

**SAFETY FEATURES OF THE CONTAINMENT OF A 4S
REACTOR-BASED POWER GENERATION FACILITY**

Prepared for the City of Galena, Alaska

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I EXECUTIVE SUMMARY

The philosophy under which nuclear power plants are designed and licensed in the United States calls for the provision of “defense in depth” against the release of radioactive contaminants from the plants to the outside environment. One of the essential elements of defense in depth is providing multiple barriers or levels of defense against radioactive releases. The containment is a crucial element in the implementation of the defense in depth concept, since it provides the final barrier against the escape of radioactive materials from a nuclear reactor. Accordingly, the adequacy of the containment design needs to be demonstrated for every new design whose licensing is proposed to the U.S. Nuclear Regulatory Commission (NRC).

While all aspects of accident prevention and mitigation are important, this paper focuses on the safety features of the containment that will be included in the design of the 4S Nuclear Reactor Based Power Generation Facility (NPF) proposed to be located in Galena, Alaska as part of the defense in depth against offsite radiation releases from the facility.

In the 4S design, fuel cladding provides the first barrier to the release of radioactive materials. The nuclear fuel is itself contained within the reactor vessel, which provides the second barrier to the release of radioactive material. The reactor vessel is supported by the pedestal via the upper reactor vessel flange. The lower portion of the reactor vessel is sealed within the guard vessel and the upper head of the reactor is protected within the top dome structure. Together, these structures comprise the containment system, which provides the final barrier to the release of radioactive material to the environment. All of these barriers are located within a heavily reinforced concrete structure below grade. The guard vessel and the top dome together are designed to have a leak rate sufficiently low to ensure that no combination of pressure and temperature conditions resulting from normal operations or from any design basis event can result in a significant leakage rate outside containment. Even in the event of an accident, it is expected that leakage through the 4S containment vessel wall will be virtually non-existent.

Unlike a water-cooled reactor, the 4S reactor cannot generate large mechanical loadings on the containment by the release of high temperature water that flashes into steam. The mechanical loading in the 4S containment boundary can only be caused by the annulus and cover gas pressure increases due to accidents that cause temperatures to increase. However, the cover gas region is a low-pressure region with no identifiable source of significant pressurization other than system temperature and level variation caused by differential expansion between the sodium and the reactor vessel. The cover gas pressure is approximately atmospheric during normal operation and increases to about 5 psig during loss of cooling events. Even beyond design basis events are not expected to challenge containment integrity through overpressure.

Other features of the 4S design serve to enhance the containment function. They include having an operating temperature for the coolant system that is well below the boiling point of the sodium coolant, thereby ensuring that cooling to the fuel cladding cannot be lost due to boiling even in the event of a breach of the pressure boundary. In addition, the reactor core is designed to have negative reactivity temperature coefficients meaning that the nuclear reactivity of the fuel decreases as the reactor heats up, with the result that the nuclear reactor will shut down if the

coolant temperature exceeds the design basis temperature. These design features ensure the stability of the reactor and prevents accidents from progressing to challenge the integrity of the containment.

The 4S design incorporates a Reactor Protection System (RPS) that is activated by redundant instruments that detect unusual neutron fluxes, temperature excursions, or fission gases outside of the reactor vessel. When activated, the RPS causes the reactor to safely shut down. This design feature further minimizes the probability that an accident will evolve into a challenge to the containment.

The Reactor Vessel Auxiliary Cooling System and the Intermediate Reactor Auxiliary Cooling System (RVACS / IRACS) are passive systems that remove energy from the reactor guard vessel and secondary sodium loops, respectively. The inherent capability of the design to remove decay heat through passive means avoids the need to resort to active systems to maintain the plant in a safe shutdown condition.

Auxiliary or supporting systems can affect the performance of safety systems. Passive auxiliary systems in place of active systems increase the likelihood that safety systems will operate as intended. In the 4S, elimination of all active cooling systems from the reactor side and elimination of the need for all emergency cooling systems results in a greater probability of successful safety system performance.

These and other design features of the 4S provide the bases for anticipating that there will be a low probability that an accident would result in the release of radioactivity to the environment outside the plant.

The risk that an accident at a 4S NPF at Galena will result in radioactive releases has not been formally evaluated. Such a risk evaluation would include consideration of three broad elements of risk:

- (1) Accident initiating events
- (2) Probability of failure of a fission product boundary (i.e., core damage/vessel failure)
- (3) Probability and consequences of release to the environment due to a failure of the containment.

Also, the timing of the releases (i.e., the response time between the initiation of the event and the arrival of the radioactive contamination at the site boundary) is of interest, since it allows assessment of the viability of potential response measures.

While a formal probabilistic risk assessment for the 4S NPF has yet to be issued, the calculated probability of a significant release from the Galena 4S NPF based power facility and potential for offsite dose consequences are expected to be lower than those for both advanced reactor designs and current generation reactors. The reasons for this expectation are:

- (1) The simple, passive features in the Galena reactor should result in a lower calculated probability of core damage than current generation plants.
- (2) The capability of the containment structure and its passive nature cooling capability provides a reliable barrier to release.

(3) The radionuclide inventory is approximately two orders of magnitude less than that used in the advanced designs currently certified or undergoing certification.

Galena and Toshiba should engage in early discussions with the NRC Staff to confirm the acceptability of the 4S containment system design. In addition, a probabilistic risk assessment should be performed for a 4S NPF in Galena. These activities will be of vital importance in determining the overall licensability and economic viability of the project.

II BACKGROUND

A. THE CITY OF GALENA

The City of Galena, Alaska (Galena) is a small community (pop. 700) located in west-central Alaska, along the banks of the Yukon River. The closest communities to Galena (within 100 air miles or less) are Koyukuk (pop. 100) approximately 30 miles to the west, Nulato (pop. 330), approximately 40 miles to the west, Kaltag (pop. 230), approximately 60 miles to the west, Ruby (pop. 190) approximately 50 miles to the east, and Huslia (pop. 300), approximately 70 miles to the northeast. The nearest major population center is Fairbanks (pop. 30,500), 270 miles to the east.¹

Galena has no roads linking it to the rest of the state. A former United States Air Force base, now known as the Edward J. Pitka Sr. Airport, is located 1.5 miles west of the city. The main runway of the airport is 7,254 feet long, and is capable of handling heavy transport type air traffic. The airport is the primary access point into and out of the Galena area, and operates year-round. The Yukon River serves as the major heavy transportation resource during the unfrozen summer months. Galena serves as an educational and cultural center for the region. There are many public use and commercial buildings in the area of the airport and the city itself including schools, workshops, and municipal buildings. Homes are predominately located around the “New Town” area, 1.5 miles east of the airport.

B. THE GALENA POWER SUPPLY

Galena has no connection to an outside power grid. The city currently depends on diesel generators for its electric power supply. Galena experiences long, severe winters. Winter low temperatures may reach -50°C (-60°F) and temperatures below -40°C (-40°F) are common. The lack of low-cost year-round heavy transport into Galena requires the city to maintain large diesel fuel tanks in order to meet energy demand. The escalating price of fuel and the associated costs of fuel transportation, storage, and financing make the cost of electricity prohibitively high to Galena residents. These economic issues, coupled with environmental pollution concerns, make it prudent for Galena to explore alternative ways to meet its energy needs.²

¹ U.S. Census Bureau, 2000 Census data, available online at <http://www.census.gov/popest/cities/tables/SUB-EST2004-04-02.csv>.

² Adams Atomic Engines, Inc., Atomic Insights, “Nuclear Power for Galena, Alaska” (March 2005), available online at http://www.atomicinsights.com/AI_03-20-05print.html.

C. THE GALENA 4S PROJECT

In 2004, Galena received presentations from Toshiba Corporation (Toshiba) on a “Super-Safe, Small and Simple” (4S) Nuclear Based Power Generation Facility. The 4S reactor was developed jointly by Toshiba and the Central Research Institute of Electric Power Industry (CRIEPI) of Japan.³ Following those presentations, Galena secured the preparation of a U.S. Department of Energy (DOE) sponsored study on ways to meet Galena’s power requirements.⁴ The study included analyses of the thermal and electric load profiles for Galena, technologies available to meet those loads (the technologies evaluated in detail were enhanced diesel power, coal, and a 4S NPF, which were determined to be the only viable alternatives), the environmental and regulatory issues associated with each of these technologies, and the overall economics of each energy option. The DOE study concluded that the 4S NPF is the best economic and environmental choice for Galena.

On December 14, 2004, the Galena City Council passed a resolution calling for the deployment of a 4S NPF in the community. The resolution stated, among other things, that: "It is in the public interest to pursue the siting of a Toshiba 4S nuclear battery in Galena." The council further directed the City Manager to "establish a process and timeline leading to evaluations, industrial partners, and financial and contractual arrangements necessary to bring the economic and environmental benefits of the 4S to Galena."

Since the passing of the resolution, Galena has been investigating the regulatory and economic feasibility of locating a 10 MWe 4S NPF in Galena. In parallel, Toshiba has been developing a Preliminary Safety Information Document (PSID) to submit to the U.S. Nuclear Regulatory Commission (NRC) for its review.⁵

In order to move the siting evaluation process forward and open lines of communication with various stakeholders and the NRC, Galena has commissioned a set of white papers that discuss important aspects of the small nuclear power facility project including a General Overview, Nuclear Liability and Insurance, Emergency Planning, Physical Security, Decommissioning, Containment, and Seismic Isolation. This paper is part of the white paper series.

D. UNIQUE FEATURES OF THE 4S REACTOR

The 4S design introduces a small liquid metal nuclear reactor to the commercial power industry in the United States.⁶ Liquid metal reactors (LMRs) have been operated successfully worldwide and have been used in the United States at test facilities, with over 300 reactor years of operational experience. The design of the 4S has several important operational and safety

³ See, e.g., <http://www.uaf.edu/aetdl/Presentations.htm>.

⁴ Robert E. Chaney et al., “Galena Electric Power- A Situational Analysis” (DE- AM26-99FT40575) (December 2004). Science Applications International Corporation (SAIC) coordinated the study, in which the University of Alaska and Idaho National Engineering and Environmental Laboratory participated.

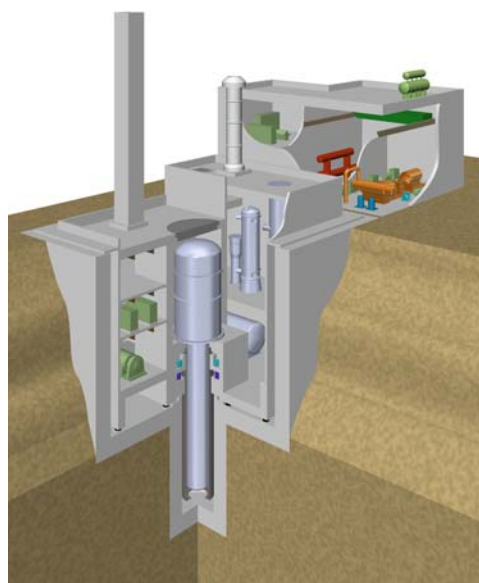
⁵ “Galena Project Officials Gear Up for Pre-Application Activities,” Inside NRC, February 6, 2006.

⁶ The peak thermal output of the 10 MWe 4S NPF is approximately 30 MW thermal (MWt), which is a small fraction of the power output of a standard sized commercial reactor.

advantages, particularly for remote location deployment, when compared to the large light-water commercial nuclear power plants currently operating in the United States. Important features of the design of the 4S include:

- Modular construction, which will reduce costs and construction time
- Nuclear systems that are embedded below grade, resulting in safety and security benefits
- Liquid sodium coolant, which does not react with core internals or piping
- Coolant that is not highly pressurized, which minimizes stresses on the plant systems
- Passive safety systems that do not depend on emergency power to function
- Negative reactivity temperature coefficients that cause the reaction rate in the core to slow down as temperatures rise
- Air-cooled reactor vessel, steam generator, and condenser, so that no coolant water or intake structures are required
- 30-year core life, which avoids the need to refuel, eliminates fuel storage, and minimizes fuel handling concerns
- Capability of load following without mechanical operation of reactor control system
- Ease of decommissioning by containment of all radioactive materials within the reactor module throughout the life of the plant.

These unique features are among those that provide the 4S reactor system with significant benefits in operational capability, physical security, and protection of public safety. Many of the systems that increase cost, raise safety concerns, and pose potential security hazards at current plants (such as use of numerous mechanical pumps and valves, the need for a spent fuel pool, and the reliance upon high and low pressure coolant injection systems) have been eliminated in the design of the 4S. While the 4S reactor system does raise some new issues, such as the need to deal with highly reactive liquid sodium and potential accident scenarios involving sodium-water interaction, these issues have been addressed in the 4S system design and in past LMR facilities. The licensing of a 4S NPF should therefore be a relatively straightforward process, provided that good communications are maintained between all parties involved and there is a timely flow of complete and accurate technical information.



On balance, the small size of the 4S NPF and its many inherently protective features should greatly facilitate plant licensing. The figure to the right is a 3D rendition of the 4S power generation facility. It depicts the reactor building (lower left), which is below grade, and the turbine generator building (upper right). To put the small size of this facility in perspective, the overall dimensions of the below ground and above ground structures are approximately 190 feet long and 90 feet wide, and can be accommodated in less than ½ acre of land.

The reactor building (also referred to as the Nuclear Island) houses the reactor module (lowest left corner of figure), the steam generator (slightly higher and to the right of the reactor module), and other vital safety equipment. As can be seen in the figure, the lower part of the reactor module (containing the reactor core) is located in its own below grade silo-like reinforced concrete structure. The heavy reinforced concrete cap like structure on the top of the reactor module encloses the reactor assembly and core and provides the only access to the nuclear fuel, which is placed within the reactor module. Heavy lifting equipment is required to remove the cap and gain access the nuclear fuel.

The major components enclosing the reactor assembly and core are the reactor vessel, the guard vessel, the guard vessel top dome, and the heavily reinforced concrete structure on top of the reactor module. The guard vessel surrounds the reactor vessel and ensures that the core will not be uncovered and core cooling can be accomplished even if the reactor vessel leaks. The reactor vessel is the container and support for the reactor core, the primary radioactive sodium coolant, and the primary sodium intermediate heat exchanger structures. The reactor core is designed to have negative reactivity temperature coefficients meaning that the nuclear reactivity of the fuel decreases as the reactor heats up, such that nuclear reactions in the core would cease for beyond design basis events that would raise the temperature of the core.

The primary reactor coolant system (PRCS) through which the radioactive sodium coolant flows is wholly contained within the sealed reactor vessel. The radioactive sodium heated by the reactor core enters and flows through two intermediate heat exchangers (IHX), located within the reactor vessel, where it is cooled as it heats non-radioactive sodium that circulates through the intermediate heat exchanger (IHX). All radioactive materials in the 4S are limited to the PRCS and the fuel, both of which are wholly contained within the reactor vessel. No radioactive materials are ever removed from 4S reactor vessel throughout its 30-year life.

The secondary heat transport system (SHTS) transports heat from the primary system to the steam generator (SG) where it produces steam that drives conventional steam turbine equipment. The SHTS is comprised of a piped loop thermally coupled to the primary system by the IHXs located in the reactor vessel and to the SG located in the SG compartment. The sodium that is circulated through the SHTS to transfer the heat from the primary system to the SG system is non-radioactive. The non-radioactive SHTS sodium entering the shell side of the SG heats the water on the tube side to produce steam, which drives two steam turbines located within the turbine generator building. The steam is cooled by the main condenser, which is also located with the turbine generator building. The condensed water is re-circulated through the SG system to repeat the process.

The 4S reactor shutdown heat removal systems consist of main condenser cooling, an intermediate reactor cooling auxiliary system (IRACS) for the SG, and the safety related reactor vessel auxiliary cooling system (RVACS) that removes heat directly from the reactor. The RVACS is a passive safety related system that transports heat to the atmosphere by natural circulation of air. It functions continuously with its heat transport rate governed by the reactor vessel temperature. During an event involving loss of ability to remove heat via the main condenser or the IRACS, the resultant higher primary sodium temperatures will raise the reactor vessel temperature and cause the RVACS to respond automatically with a corresponding increase in heat removal rate sufficient to maintain cooling of the 4S.

The above features of the 4S design and related features of the design are discussed at greater detail in the overview white paper⁷ prepared as part of this series.

III RELEVANT LAW AND REGULATIONS

Accident prevention and mitigation requirements for entities that possess reactor operating licenses are spread throughout Title 10 of the Code of Federal Regulations (CFR), particularly Part 50, which provides the main regulations covering the design of current generation reactors.⁸ Numerous design, construction and operation requirements must be met to ensure the safety of the facility. However, the four most significant sections regarding the containment function are 10 C.F.R. §§ 50.40(a), 50.40(c), 50.57(a)(6) and Appendix A to Part 50, Criterion 16.

Section 50.40(a) indicates that a “common standard” that must be met by all prospective Part 50 licensees is that the license application must demonstrate that

The processes to be performed, the operating procedures, the facility and equipment, the use of the facility, and other technical specifications, or the proposal, in regard to any of the foregoing collectively provide reasonable assurance that the applicant will comply with the regulations in this chapter, including the regulations in part 20, and that the health and safety of the public will not be endangered.

This requirement is reinforced later in that same section with the requirement that

The issuance of a license to the applicant will not, in the opinion of the Commission, be inimical to the common defense and security or to the health and safety of the public.

§ 50.40(c).

In the section of Part 50 concerning the criteria for issuing an operating license to a facility, the regulations state that an operating license may be issued upon finding that:

* * *

(6) The issuance of the license will not be inimical to the common defense and security or to the health and safety of the public.

⁷ Overview of Galena’s Proposed Approach to licensing a 4S Reactor-Based Power Generation Facility (Overview White Paper).

⁸ A 4S NPF at Galena would be licensed under the provisions of 10 C.F.R. Part 52. However, 10 C.F.R. § 52.83 directs that, unless provided otherwise in 10 C.F.R. Part 52, the provisions in 10 C.F.R. Part 50 apply to Part 52 reactors. Proposed amendments to Part 52 would not change the applicability of relevant Part 50 provisions but would incorporate them explicitly into the amended Part 52. See 71 Fed. Reg. 12782 (2006). Part 52, both in its current form and under the proposed amendments, does not impose any additional or different containment design standards than those contained in Part 50.

§ 50.57(a).

Finally, and most directly on point, Appendix A to Part 50 contains the “General Design Criteria (GDC) for Nuclear Power Plants.” Criterion 16 of Appendix A states

Containment Design. Reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

This paper addresses the design features of the 4S NPF containment that would affect the ability of the proposed Galena facility to comply with these requirements.

Appendix A to Part 50 also contains numerous other containment-related Design Criteria that are more appropriate to Light Water Reactors (LWRs) and their special requirements. The following criteria are either inapplicable or are clearly satisfied by the 4S design:

Criterion 38 - Containment Heat Removal

A system- to remove heat from the reactor containment shall be provided. The system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any LOCA and maintain them at acceptable low levels. Suitable redundancy in components and features and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that, for onsite electric power system operation (assuming off site power is not available) and for off site electric power system operation (assuming onsite power is not available), the system safety function can be accomplished, assuming a single failure.

Evaluation of the 4S against Criterion 38

For 4S, a containment heat removal system in the safety system is not required.

Criterion 39 - Inspection of Containment Heat Removal System

The containment heat removal system shall be designed to permit appropriate periodic inspection of important components, such as pumps, spray nozzles, and piping, to assure the integrity and capability of the system.

Evaluation of the 4S against - Criterion 39

For 4S, a containment heat removal system in the safety system is not required.

Criterion 40 - Testing of Containment Heat Removal System

The containment heat removal system shall be designed to permit appropriate periodic pressure and functional testing to assure:

1. the structural and leak-tight integrity of its components;
2. the operability and performance of the active components of the system; and

3. the operability of the system as a whole, and, under conditions as close to the design as practical, the performance of the full operational sequence that brings the system into operation including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

Evaluation of the 4S against Criterion 40

For 4S, a containment heat removal system in the safety system is not required.

Criterion 41 - Containment Atmosphere Cleanup

Systems to control fission products, hydrogen, oxygen, and other substances which may be released into the reactor containment shall be provided as necessary to reduce, consistent with the functioning of other associated systems, the concentration and quantity of fission products released to the environment following postulated accidents and to control the concentration of hydrogen or oxygen and other substances in the containment atmosphere following postulated accidents to assure that containment integrity is maintained. Each system shall have suitable redundancy in components and features and suitable interconnections, leak detection, isolation, and containment capabilities to assure that, for onsite electric power system operation (assuming off site power is not available) and for off site electric power system operation (assuming onsite power is not available), its safety function can be accomplished, assuming a single failure.

Evaluation of the 4S against Criterion 41

As the containment facility capacity is very small and aerosol concentration is likely to decrease due to natural processes, this GDC is not applicable to 4S. Recovery from a very low frequency accident, such as release of fission products, is performed using an ordinary gas processing system.

Inspection of Containment Atmosphere Cleanup Systems Criterion 42

The containment atmosphere cleanup systems shall be designed to permit appropriate periodic inspection of important components, such as filter frames, ducts, and piping, to assure the integrity and capability of the systems.

Evaluation of the 4S against Criterion 42

As 4S does not require a cleanup system in the containment facility (see GDC41), this Criterion is not applicable to 4S.

Criterion 43 - Testing of Containment Atmosphere Cleanup Systems

The containment atmosphere cleanup systems shall be designed to permit appropriate periodic pressure and functional testing to assure:

1. the structural and leak-tight integrity of its components;
2. the operability and performance of the active components of the systems such as fans, filters, dampers, pumps, and valves; and
3. the operability of the systems as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the systems into operation including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of associated systems.

Evaluation of the 4S against Criterion 43

As 4S does not require a cleanup system in the containment facility (see GDC41), This Criterion is not applicable to the 4S.

Criterion 50 - Containment Design Basis

The reactor containment structure, including access openings, penetrations, and the containment heat removal system shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any LOCA. This margin shall reflect consideration of:

1. The effects of potential energy sources which have not been included in the determination of the peak conditions such as energy in steam generators and energy from metal water and other chemical reactions that may result from degraded emergency core cooling functioning;
2. The limited experience and experimental data available for defining accident phenomena and containment responses; and
3. The conservatism of the calculation model and input parameters.

Evaluation of the 4S against Criterion 50

The 4S containment facility is designed such that it can accommodate pressure and temperature conditions calculated in the normal operating state and any and all design basis events, including leakage of coolant sodium into the containment facility, without exceeding the design leakage rate and with sufficient margin. Margin of uncertainty in the event of accidents is included in the design.

Criterion 51 - Fracture Prevention of Containment Pressure Boundary

The reactor containment boundary shall be designed with sufficient margin to assure that under operating, maintenance, testing, and postulated accident conditions:

1. Its ferritic materials behave in a non-brittle manner; and
2. The probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the containment boundary material during operation, maintenance, testing, and postulated accident conditions and the uncertainties in determining:
 1. Material properties;
 2. Residual, steady-state, and transient stresses; and
 3. Size of flaws.

Evaluation of the 4S against Criterion 51

The 4S containment facility is designed with sufficient margin so that integrity is ensured with respect to operation, maintenance, testing and postulated accident conditions. The containment facility is made of stainless steel (304) with no-brittle behavior and the probability of rapid propagating fracture is minimized.

Criterion 52 - Capability for Containment Leakage Rate Testing

The reactor containment and other equipment which may be subject to containment test conditions shall be designed so that periodic integrated leakage rate testing can be conducted at containment design pressure.

Evaluation of the 4S against Criterion 52

The leakage rate is measured periodically.

A periodic leakage rate test is performed by measuring temperature and pressure at a specified time interval. The change in gas pressure is converted into a percent volume leakage rate per day using the gas state equation. The temperature between the reactor vessel and the containment vessel is measured in the reactor cover gas space and annulus between reactor vessel and guard vessel. Pressure is measured using a high precision gage. From these measurements, the leakage rate of the containment facility is obtained.

Criterion 53 - Provisions for Containment Testing and Inspection

The reactor containment shall be designed to permit:

1. Appropriate periodic inspection of all important area such as penetrations;
2. An appropriate surveillance program; and
3. Periodic testing at containment design pressure of the leak-tightness of penetrations which have resilient seals and expansion bellows.

Evaluation of the 4S against Criterion 53

Tests and inspections are performed in accordance with the following criteria:

Refer to the 2001 Edition with the 2003 Addenda of ASME Code, 2004 Edition, Section XI, Division 3. Refer to Examination and pressure test requirements in Subsection IME-2500 of the 2001 Edition with the 2003 Addenda, Section XI, Division 3.

Criterion 54 - Piping Systems Penetrating Containment

Piping systems penetrating primary reactor containment shall be provided with leak detection, isolation, and containment capabilities having redundancy, reliability and performance capabilities which reflect the importance to safety of isolating these piping systems. Such piping systems shall be designed with a capability to periodically test the operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits.

Evaluation of the 4S against Criterion 54

The piping system for auxiliary components that penetrates all the containment vessels of 4S is designed to enable periodic tests and leakage confirmation tests in order to provide leak detection, isolation and containment capabilities.

Criterion 55 - Reactor Coolant Pressure Boundary Penetrating Containment

Each line that is part of the reactor coolant pressure boundary and that penetrates primary reactor containment shall be provided with containment isolation valves as follows unless It can be demonstrated that the containment isolation provisions for a specific class of lines such as instrument lines are acceptable on some other defined basis:

1. One locked-closed isolation valve inside and one locked-closed isolation valve outside containment; or
2. One automatic isolation valve inside and one locked-closed isolation valve outside containment; or
3. One locked-closed isolation valve inside and one automatic isolation valve outside containment (a simple check valve may not be used as the automatic isolation valve outside containment); or
4. One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment. Isolation valves outside containment shall be located as close to the

containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.

Other appropriate requirements to minimize the probability or consequences of an accidental rupture of these lines or of lines connected to them shall be provided as necessary to assure adequate safety. Determination of the appropriateness of these requirements (such as higher quality in design, fabrication, and testing, additional provisions for in-service inspection, protection against more severe natural phenomena, and additional isolation valves and containment) shall include consideration of the population density, use characteristics, and physical characteristics of the site environs.

Evaluation of the 4S against Criterion 55

Since all the primary sodium is held in the reactor vessel during reactor operation, this Criterion is not applicable to 4S.

Criterion 56 - Primary Containment Isolation

Each line that connects directly to the containment atmosphere and penetrates primary reactor containment shall be provided with containment isolation valves as follows unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis:

1. One locked-closed isolation valve inside and one locked-closed Isolation valve outside containment; or
2. One automatic isolation valve inside and one locked-closed isolation valve outside containment; or
3. One locked-closed isolation valve inside and one automatic isolation valve outside containment (a simple check valve may not be used as the automatic isolation valve outside containment); or
4. One automatic isolation valve inside and one automatic isolation valve outside containment (a simple check valve may not be used as the automatic isolation valve outside containment). Isolation valves outside containment shall be located as close to the containment as practical and automatic Isolation valves shall be designed to take the position that provides greater safety upon loss of actuating power.

Evaluation of the 4S against Criterion 56

All sodium and gaseous service lines that penetrate the top dome are closed with redundant isolation valves. Lines penetrating containment vessel of 4S are listed as following;

- Nitrogen gas supply line
- Argon gas supply line
- Top dome cooling system
- Sodium supply line

Criterion 57 - Closed System Isolation Valves

Each line that penetrates primary reactor containment and is neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere shall have at least one containment isolation valve which shall be either automatic, locked-closed, or capable of remote manual operation. This valve shall be outside the containment

and located as close to the containment as practical. A simple check valve may not be used as the automatic isolation valve.

Evaluation of the 4S against Criterion 57

Isolation valves are installed at relevant locations.

IV ANALYSIS

A. THE 4S CONTAINMENT

1. Introduction

The philosophy under which nuclear power plants are designed and licensed in the United States calls for the provision of “defense in depth” against the release of radioactive contaminants from the plants to the outside environment. One of the essential properties of defense in depth is the concept of successive barriers or levels of defense against such releases. The multiple physical barrier approach is often described as encompassing the fuel cladding that holds radioactive matter within the metal enclosure of the cladding; the primary reactor system within which any radioactive matter that escapes the cladding will be held; and the containment that provides a direct barrier against the escape of radioactive materials to the outside environment.⁹

The containment is a crucial element in the implementation of the defense in depth concept, since it provides the final protection against the escape of radioactive materials from a nuclear reactor. Accordingly, the adequacy of the containment design needs to be demonstrated for every new design whose licensing is proposed to the NRC.

2. Description of the 4S Containment

In the 4S design, the reactor vessel contains the radioactive core, the primary the primary radioactive sodium coolant, and the primary sodium intermediate heat exchanger structures. In turn, the reactor vessel is surrounded by the guard vessel and the top dome within a heavily reinforced concrete structure. The entire containment is installed underground.

As provided in Criterion 16 of Appendix A to 10 CFR Part 50, the function of the 4S containment is to provide a leak tight barrier that ensures that no radioactivity escapes outside its boundary even if the reactor vessel leaks or if radioactive matter is otherwise released from the reactor. The containment is designed to accommodate the calculated pressure and temperature conditions resulting from normal operations and all design basis events, including coolant inleakage, without exceeding the design leak rate.

⁹ See, e.g., Advisory Committee on Reactor Safeguards (ACRS), “On the Role of Defense in Depth in Risk-Informed Regulation,” attached to ACRS letter, May 19, 1999.

3. Containment Pressure and Leakage Protection Features

Mechanical loading in the containment boundary can be caused by the annulus and cover gas pressure increases due to accidents that cause temperatures to increase. (Sodium coolant pressure is not an important contributor because the sodium is considerably subcooled even during severe transients.) The cover gas region is also a low-pressure region with no identifiable source of significant pressurization other than system temperature and level variation caused by differential expansion between the sodium and the reactor vessel. The cover gas pressure is approximately atmospheric during normal operation and increases to about 5 psig during loss of cooling events. Because of the inherent negative feedback mechanisms within the 4S core, even beyond design basis events do not challenge containment integrity through overpressure.

During reactor operation, the guard vessel annulus is continuously monitored to detect sodium or helium cover gas leakage from the reactor vessel, thus providing an implicit on-line containment test. Even in the event of an accident, it is expected that leakage through the 4S containment vessel wall will be virtually non-existent.

B. 4S DESIGN FEATURES BEARING ON CONTAINMENT

There are a number of features of the design and operation of the 4S reactor that relate to the containment function. Generally, the passive nature of the plant's safety systems, the low power output, and the underground location of the nuclear steam supply system facilitate providing an effective containment of radioactive materials.

The following discussion summarizes those features of the 4S NPF design that most directly affect containment. These design features provide the bases for anticipating that there will be a low probability that an accident would result in the release of radioactivity to the environment outside the plant.

1. Accident Prevention

a. Load-following during normal operation

A desirable feature of an electric power generation system is the ability of the system to "follow" the system load; that is, to adapt the power output to meet moment-to-moment demand in the electric load it serves in order to ensure that the power source is producing neither too little nor too much energy. Load-following is achieved innovatively in the 4S by controlling water flow to the steam generator based on generation demand, which subsequently manipulates the core inlet temperature. As the generator output matches the load, changes in the coolant temperature introduce a positive or negative reactivity effect in the core, causing the reactor power to follow. This load-following capability reduces the likelihood of reactor trips and increases plant safety by minimizing the occurrence of off-normal events.

b. Criticality and power control

The 4S reactor has a negative sodium void reactivity coefficient, which means that power output naturally decreases as system temperature increases. This negative void reactivity coefficient is achieved by keeping the core diameter small. The fuel temperature coefficient is also negative, enhancing this characteristic.

Reactivity in the 4S is controlled by a unique neutron reflector system surrounding the core that slowly moves upward from the bottom of the core to compensate for reactivity loss from fuel burnup. A drive system moves the reflector at a rate of 1 mm per week over the life of the core. If an accident occurs, the reflector drops down, causing the reaction to stop. In addition, a gravity-driven neutron absorber rod located at the center of the core provides a second, independent shutdown system.

2. Operating Pressure and Temperature

Pressurization of the 4S NPF containment is much less likely than in LWRs, because the reactor coolant system is operated at ambient pressure. The use of liquid sodium as a coolant for the 4S reactor permits operation at nearly atmospheric pressure with a large margin to the boiling point of the coolant (subcooling margin). Maintaining the core coolant below boiling provides assurance that the fuel cladding is not overheated. The high boiling point of liquid sodium means that less energy is transferred to the containment vapor space if the reactor pressure boundary fails. The use of liquid sodium also eliminates hydrogen generation from water-cladding interaction. As a result, the guard vessel and containment dome volumes can be small, which allows for effective passive cooling.¹⁰

The reactor core is designed to have negative reactivity temperature coefficients, including coolant void reactivity, meaning that the nuclear reactivity of the fuel decreases as the reactor heats up, such that nuclear reactions in the core would cease upon the occurrence of beyond design basis events that would raise the temperature of the core.

3. Reactor Protection System

While the negative void reactivity coefficient will always cause the reactor power level to self-moderate, the 4S NPF design nonetheless incorporates a Reactor Protection System (RPS) that is activated by loss of power, by the neutron detectors installed outside of the reactor vessel, or by the IHX outlet temperature detector in the reactor vessel. When activated, the RPS causes the reactor to safely shut down. This design feature minimizes the probability that an accident will evolve into a challenge to the containment.

4. Passive Heat Removal Systems

The Reactor Vessel Auxiliary Cooling System and the Intermediate Reactor Auxiliary Cooling System (RVACS / IRACS) are passive systems that remove energy from the reactor guard vessel and secondary sodium loops, respectively. These passive safety systems are designed so that no power is required for valve movements to initiate them.

¹⁰ Safety and Security Features of 4S Type Reactors,” D. Wade (Argonne) presentation to Alaska Rural Energy Conference, Sept 20-22, 2005.

The inherent capability of the design to remove decay heat through passive means avoids the need to resort to active systems to maintain the plant in a safe shutdown condition. Table 1 below illustrates the simplicity of the 4S safety systems by comparing them to those in current generation nuclear power plants. As Table 1, shows, none of the systems/features upon which a current generation reactor relies is required to maintain plant safety in a 4S NPF. Elimination of the need for these active systems increases the safety of the 4S design.

Table 1
Comparison of Current Generation Plant Safety Systems to the 4S Design

Current Generation Safety Related Systems	4S Reactor Safety Systems
High Pressure Injection System Low Pressure Injection System	No active safety injection systems are required. Core cooling is maintained without injection because of the low pressure and high heat capacity of liquid sodium. The guard vessel provides assurance against the possibility of the core becoming uncovered.
Emergency sump and associated net positive suction head requirements for safety-related pumps	No safety-related pumps; therefore, no need for sumps and protection of their suction supply.
Emergency Diesel Generators	Passive design does not require emergency power to maintain core cooling. Core heat is removed by heat transfer through the vessel.
Active Containment Heat Systems	None required because of passive heat rejection out of containment.
Containment Spray System	Spray systems are not required to reduce steam pressure or to remove radioiodine from containment.
Emergency Core Cooling System Initiation, Instrumentation and Control Systems - complex systems require significant amount of online testing.	Simple monitoring of IHX outlet temperature and neutron flux initiates reactor trip. No other protective actions are required (the reactor still shuts down even without an automatic trip signal).
Emergency Feedwater System, Condensate Storage Tanks and associated emergency cooling water supplies	In an accident scenario, core cooling is accomplished by means of the high heat capacity of liquid sodium. The ability to remove core heat without an emergency feedwater system is a significant safety enhancement.

5. Support systems

Auxiliary or supporting systems can affect the performance of safety systems. Use of passive auxiliary systems in place of active systems increases the likelihood that safety systems will operate as intended. In the 4S NPF, elimination of all active cooling systems from the reactor

side and elimination of the need for all emergency cooling systems results in a greater probability of successful safety system performance.

Radiated heat from the reactor vessel is removed by the Reactor Vessel Auxiliary Cooling System. The heat conducted into the containment is removed by the natural air cooling from the surface of the containment. The primary pump is a sodium-immersed electromagnetic pump, so no motor or pump seal cooling is required. Natural circulation within the reactor is sufficient to cool the reactor if the electromagnetic pump stops. As the result, all active cooling systems are eliminated. The differences between traditional LWRs and the 4S NPF with respect to support systems is summarized in Table 2.

Table 2
Comparison of Current Generation Plant Support Systems to those in the 4S Design

Current LWR Support Systems	4S Reactor Support Systems
<i>Reactor Coolant Pumps.</i> Leakage of pump seals has been a safety concern. Seal maintenance and replacement is costly and time consuming.	The electromagnetic pumps in the 4S NPF design eliminate the need for seals.
<i>Ultimate heat sink and associated interfacing systems.</i> River and seawater systems are active systems, subject to loss of function from such causes as extreme weather conditions and bio-fouling.	The 4S NPF design is cooled passively, and rejects heat by conduction and convection.
<i>Closed cooling water systems.</i> These systems are required to support safety related systems for removal of core and equipment heat.	No closed cooling water systems are required for safety-related systems.
<i>Heating, ventilation and air conditioning (HVAC).</i> These systems are required to function to support proper operation of safety-related systems.	The reactor building compartments contain the primary flow coastdown system and electric panels. The plant design does not require safety-related room cooling, eliminating the need for a safety-related HVAC system. ¹¹

C. RISK CONSIDERATIONS

1. Introduction

A probabilistic risk assessment for the 4S NPF is not currently available. It is expected, however, that a 4S NPF located at the Galena site would have a much lower calculated probability of release than current generation plants.¹² This low level of risk is consistent with

¹¹ Hattori, S. and A. Minato, ASME 1993. "Passive Safety Features In 4S Plant," Central Research Institute of Electric Power Industry. Tokyo, Japan ASME/JSME Nuclear Engineering Conference, Volume 1.

¹² Advanced light water reactor core damage frequencies due to internal events are in the range of 10^{-6} to 10^{-7} per reactor year compared to the NRC safety goal of 10^{-4} per year.

the significantly improved risk profile due to the smaller core inventory and inclusion of advanced design features such as passive safety systems.¹³

The discussion that follows addresses some of the key features of the 4S NPF that contribute to a reduced likelihood of core damage and release in comparison to the large, current generation facilities. These features would be taken into account in implementing a risk-based licensing approach for the Galena 4S NPF. Such a risk evaluation would include consideration of three broad elements of risk: (1) accident initiating events, (2) probability of failure of a fission product boundary (i.e., core damage/vessel failure), and (3) probability and consequences of release to the environment due to a failure of the containment. Also, the timing of the releases (i.e., the response time between the initiation of the event and the arrival of the radioactive contamination at the site boundary) is of interest, since it allows assessment of the viability of potential response measures.

2. Accident Initiators for Galena 4S

In general, the frequency of events that could lead to core damage in the 4S NPF would be less than that for current generation plants because of the simplicity of the design, reduced need for operator action, and the physical capability of the 4S NPF to passively accommodate heat removal from both the reactor and containment. A detailed probabilistic risk analysis (“PRA”) would need to be performed to quantify the likelihood that the various initiating events would lead to releases outside containment.

Potential accident initiators are grouped into two categories: (1) “internal” events and (2) “external” events. Internal initiating events include system failures such as loss of site power. External events include natural occurrences such as earthquakes and common mode failures such as fires. The location of the Galena 4S NPF introduces the possibility that some external events initiators may have a higher frequency than typically observed for large power reactors. For example, external initiating events associated with extreme weather conditions might be more likely at the Galena site than at most currently operating reactors. If that were the case, the 4S design would need to take into account the increased initiator frequencies in assessing the risk of containment failure at Galena.

a. Internal Events

The spectrum of internal events typically considered as accident initiators for the current generation light water reactor includes anticipated transients during normal operation and the less likely postulated accidents such as a loss of reactor coolant. Transients may be associated with the reactor function (e.g., failure to scram) or with the power generation function (e.g., closure of steam stop valves). Some of these events have a reduced frequency or can be eliminated as accident initiators in the 4S NPF design based on the plant’s capability to cope with the event.

¹³ S. Hattori, “Energy Source for the Human Demand,” Central Research Institute of Electric Power Industry, Japan cites a potential risk reduction of more than a factor of 1000 for small passive reactors in comparison to large current generation plants.

Table 3 below provides examples of typical current generation plant accident initiators and their applicability to the 4S NPF. A 4S-specific PRA would identify initiators that are unique to this design and the associated frequencies of such initiators.

Table 3
Applicability of Typical Internal Accident Initiators to 4S NPF

Initiator from Current Generation Plant	Applicability to 4S Design
Inadvertent Reactivity Insertion (Transient Overpower) Events: Rod ejection accident Boron dilution accidents Rod withdrawal	The 4S plant does not have control rods or rely on boron for reactivity control. Reactivity control for start-up, shutdown and core burnup is performed automatically without operator intervention. Rapid withdrawal of the reflector at cold shutdown could result in a reactivity accident. The system must be heated to 350°C before start up to preclude excess reactivity additions from this event. At system temperatures below 350°C, the neutron absorber is physically prevented from being withdrawn. The most limiting reactivity insertion is the reflector rising continuously to its highest position at the highest speed mechanically possible. ¹⁴ Neither the coolant nor the center of the fuel rises higher than 750°C. ¹⁵ With this reactivity, the low linear power design (about 1/3 of an ordinary design) gives a significant margin to fuel melting. ¹⁶
Overcooling transients	The overcooling events would be expected to result in small positive reactivity caused by changes in pump speed and the corresponding reduced primary system temperatures. Core temperatures increase slowly and the peak temperature is bounded by the transient overpower events. Negative reactivity assists in moderating an overcooling event.
Loss of reactor coolant flow Loss of Load Loss of Electro Magnetic Pump (EMP)	In Loss of Flow events, the fuel integrity depends on the peak fuel/cladding interface temperature. Important key parameters are the primary flow coastdown characteristics and the reactivity feedback. Analyses performed for this event show safe transient responses. Pump flow coastdown and negative reactivity feedback caused by core and axial fuel expansions maintained fuel integrity protection during this event. ¹⁷
Loss of heat sink Partial loss of feedwater Complete loss of feedwater	System behavior for a Loss of Heat Sink is similar to but milder than a Loss of Flow initiated accident.
Loss of Coolant Accident (LOCA)	The 4S primary coolant system is not subject a loss of coolant accidents such as those that may occur in light water reactors. The core is near the low point of the reactor vessel and the system operates at nearly atmospheric pressure. Failures in the system boundary do not uncover the core, as the coolant is retained in the guard vessel. Since the liquid sodium operates at hundreds of degrees below its boiling point, there is no loss of inventory by vaporization through a break in containment. Core cooling is maintained by natural passive conduction of heat to the vessel walls.
Local blockage	The 4S design provides a small potential for local blockages, because the 4S design has a low potential for creating loose parts. The 4S maintains the coolant in an enclosed cover gas, and the 4S metallic fuel is compatible

¹⁴ "Passive Safety Features in 4S Plant," Hattori, S. and A. Minato, ASME/JSME Nuclear Engineering Conference, Vol. 1, 1993.

¹⁵ "Meet Global Needs by 4S Plant," Hattori, S. and A. Minato, ASME/JSME Nuclear Engineering Conference, Vol. 2, 1993.

¹⁶ "Present Design Features Of The Super Safe, Small And Simple Reactor," Hattori, S. et al., Central Research Institute of Electric Power Industry.

¹⁷ Ibid.

Table 3
Applicability of Typical Internal Accident Initiators to 4S NPF

Initiator from Current Generation Plant	Applicability to 4S Design
	with sodium. The excellent thermal conductivity of metallic fuel mitigates the fuel temperature rise if a blockage were to occur.
Steam generator tube rupture	Conservative analyses indicate that the maximum pressure remains below and the boundary tube is maintained. ¹⁸
Total loss of all station power	The system temperature does not exceed 650°C in this accident, which is within safe design limits. ¹⁹
Loss of decay heat removal	The system temperature does not exceed 650°C in this accident. ²⁰
Anticipated Transient without Scram Reactivity insertion and loss of offsite power under horizontal earthquake Reactivity insertion by sudden ΔT Loss of load Sudden loss of EMP Total loss of electric power	The largest temperature rise is caused by a horizontal earthquake accompanied by simultaneous insertion of reactivity and loss of flow under assumption of no scram. The temperature of the fuel cladding rises up to 850°C, but it drops in a short period of time without causing cladding damage by eutectic reaction. The fuel does not melt. ²¹ The 4S reactor fuel starts melting at a temperature of xxx [?] ^o C.

b. External Events

The characteristics of the Galena site introduce the possibility that certain external events may be the dominant accident initiators. In particular, earthquake risk is a dominant contributor in some Japanese reactors; the Galena location could introduce a similar situation. Additional external events that would be of particular concern for the Galena reactor include:

- Flood: The site is located adjacent to the Yukon River above the 100 year flood plain. It appears that flood levels would not be substantial. However, the reactor is located underground and groundwater intrusion or flooding of the buildings will be a consideration when designing the containment and the outer vault containment liner.
- External fire: The site includes wooded areas that could challenge plant operation from an offsite forest fire.

¹⁸ “Passive Safety Features In 4S Plant”, Hattori, S. and Minato A., ASME/JSME Nuclear Engineering Conference, Vol 1, 1993.

¹⁹ “The Super Safe, Small and Simple Reactor (4S-50)”, Hattori, S. and Minato A., International Conference on Design and Safety of Commercial Nuclear Power Plants. Oct 25-29, 1992 Tokyo, Japan.

²⁰ Ibid.

²¹ “Current Status Of 4S Plant Design,” Hattori, S. and A. Minato, ASME/JSME Nuclear Engineering Conference, Vol. 2 ASME 1993.

- Extreme cold: Temperatures of -60°F represent unique challenges to equipment. A reactor trip under extreme cold conditions could interfere with the operation of plant equipment until auxiliary power is available to provide heat (e.g. a long station blackout coping period).
- Extreme snow or ice: The potential for conditions preventing access to the plant will require consideration of measures to ensure operator presence at the facility.
- Volcanic ash conditions: Affects machinery and could limit access to the plant.

Although a formal risk assessment will be required to quantify the risks posed by these postulated external events, it is expected that the safety design of the 4S NPF could accommodate their occurrence because of the capability to provide core cooling with natural circulation in the absence of offsite power and without operator intervention.

3. Probability of core damage/vessel failure

Analysis of the INL EBR-II, a sodium cooled reactor, indicates a core damage frequency one to two orders of magnitude below that calculated for current generation large commercial reactors.²² Until the results of a 4S-specific PRA are available, the EBR II analyses could be reviewed for potential applicability to the 4S.

The 4S NPF design accommodates design basis and beyond design basis events without sustaining fuel failure. Events that cause sodium explosions because of sodium-water reactions would, however, have to be postulated and evaluated for probability of occurrence. Such reactions are improbable because, among other design features, concrete barriers will have a lining to prevent ground water intrusion. A potential mechanism for such a reaction could be postulated to occur if the double walled interface in the steam generator is assumed to fail (a very low probability event).

Any explosion that is postulated to cause core damage would have to be sufficient to cause a breach of containment and allow a sodium fire that burns until the core is uncovered.

4. Probability and Consequences of Containment Failure

Maintaining the integrity of the containment function remains a vital NRC regulatory requirement, regardless of reactor design.²³ Accordingly, there is a need to demonstrate the

²² “Safety and Security Features of 4S Type Reactors”, D. Wade (Argonne) presentation to Alaska Rural Energy Conference, Sept 20-22, 2005, Sht 18.

²³ In SECY-05-0006, the Staff states: “Of the four options evaluated, the current staff position endorses Option 3:

The containment must adequately reduce radionuclide releases to the environs to meet the onsite and offsite radionuclide dose acceptance criteria for the events selected for the event categories and have the capability to establish controlled leakage and controlled release of delayed accident source term radionuclides.

Resolution of this issue will also establish a key element of the policy description of defense-in-depth. Option 3 requires that the containment have an independent capability to reduce delayed radionuclide releases to the environment independent of other radionuclide transport barriers associated with the fuel, core, and reactor coolant pressure boundary. This is consistent with the Commission’s defense-in-depth

containment effectiveness as a radionuclide barrier; a typical means of doing so is to evaluate the conditional containment failure probability (CCFP). The CCFP represents the probability of a release given core damage. Traditionally, containment failure modes that require evaluation can be grouped into three categories: those that involve failure at the time of core damage, those involving early and potentially energetic failure, and those with longer term failure.

Traditional mechanisms for energetic containment failure include steam and hydrogen explosion as well as direct containment heating. Steam and hydrogen explosions, however, do not occur in the 4S design because of the use of sodium coolant. Likewise, direct containment heating is a phenomenon that is associated with reactor vessel failure at high pressure; this phenomenon appears unrealistic given the 4S low system operating pressure.

Sodium-water reactions will require evaluation for the 4S design, although as discussed above the probability of such reactions is anticipated to be low. There are no water systems inside the 4S containment, so sodium-water interactions inside containment could only be caused by water intrusion from some unlikely and unanticipated condition. To ensure safety, the plant has been designed with multiple containment barriers that protect against such water intrusion scenarios. Protection against groundwater intrusion is a fundamental aspect of the plant design, and will be addressed comprehensively. The intermediate sodium circulating through the intermediate heat transport loop and the steam generator does not come into direct contact with the radioactive core or the radioactive primary sodium. Rather, the intermediate sodium extracts heat from the primary heat transport system via the intermediate heat exchangers and, as such, remains free of corrosion products. The intermediate sodium does not add to the risk of radioactive release in the event of a sodium-water interaction. The steam generator tubes are constructed of double walled tubing and integrated with a leak detection system to minimize the potential for sodium-water interactions. In the unlikely event of an sodium-water interaction in the steam generator, the SHTS is designed so that the rise in pressure caused by high temperature water intrusion and the ensuing chemical reactions will cause the SHTS dump tank valves to open, releasing the SHTS sodium into the inerted dump tanks. The dump tanks are designed to handle the sodium and reaction by-products.

Barring an explosion, the only mechanisms for containment releases are the bypassing of containment (i.e. containment failure at the time of core damage) or containment overpressurization caused by inadequate cooling of containment (i.e., longer term failure). As indicated earlier, containment overpressurization is very unlikely and is beyond the design basis. Containment leakage conditions are also less likely in a 4S than in current generation light water reactors because there are fewer active systems (thus fewer penetrations).

Because there is no probable cause of an energetic release to significantly pressurize containment, the energy associated with a release path is low and likely to result in a small plume exposure pathway.

safety philosophy that safety functions (e.g., control of fission product release) should not depend on a single element of design, construction, maintenance, or operation.

SECY-05-006, "Second Status Paper on the Staff's Proposed Regulatory Structure for New Plants and Update on Policy Issues Related to New Plant Licensing" (Jan. 7, 2005).

5. Timing of Releases

Liquid sodium is a coolant with excellent heat absorption capacity, very high thermal conductivity, low operating pressure (basically atmospheric), and superb natural convection capability. The heat capacity of liquid sodium provides a large heat sink for the core. Thus, the time required to heat the liquid surrounding the core to dangerous levels is substantially longer than for water cooled reactors, and the available time for responding to accidents is thus significantly increased.

The timeframe of potential releases should be determined to establish the range of required emergency response actions. Current advanced designs for large power reactors demonstrate that releases will not occur for at least 24 hours in the absence of operator intervention or active safety systems. For comparison purposes, it should be possible to demonstrate a longer release time for the 4S NPF. Analyses performed for the PRISM design indicated that for all but the most energetic release categories, the time to guard vessel/containment dome failure exceeds 24 hours.²⁴ Given the lower power level associated with the 4S design and the long time required to heat the sodium coolant to dangerous levels, it is anticipated that credible release scenarios would require an even longer time for releases to occur and would provide sufficient time for any emergency response actions that might be required.²⁵

D. POTENTIAL REQUESTS FOR EXEMPTIONS FROM REGULATORY REQUIREMENTS

Under 10 C.F.R. § 50.12, the NRC may grant nuclear power plant licensees an exemption from otherwise applicable regulatory requirements upon determining that (1) the requested exemption is “authorized by law, will not present an undue risk to public health and safety, and [is] consistent with the common defense and security,”²⁶ and (2) “special circumstances are present” that warrant the granting of the exemption.²⁷ The regulation identifies the “special circumstances” or justifications for which an exemption may be granted.²⁸

It is not anticipated that Galena would need to seek exemptions in order to be able to license the containment design for a 4S NPF.

V CONCLUSIONS

The containment requirements for the 4S NPF were compared to the NRC functional area requirements. While a formal probabilistic risk assessment for the 4S NPF has yet to be issued, the calculated probability of a significant release from the Galena 4S NPF based power facility and potential for offsite dose consequences can be expected to be lower than those for both

²⁴ PRISM Preliminary Probabilistic Risk Assessment, Amendment 8, Appendix A, Section A4.

²⁵ It will be necessary to demonstrate that releases to the groundwater and the ingestion pathway do not require emergency response actions at Galena.

²⁶ 10 C.F.R. § 50.12(a)(1).

²⁷ 10 C.F.R. § 50.12(a)(2).

²⁸ 10 C.F.R. § 50.12(a)(2)(i)-(vi).

advanced reactor designs and current generation reactors. The reasons for this are:

- The simple, passive features in the Galena reactor should result in a lower calculated probability of core damage than current generation plants.
- The capability of the containment structure and its passive nature cooling capability provides a reliable barrier to release.
- The radionuclide inventory is approximately two orders of magnitude less than that used in the advanced designs undergoing certification.

VI RECOMMENDATIONS

Galena should engage in early discussions with the NRC Staff to confirm the acceptability of the engineered safety feature and containment system design. These determinations would be of vital importance in the plant design, operating costs and the overall economic viability of the project.