operation. This can be stated as:

- Use a minimum number of systems, valves, pumps, instruments, and other mechanical and electrical equipment, consistent with essential functional requirements
- Use standardized components to facilitate operations and maintenance
- Design equipment and arrangements which simplify and facilitate maintenance
- Provide protective logic and actuation systems which are simplified
- Provide system and component designs which minimize demands on the operator during normal operation as well as transient and emergency conditions (e.g., minimizing system realignments to accomplish safety functions, segregation of safety and non-safety functions unless otherwise justified);

Design Margin

- Provide sufficient margin to reduce the likelihood of exceeding limiting conditions of operation
- Provide significant margin between normal operating range and reactor trip set points.

Investment Protection

- There shall be no fuel damage for at least two hours after sustained loss of all feedwater with no operator action. Note that the stated requirement is for LWRs; however, for NGNP comparable fuel damage and the appropriate time duration would need to be defined.
- Reduce the risk from station blackout by providing a non-safety grade, alternate ac on-site power source.

Performance Design Requirements

These requirements are the utility-mandated plant capabilities that are related to the plant's size, life, maneuvering capability, event transient response, and event response times.

Plant Design Life

The plant shall be designed to operate for 60 years without necessity for an extended refurbishment outage. The plant shall be designed to permit expeditious component replacement for obsolescence and failure over a lifetime of 60 years.

Load Following

The plant shall be designed to permit the utilization of the load following capability during 90 percent of each fuel cycle throughout the entire design life of the plant.

Load Rejection[3]

The plant shall be capable of a generator load rejection from 100 percent or less of rated power, without reactor trip or turbine trip and without lifting the main steam safety valves, and be able to continue stable operation with minimal house electrical loads. Atmospheric release of steam shall not be permitted to achieve this capability.

Event Capabilities

The capability to operate at reduced power with a secured circulator shall be provided. Additionally, system capability to restart the secured circulator shall be provided.

Refueling Cycle

A capability of operating on a fuel cycle with a refueling interval of 18 to 24 months shall be provided.

High Level Waste Disposal

Continued plant operation shall not dependent on offsite shipments of spent fuel.

Structural

Internal Missiles

All plant equipment and structures shall be designed to minimize potential internal missile generation which might adversely interact with safety-related equipment and structures.

Systems and Equipment

[3] Turbine bypass for PWR and BWR plants is discussed in detail in EPRI URD Requirements, Volume III, Chapter 2, section

In the modeling of pipe support stiffness for the pipe stress analysis, either generic stiffness values provided by the vendor or a value corresponding to "rigid" support assumptions, shall be used. If support stiffness is modeled, it shall be used consistently in the analysis for all loading conditions. Pipe support deflection due to dynamic load alone shall be limited to less than 1/8-inch, or the support shall be proven to be rigid by a frequency calculation. Design of the support shall include a minimum assumed load criterion. If physical limitations are such that these requirements make a design impractical, analytical models of the piping support structures may be included in the piping analysis or calculated stiffness values may be used.

Nozzle Loads

The Plant Designer shall establish reasonable generic design nozzle (at least 50 percent of the ASME Code allowable of the attached piping) loads based upon the size and strength of attached piping. These values shall be achievable without requirements for excessive piping supports or restraints. This is aimed at reducing reanalysis of equipment design.

Dynamic Analysis

The Plant Designer shall perform a dynamic analysis of major equipment and supports when appropriate to produce a more balanced design. If this option is taken, the analysis shall include significant inertial and stiffness effects of the supports and their attachments to the primary structure. The basis of this requirement is that dynamic analysis of major pipe and equipment supports will result in more economical and realistic designs for major supports.

<u>Materials</u>

Codes and Standards

The materials used for the component shall meet the Code requirements as a minimum, but it should be recognized by the Plant Designer that actual service conditions of the NGNP may be more severe and necessitate more restrictive conditions than imposed by Code specifications.

Preference shall be given to the designs which do not push material limits and which make use of conventional materials applied well within the limits where successful experience has been obtained. In particular, the use of high strength bolts or fasteners shall be eliminated where practical and sufficiently robust designs utilized so that special high strength materials are not required.

Metallic Materials

Cobalt-base hardfacing alloys to provide wear resistance shall only be used in those applications where no proven alternative exists. Hardfacing alloys to provide wear resistance shall only be used in accordance with the principles of EPRI NP-6737, Cobalt Reduction Guidelines Revision 1. The plant designer shall identify to the plant owner all applications of cobalt-base alloys and provide technical justification for their selection.

The Plant Designer shall select materials for the reactor vessel support structures that are resistant to brittle fracture and experience a minimal shift in RTNDT as a result of neutron fluence for the support scheme utilized. The Plant Designer shall provide an analysis of the reactor vessel support structure to demonstrate its acceptability for the maximum design plant life and capacity factor and to identify any operational constraints to avoid the potential for brittle fracture. Sufficient access shall be provided to the reactor vessel support structure to enable monitoring of the material temperature and performing modifications or heat treatment of the structural materials, if required.

Non-Metallic Materials

For NGNP secondary side components where the non-metallic materials may contaminate the feedwater, or are in the form of substances applied to clean stainless steel or nickel base alloys used in secondary system components, the impurity limits are as specified below:

- Total chlorine plus fluorine 500 ppm max.
- Heavy metals (total Hg, As, Pb) 250 ppm max.
- Total sulfur 500 ppm max.

Thermal Insulation Materials

Metallic insulation (Type 304) or blanket insulation with metallic jackets shall be required for piping and components where in-service inspection or possible contamination make nonmetallic materials unsuitable. (Experience has shown, when selecting insulation, that it is important that it be compatible with the metal surface it contacts, can be easily removed and handled to minimize maintenance and personnel exposure, and is resistant to environment and deterioration with time).

Reliability and Availability

Station Blackout

The Plant Designer shall provide plant design features to assure that the frequency of station blackout exceeding 2 hours duration can be demonstrated by PRA methods to have an expected frequency of less than 10^{-3} events per reactor year.

Trip Frequency

The Plant Designer shall perform an evaluation of potential system failures that will result in reactor trips which shows the expected trip frequency to be less than one per year. Because human errors cause a large fraction of reactor trips, emphasis shall be placed on the human factors aspects of operation and surveillance testing.

Forced Outages

The plant shall be designed so that the frequency and duration for forced outages does not exceed 5 days/year.

Refueling Duration Capability

The Nth of kind HTR plant shall be designed so that the total duration of a no problem refueling outage shall be 17 days or less (breaker-to-breaker) assuming 24-hour productive days.

Planned Outages

The HTR plant shall be designed so that the refueling and regular maintenance shall be completed in an average of less than 25 days per year. An average of 25 days per year for refueling and plant maintenance corresponds to 50 days on a 24-month fuel cycle.

Major Outages

The HTR plant shall be designed so that the frequency and duration of major outages does not exceed 180 days/10 years.

<u>Construction and Constructability</u>

Milestone Summary Schedule

The construction duration shall not exceed 48 months from the initiation of first structural concrete to fuel load, and the total duration from start of construction (first concrete) to completion of the plant for warranty demonstration shall not exceed 54 months. Note that this requirement is for a large commercial passive or evolutionary LWR, a similar construction milestone has to be developed for the modular HTR plant design.

Modular Construction

The NGNP plant shall be designed to maximize the benefits which can be obtained through use of modular construction techniques. A list of module types which will be investigated for application in the plant shall be scoped at an early stage of the plant design. Module types shall be classified as to whether the module will be fabricated entirely in an off-site shop, fabricated in major elements off site with final assembly at an on-site shop or laydown area, or will be fabricated entirely on site in a module assembly area.

The design of modules shall attempt to maximize the use of standardized elements and components to simplify the work. The design shall consider access space provisions for installation and construction fit-up and for maintenance, operation, and component removal/replacement. The design shall be 100 percent complete without “holds” at the time the module is released for fabrication.

A modularization plan shall be developed for major packages which involve complex interfaces, such as permanent structures and multidiscipline components or material, to determine whether they would best be fabricated by an off-site supplier or subassembled for fabrication on site. In developing the plan, an evaluation shall be performed of the procurement and inspection process to assess the cost advantages and risks involved with assigning material purchase responsibilities to the supplier. Also the risks involved with delivery schedule for very large modules shall be compared to smaller modules insofar as the effect of later delivery on the flexibility of installation and possible restraint of adjacent work activities. Owner concurrence with the modularization plan is required.

Operability and Maintainability

This section specifies requirements that enhance the operability and maintainability of the plant, through incorporating the lessons learned from the operation and maintenance of LWRs used in the production of electricity.

Components

Different types and sizes of valves, pumps, other components and systems, and the different types of instrumentation, control, and power equipment used shall be minimized. However, each component must be selected to meet the required function for each specific application.

Components shall be designed with sufficient extra margin to allow for wear, normal corrosion and erosion experienced in plant service, etc., without affecting plant performance and incurring unnecessary maintenance for an operating period of 60 years minimum or the design life of the plant.

Instrumentation and Control

The instrumentation and control system shall be designed to permit in-place calibration and schemes shall be specifically designed to allow the required periodic testing to be done without placing the plant in an unacceptable one out of two or three trip logic. This shall also apply to turbine generator control logic. In-core instrumentation to accurately measure coolant temperature profile is highly desirable.

Control Room Design

Multi-unit stations shall have identical, but separate control rooms except those controls which are common to both units. Separate control rooms are specified by URD to assure independent control under all anticipated conditions. Use of a single control room for modular plant operation must demonstrate no common mode failure of independent controls under all normal, off-normal and accident conditions.

Maintenance Related Human Factors

The NGNP plant design shall enhance maintainability by including human factors considerations and providing adequate space, illumination, platforms, lifting equipment, handling gear, shielding, communications, HVAC, and compressed air supply.

Equipment modules shall be used in a way that modules may be removed for maintenance and interchanged. In areas expected to experience high radiation, plant layout shall consider the use of robotics. The Plant Designer shall provide features to facilitate critical maintenance activities. The Plant Designer shall consider providing mockups of critical equipment to facilitate training, minimize the potential for maintenance errors, and reduce personnel exposure to high radiation.

Preventive Maintenance and Inspections

The Designer/Constructor shall make recommendations for a preventive maintenance program to be established by the owner to ensure optimum performance and reliability of NGNP plant equipment.

Design Process

Balance in Design

The design process shall resolve competing interests in a manner which recognize the following priorities: (1) public and personnel safety, (2) investment protection, (3) plant performance, and (4) plant cost. Cost benefit and risk analyses may play a role in the resolution of these differences.

Design for Constructability

The design process shall be structured in a manner which acknowledges the close relationship between system design and constructability, operability, maintainability, and the associated staffing and training.

Technology Base

The HTR plant shall be designed using systems, components, and equipment proven through several years of acceptable service in similar plants. Proven systems, components, and equipment are those which have the same characteristics and that use materials proven under the same environmental and working conditions (stress levels, water chemistry, for example) as those which have been successfully applied for at least several years in similar operating environments and applications.

If the Plant Designer proposes to use systems or equipment not proven in the sense defined above, he shall provide justification to support his proposal. The experience base (such as in-plant use or testing) justifying the system and equipment shall be available for Plant Owner review.

Design Life

The design life for the NGNP shall be a minimum of 60 years. A design life plan shall be provided to meet reliability standards and minimize the cost of procuring and maintaining systems, structures, and components over this design life. This plan shall include a design life classification system, condition monitoring (part of the preventive maintenance and inspection program), and plant environmental monitoring system.

A 60-year design life is a requirement for all major plant components and equipment with reasonable expectation of meeting this design life. However, all plant operational components and equipment except the reactor vessel shall be replaceable, design life notwithstanding. For those components which are expected to be replaced for obsolescence or to avoid early failure, the Plant Designer shall provide a plan for replacement, refurbishment, and repair activities, as appropriate, to assure the design life of the overall plant is achieved at minimum cost.

Condition Monitoring

The design life plan and preventive maintenance programs shall include condition monitoring of all components, systems, and structures for which such monitoring is cost effective with respect to safety, reliability, and maintenance costs.

Transients

The design life plan and preventive maintenance programs shall include condition monitoring of all components, systems, and structures for which such monitoring is cost effective with respect to safety, reliability, and maintenance costs. (Avoidance or minimization of unnecessary transients enhances component life, and present plant designs indicate the need for increased attention in this area. Examples which deserve particular attention include heat exchanger loss of cooling transients, plant startup and cooldown transients, and spurious reactor trip).

Probabilistic Risk Assessment

A probabilistic risk assessment shall be carried out in the course of design, consistent with the PRA methodology. This PRA shall be integrated with the design process so that insights are used to enhance and optimize the design as it evolves. Significant changes to the design made primarily as a result of insights from the PRA shall be documented as part of the PRA. (It is the intent that the PRA be a "living" model which helps the Plant Designer in making decisions throughout the course of design. This level of integration provides a perspective on the design that can be of significant benefit in identifying and eliminating potential weaknesses early in the process. Close interactions also enhance the realism and meaningfulness of the PRA models).

<u>Mechanical Equipment Design Requirements</u>

The purpose of this section is to define common requirements that apply to mechanical equipment used in many systems of the plant. The following requirements are applicable to the passive and evolutionary LWRs. The requirements would need to be applied appropriately to the NGNP and HTR systems employing helium working fluid and environment in the primary heat transport and other systems.

Valves

In general, valves shall be applied and used in the NGNP in accordance with the guidelines provided in EPRI Report NP-6516, Guide for the Application and Use of Valves in Power Plant Systems.

Check valves shall be applied in the NGNP in accordance with the guidelines in EPRI Report NP-5479, Application Guidelines for Check Valves in Nuclear Power Plants, as applied to a new plant design. (Check valve failures in nuclear power plants have caused such problems as water hammer, system over pressurization, and steam binding of pumps. They have also been responsible for generating loose parts and, in general, have been a significant source of operational and maintenance problems).

MOVs shall be applied in the NGNP in accordance with the guidelines in EPRI Report NP-6660-D, Application Guide for Motor-Operated Valves in Nuclear Power Plants.

APPENDIX D: ULTIMATE HEAT SINK

D.1 Introduction

The Reactor Cavity Cooling System (RCCS) performs two functions:

- Protect the reactor cavity concrete temperatures (and thereby the reactor vessel and fuel temperatures) from exceeding the code permissible limits under normal, off-normal and accident conditions; and
- To provide an alternate means of core decay heat removal

Thus, the RCCS is required to operate continuously under all modes of plant operation.

D.2 Protecting Reactor Cavity Concrete

The temperature in the reactor cavity is controlled by the RCCS which is designed to maintain reactor cavity walls below maximum code acceptable concrete temperature limits during normal and accident conditions. ACI 349-80, Appendix A specifies allowable concrete temperatures as follows:

Normal operations or any other long period

- Bulk concrete: Temperature < 65.6 C (150 F)
- Local areas (such as around penetrations) < 93.3 (200 F)

Accident or any other ***short term*** *period*

- Bulk concrete: Temperature < 176.7 C (350 F)
- Local areas (such as around penetrations) < 343.3 (650 F)

Note that ACI 349-80 does not define what the “short term” duration is.

The Code allows higher temperature limits than those indicated above “if tests are provided to evaluate the reduction in strength and this reduction is applied to design allowables. Also, evidence shall be provided which verifies that the increased temperatures do not cause deterioration of the concrete either with and without load".

Under normal plant operating conditions, the RCCS is designed to maintain the concrete temperatures in the reactor cavity within the permissible limits by rejecting heat to the ultimate heat sink- air as shown on Figure D-1

D.3 Protecting reactor vessel

There is a code specified hard limit of 371 C for SA508, not counting margins to stay within negligible creep regime. If 9Cr vessel is used higher temperature are allowed.

D.4 Decay Heat Removal (while protecting reactor cavity concrete)

The means of decay heat removal is dependent on the event. The first lines of defense are through active systems that use secondary loop (and the steam generator) and the Shutdown Cooling System (SCS). In both cases the decay heat is rejected to conventional heat sinks- a river/large water reservoir and air.

Should the two active decay heat removal systems be not operational, the next stage of decay heat removal is accomplished through the RCCS in a passive mode. Thus, under a loss of offsite power or an accident condition, the RCCS accomplishes the task of maintaining the concrete temperatures as defined above while removing the decay heat from the reactor core. This is essentially accomplished by heat conduction from core to the reactor vessel. The reactor vessel rejects heat to the cooling panels principally through radiative heat transfer. Thus, high value of emissivity of the reactor vessel would have to be assured throughout the plant life to maintain the fuel and vessel temperatures within the permissible limits.

Under accident conditions, the inventory of water in the passive cooling system closed loop constitutes the ultimate heat sink. The inventory of the water in this system is relied upon to remove decay heat through sensible heat and subsequent boil-off. The passive system is maintained at a predetermined pressure level so that decay heat can be removed by sensible heat transfer for low end of the severe accidents. The duration and severity of the accident would be analyzed to determine:

- Inventory of water needed
- System pressure setting
- Concrete temperature transients following various accidents

- Reactor vessel temperature

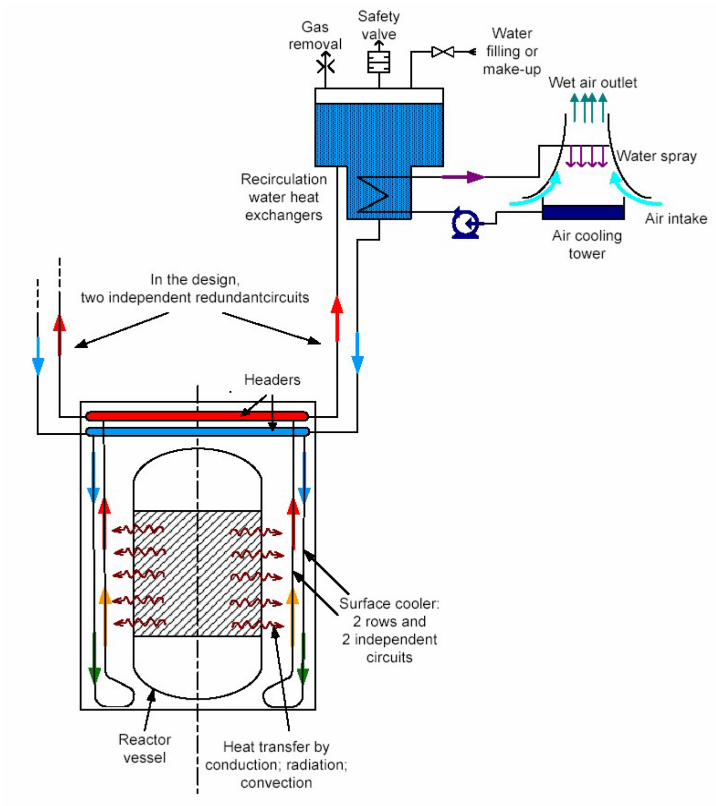


Figure D-1 Reactor Cavity Cooling System