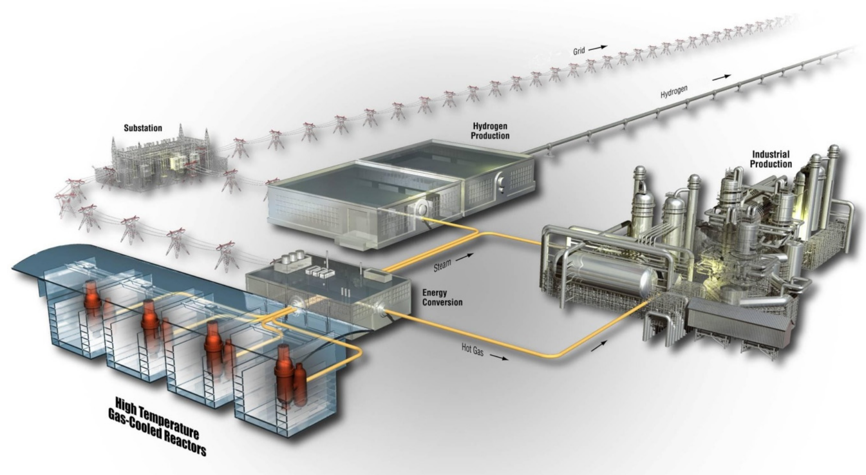
INL/EXT-11-21605

NGNP Nuclear-Industrial Facility and Design Certification Boundaries

July 2011



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July 2011

Idaho National Laboratory
Next Generation Nuclear Plant Project
Idaho Falls, Idaho 83415

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Next Generation Nuclear Plant Project

NGNP Nuclear-Industrial Facility and Design Certification Boundaries

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July 2011

Approved by:

Thomas Hicks
NGNP Regulatory Affairs

7/20/11
Date

Jim Kinsey
NGNP Regulatory Affairs Director

7-21-11
Date

Greg Gibbs
NGNP Project Director

7/22/11
Date

ABSTRACT

The Next Generation Nuclear Plant (NGNP) Project was formally established by the Energy Policy Act of 2005, designated as Public Law 109-58, 42 USC 16021, and is based on research and development activities supported by the Generation IV nuclear technology. The mission of the NGNP Project is to support commercialization of the high temperature gas-cooled reactor (HTGR) technology. The HTGR is helium cooled and graphite moderated and can operate at reactor outlet temperatures much higher than those of conventional light water reactor technologies. Accordingly, it can be applied in many industrial applications as a substitute for burning fossil fuels, such as natural gas, in addition to producing electricity, which is the principal application of current light water reactors. These varied industrial applications may involve a standard HTGR modular design in combination with different site-specific Energy Conversion Systems. Some of these process heat applications will require process heat delivery systems to lie partially outside the HTGR operator's facility. Energy Conversion Systems are conventional, non-nuclear equipment and buildings.

Given these varied applications there should be a clear understanding between the HTGR applicant and the Nuclear Regulatory Commission (NRC) regarding the demarcation between those systems that are within the nuclear facility and under the regulatory jurisdiction of the NRC (within the scope of a 10 CFR Part 52 design certification and a combined license) and those that fall outside the scope of the NRC (industrial facility). Additionally, it is important to have a clear understanding regarding the plant scope that should be addressed in an HTGR facility Part 52, design certification application and the part of the plant scope that could be addressed as part of a site specific combined license application.

CONTENTS

ACRONYMS

AOO	Anticipated Operational Occurrences
ASLB	Atomic Safety and Licensing Board
BDBE	Beyond Design Basis Events
BOP	balance of plant
CFS	condensate and feedwater system
COL	Combined Operating License
COLA	COL application
CP	construction permit
DBE	Design Basis Events
DC	design certifications
DCD	Design Certification Document
DOE	Department of Energy
ECA	Energy Conversion Area
ECS	Energy Conversion System
ESF	Engineered Safety Features
FSAR	final safety analysis report
GDC	General Design Criterions
HTGR	high temperature gas-cooled reactor
LWR	light water reactor
MC	main condensers
MHTGR	Modular High Temperature Gas-cooled Reactor
MSIV	main steam isolation valve
MSSS	main steam supply system
NGNP	Next Generation Nuclear Plant
NHSS	nuclear heat supply system
NI	nuclear island
NRC	Nuclear Regulatory Commission
NSSS	nuclear steam supply system
PWR	pressurized water reactor
SAFR	Sodium Advanced Fast Reactor
SAR	safety analysis report
SBWR	Simplified Boiling Water Reactor
SMR	small modular reactor

SRM	standard reactor module
SRP	Standard Review Plan
SSAR	Standard Safety Analysis Report
SSC	structures, systems, and components

1. INTRODUCTION

The modular high temperature gas reactor (HTGR) technology can be used to supply energy in several forms to a wide range of applications. The HTGR-Nuclear Heat Supply System (NHSS), comprised of the reactor (including fuel, graphite moderator, control and safety rods, reflector, and support structure), pressure vessels, support systems and primary helium heat transport circuit, uses the nuclear energy to heat helium to high temperatures (725 to 950°C reactor outlet temperature). For each application, an Energy Conversion System converts the energy in the high temperature helium to the form(s) required to meet the energy needs of the application. Applications identified to date include supplying: electricity to the grid; co-generation of steam; electricity and/or high temperature heat to industrial facilities; process heat and electricity for hydrogen production; steam or other high temperature fluid for bitumen recovery from oil sands or enhanced oil recovery from oil shale; process heat, steam and electricity for petro-chemical and refining processes; and process heat for conversion of coal to synthetic transportation fuels and hydrocarbon feedstock.

Designs of the HTGR-NHSS module developed to-date in the NGNP Project include prismatic and pebble bed reactor concepts with ratings between 200 and 625 MWth and reactor outlet temperatures in the range of 725 to 950°C. The varying forms of energy demands require that the HTGR-NHSS modules support different Energy Conversion System configurations and that multiple NHSS modules be provided to meet full power and plant availability requirements.

Given that the modular HTGR configuration will be different than the current fleet of licensed reactors, there should be a clear understanding between the HTGR applicant and the Nuclear Regulatory Commission (NRC) regarding the demarcation between those systems that are within the nuclear facility and under the regulatory jurisdiction of the NRC (within the scope of a 10 CFR Part 52 design certification [DC] and a combined operating license [COL]) and those that fall outside the scope of the NRC (industrial facility). Additionally, it is important to have a clear understanding regarding the plant scope that should be addressed in an HTGR facility Part 52, DC application and the part of the plant scope that could be addressed as part of a site specific COL application. Although areas such as the control room, the radwaste facility, and reactor service building would be included within the nuclear facility boundary, they may be excluded from a standard design certification, along with the typical secondary side design. Systems within the nuclear facility that are not included within a DC application would be reviewed by the NRC as part of the site specific COL application review.

Figure 1 summarizes examples of HTGR configurations that have been evaluated to meet the potential end-user needs identified to-date.

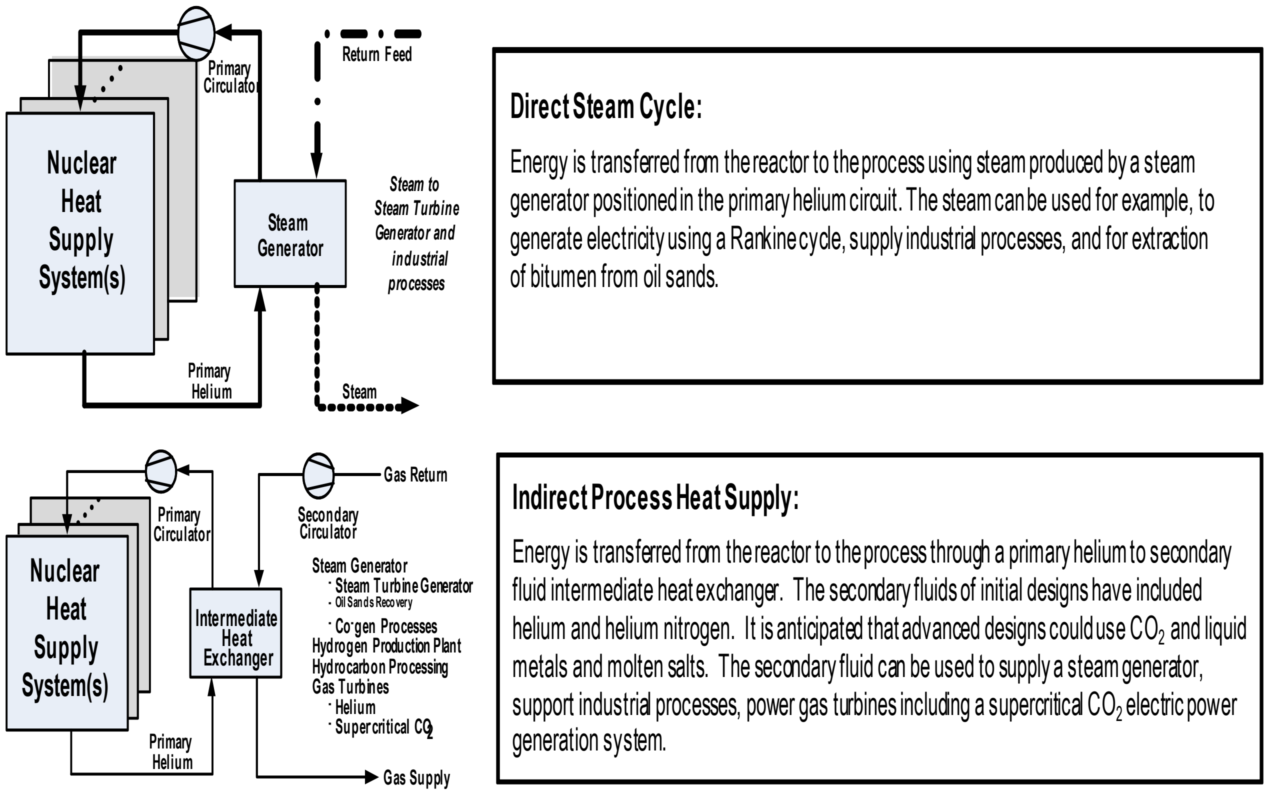

Figure 1. Example HTGR configurations.

Figure 2, adapted from the General Atomics NGNP Conceptual Design Report (Ref 1), illustrates a typical single module HTGR plant arrangement with an onsite[a] turbine generator for electric power generation and process heat transfer lines running to an offsite location. The configuration illustrated in this figure is useful in understanding the need for defining boundaries between onsite and offsite systems, but represents only one of many possible HTGR configurations.

[a] The use of the terms "onsite" and "offsite" refer to inside or outside of the HTGR protected area which coincides with inside or outside of the nuclear facility boundary.

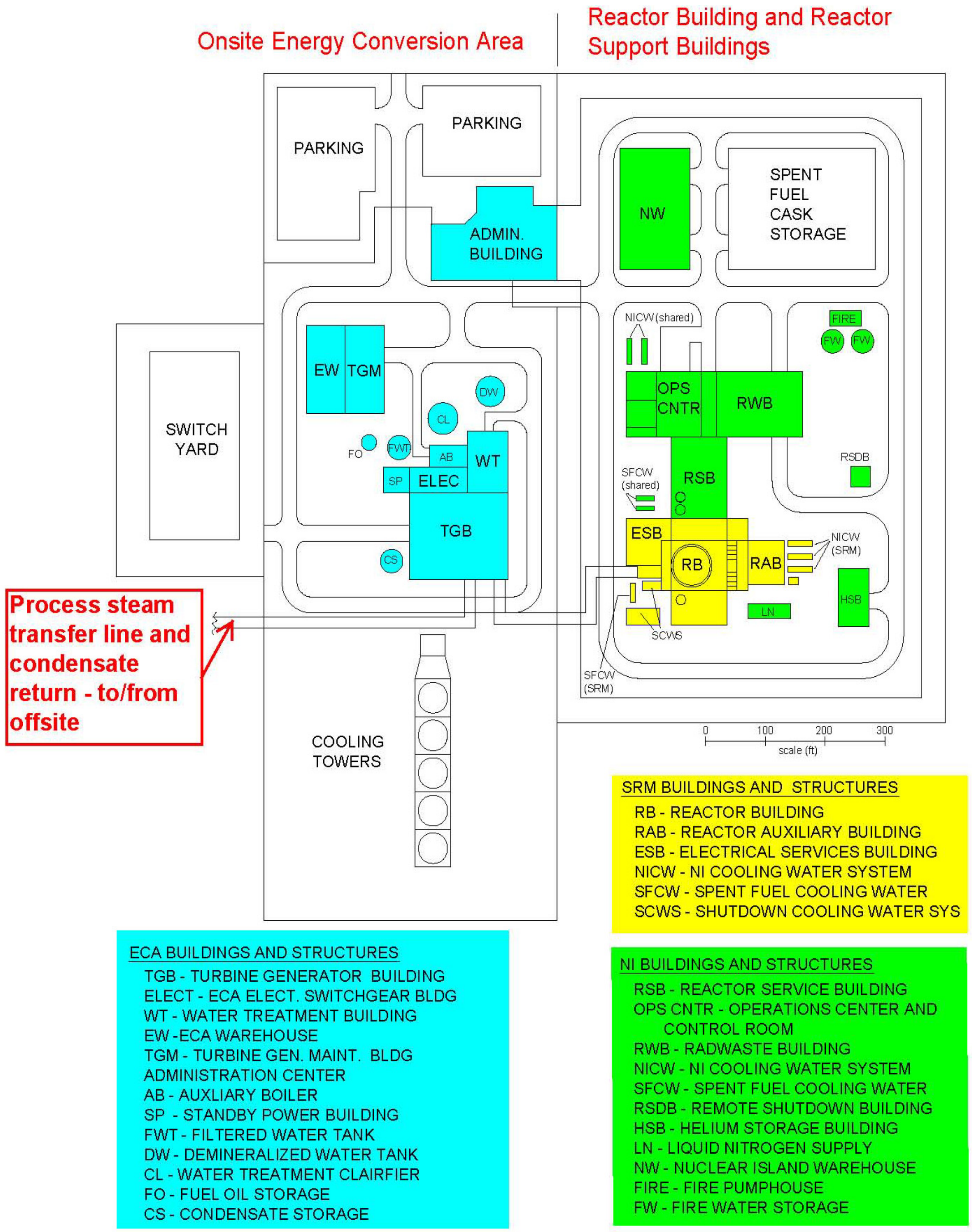

Figure 2. Typical HTGR plant general arrangement.[b]

1.1 Purpose

Because the HTGR modular design can support different commercial applications, the site-specific designs that include the Energy Conversion Systems can vary widely between applications. Given these

b. "SRM" means "standard reactor module." The SRM is the part of the facility that would be certified under a design certification process. The "NI" is the nuclear island that includes many of the nuclear plant support systems. The "ECA" is the energy conversion area that includes the onsite energy conversion system. Other possible configurations could include multiple modules.

varied applications and the need to not limit configuration options in a DC application, this paper defines two sets of regulatory boundaries in order to properly structure the possible HTGR licensing approaches and streamline the licensing process. Since most of the anticipated HTGR applications involve the delivery of process heat to an offsite commercial customer, the first boundary needs to be defined between the HTGR plant which is within the scope of NRC regulatory jurisdiction, or the “nuclear facility,” and the “industrial facility” that is outside the NRC regulatory scope.

A second boundary definition would subdivide the nuclear facility portion of the HTGR plant to address the scope of the HTGR systems that would need to be part of the certified portion of a DC application and those that would be more fully described in a site specific COL application that references the DC.[c] It is expected to be advantageous to obtain a DC for the standard part of the HTGR so that NRC regulatory issues can be resolved one time and facilitate future HTGR commercial deployment using different Energy Conversion Systems, such as turbine generators for electric power production or process steam delivery system equipment. The DC for a HTGR standard reactor module could be structured for application in a multi-module configuration to facilitate development of the COLA and licensing of such configurations in industrial applications.

In summary, there should be a clear understanding between the HTGR applicant and the NRC regarding the boundary between the nuclear facility, under the regulatory jurisdiction of the NRC (i.e., within the scope of the DC and COL applications), and the interfaces to an end user facility that would fall outside the scope of the NRC (industrial facility). A clear description is also needed to define the plant scope to be addressed in an HTGR DC.

1.2 Objective of this Paper

The objectives of this paper are to communicate the NGNP Project’s position to, and receive concurrence from, NRC staff regarding:

1. A definition of a boundary between the HTGR nuclear facility and the industrial facility with respect to regulatory jurisdiction.
2. A description of the typical nuclear facility design requirements and interface requirements that would need to be defined to ensure safe operations for an interconnection to an industrial facility.
3. A description of a minimum set of HTGR nuclear facility systems and interface requirements that should be addressed within the scope of the certified portion of a 10 CFR Part 52, DC and those that may be appropriately described in a site-specific Part 52 COL application.

1.3 Scope

The scope of this paper is to define two sets of regulatory boundaries for the HTGR:

- The boundary between the nuclear facility, i.e., those systems under NRC regulatory jurisdiction; and the industrial facility, i.e., those systems that fall outside NRC regulatory jurisdiction
- The boundary between the minimum set of plant systems that should be addressed in a 10 CFR Part 52 DC application and those that could fall outside of the DC scope and within the COL application scope.

Since the potential impact of onsite hazards and nearby industrial facility hazards, such as chemical toxicity, or explosion, on the HTGR nuclear facility are required to be analyzed by the HTGR safety analyses per existing regulatory requirements, and no clarification to these requirements is being sought in this paper, this topic area is not within the scope of this paper.

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c. It is noted that the NGNP Licensing Plan (Ref 2) specifies that the preferred regulatory approach for the HTGR demonstration plant should be a Part 52 COL application that does not reference a DC. However, a DC is recognized as an option for any HTGR vendor, and would likely be utilized for future HTGR commercial deployments.

1.4 Summary of Outcome Objectives

The NGNP Project is seeking NRC's general concurrence and/or comments regarding an approach to defining a boundary between the HTGR nuclear facility, under the regulatory jurisdiction of the NRC (within the scope of the DC and COL applications), and an end user facility that would fall outside the scope of the NRC (industrial facility), as well as defining the plant scope to be addressed in an HTGR DC. The NGNP Project is seeking agreement on the following specific aspects of its approach to define these boundaries:

1. Since the NRC clearly has regulatory jurisdiction over plant facilities that are required to be protected under physical security regulations, structures, systems, and components (SSCs) needed to be within the plant's security boundary would be part of the nuclear facility.
2. All SSCs that perform safety-related or risk significant functions for the HTGR would be within the nuclear facility boundary.
3. An energy conversion system located within the HTGR protected area boundary that is integral to the HTGR facility and controlled by the HTGR control room would be considered within the nuclear facility. An energy conversion system can be excluded from the nuclear facility scope if it is located outside the protected area boundary and is separated from the HTGR facility by a transfer system with interface criteria that act to ensure that the HTGR facility is not dependent on, or adversely affected by, events that occur within the separate industrial facility.
4. Regardless of whether the energy conversion system site is within the nuclear facility or the separate industrial facility, analyses would be required with respect to potential missiles, security issues, flooding issues, or other impacts to HTGR SSCs that perform a safety function.
5. Beyond the criteria defined in Items 1 through 4 above, the boundary between the HTGR nuclear facility and the industrial facility with respect to regulatory jurisdiction can be defined by properly describing the nuclear facility system design, the transfer system design, and interfaces, using an appropriate set of conceptual design information and interface requirements. The following represent an appropriate set of high level design and interface requirements for this boundary[d]:
    a. Failures or transients within the industrial facility portion of the transfer system[e] would not preclude safety-related portions of the nuclear facility from functioning as required during normal operations, anticipated operational occurrences, and accident conditions.
    b. Nuclear facility plant system transients caused by industrial facility systems or the electrical transmission grid would be limited (in frequency and severity) and analyzed in the plant's safety analyses, similar to the way transmission grid disturbances are evaluated in existing light water reactors (LWRs).
    c. No portion of the transfer system within the scope of the industrial facility would be required to perform any safety or safe shutdown function, or be relied upon as a supporting system to a safety-related system.
    d. The transfer system would have monitoring capability to detect disturbances and, if required by the HTGR safety analysis, isolate the industrial facility system during transients and accidents.
    e. Releases of radioactive material from the transfer system would meet required limits. Monitoring and/or sampling would be performed, as necessary, to ensure limits are met.

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d. Any interface with the industrial facility would involve a transfer system that would provide steam or process heat to the customer and return condensate or makeup fluid to the nuclear facility.

e. The term "transfer system" as used in this paper refers to the process lines that transfers steam or process heat from the secondary side of the HTGR steam generator or intermediate heat exchanger to the end-user facility and provides return condensate or makeup fluid to the nuclear facility.

6. No specific system descriptive information would be needed for the DC/COL application for the scope of the plant outside the nuclear facility, since this part of the plant would be outside the scope of the NRC review. Instead, the COL application would demonstrate how interface requirements specified in either the COL application or DC would be met by the specific industrial facility interface.
7. The HTGR nuclear facility can be further divided into systems that should be addressed within a 10 CFR Part 52 DC and those that may be appropriately described in a site-specific Part 52 COL application. The DC application would provide, as necessary to address the degree of flexibility desired by the applicant regarding the deployment of the HTGR, a description and analyses of the possible operating configurations of the reactor modules with common systems, interface requirements, and system interactions, accounting for differences among the configurations, including any restrictions that will be necessary during the construction and startup of a given module to ensure the safe operation of any module already operating. At a minimum, the SSCs within the scope of an HTGR DC should include:
    a. The reactor pressure vessel, including core support structure, fuel, reactivity control system, and graphite matrix
    b. The cross-vessel(s)
    c. The primary-to-secondary heat transfer pressure vessel(s)
    d. Piping connecting the primary helium circuit to support systems, (e.g., shutdown cooling system, primary helium service and purification system)
    e. Support systems: shutdown cooling system, primary helium service and purification system
    f. NHSS protective system
    g. Secondary piping penetrating the primary-to-secondary heat transfer pressure vessel(s) up to, if applicable, the isolation valves
    h. If required by the HTGR safety analysis, the isolation valves in secondary steam and/or high temperature fluid supply to and return from the Energy Conversion Systems
    i. Reactor building
    j. New and spent fuel handling systems within the reactor building.
8. Conceptual design information and interface requirements should be provided in the HTGR DC application, as appropriate, for SSCs not within the scope of the DC. The interface requirements would address [f]:
    a. Requirements for safe operation of the standard design that must be satisfied by matching portions of the site-specific design
    b. Site-related design assumptions upon which the standard design is based
    c. Criteria pertinent to the standard design described in the DC application that may be useful for the design and review of matching systems, components, and structures (within the standard design, safety criteria for the items including codes and standards, Principal Design Criteria, and regulatory guides)
    d. Requirements to preserve the following specific HTGR functions[g]:

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f. These interface requirements for site specific SSCs are beyond those specified for the nuclear – industrial facility boundary.

g. Functions from NGNP white paper, INL/EXT-10-19509, “Structures, Systems, and Components Safety Classification,” dated September 2010.

(1) Prevent or mitigate the consequences of Design Basis Events (DBEs) to comply with the 10 CFR §50.34 offsite dose limits

(2) Prevent the frequency of Beyond Design Basis Events (BDBEs) with consequences greater than the 10 CFR §50.34 offsite dose limits from increasing into the DBE region

(3) Mitigate the consequences of Anticipated Operational Occurrences (AOOs) to comply with the offsite dose limits of 10 CFR Part 20

(4) Prevent the frequency of DBEs with consequences greater than the 10 CFR Part 20 offsite dose limits from increasing into the AOO region.

9. A site specific COL application that references a DC would provide site specific design information for all areas addressed as conceptual design in the applicable DC including the Energy Conversion System. The COL application would also need to provide information demonstrating that the site specific design satisfied the interface requirements in the DC.

10. The applicant for a COL application that does not reference a DC will need to submit design information on all SSCs within the nuclear facility and should not include any conceptual design information for the facility in the application. This COL application would need to describe the nuclear – industrial facility boundary interface requirements (see item 4 above) and show that they are satisfied by the site-specific design.

1.5 Relationship to Other NGNP Pre-licensing Topics/Papers

Functions identified in the interface criteria evaluation were extracted from the NGNP white paper, INL/EXT-10-19509, “Structures, Systems, and Components Safety Classification,” dated September 2010 (Reference 15).

NGNP white paper, INL/EXT-10-18178, “License Structure for Multi-Module Facilities, “ describes the NGNP positions regarding whether a multi-module reactor plant can be licensed with a single NRC review, hearing, and safety evaluation report. This paper further explains the structure and the duration of such a license (Reference 16).

2. REGULATORY FOUNDATION

2.1 U.S. Regulatory Foundation for the Nuclear-Industrial Facility and Design Certification Boundaries

2.1.1 NRC Requirements

In 1989, the NRC published the final rule, 10 CFR 52, "Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Reactors" (Ref 3). The new rule set out the review procedures and licensing requirements for applications for these new licenses and certifications. The rule was modified in 2007, to clarify the applicability of various requirements to each of the licensing processes by making necessary conforming amendments throughout the NRC's regulations to enhance the NRC's regulatory effectiveness and efficiency in implementing its licensing and approval processes. The DOE and NRC jointly determined that the HTGR will be licensed under Part 52. Hence, the boundary evaluations presented in this paper are presented in the context of a Part 52 licensing process.

In determining how and where to define the proper HTGR boundary between the nuclear facility and industrial facility, and how to define the boundary between a DC application and a COL application, it is important to identify applicable NRC regulations and guidance that specify the expectations for the two application types.

10 CFR 52, Subpart B, "Standard Design Certifications"

Subpart B of 10 CFR Part 52 defines the regulatory requirements for design certification applications.[h] Section 52.47, "Contents of Applications; Technical Information," defines the requirements for technical content of a design certification application. Because the contents of design certification applications, including inspection, test, analysis, and acceptance criteria (ITAAC)[i], are certified by rulemaking, it is not practical to include optional configurations and equipment as part of the certified portion of the plant. The regulations make provisions for design certifications to include optional configurations by allowing these applications to include "conceptual design" information[j]. Paragraph 52.47(a) states general requirements for the DC Final Safety Analysis Report (FSAR):

> *(a) The application must contain a final safety analysis report (FSAR) that describes the facility, presents the design bases and the limits on its operation, and presents a safety analysis of the structures, systems, and components and of the facility as a whole, and must include the following information:*
>
> > *(1) The site parameters postulated for the design, and an analysis and evaluation of the design in terms of those site parameters;*
> >
> > *(2) A description and analysis of the structures, systems, and components (SSCs) of the facility, with emphasis upon performance requirements, the bases, with technical justification therefor, upon which these requirements have been established, and the evaluations required to show that safety functions will be*

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h. A standard design certification from the NRC is submitted separately from an application for a COL filed under Subpart C of Part 52 for a nuclear power facility. An applicant for a COL may reference a standard design certification.

i. ITAAC provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a facility that incorporates the design certification has been constructed and will be operated in conformity with the design certification, the provisions of the Act, and the Commission's rules and regulations. All ITAAC in the design certification must be verified as complete before fuel load is authorized by the NRC.

j. NRC's use of the term conceptual has a different context than citing the status of design development as conceptual. In the NRC's context the energy system configurations and performance would be conceptualized to support defining interfaces, transients, and accident conditions for which the NHSS is certified. The conceptualized energy conversion systems would not be included in the certification.

> *accomplished. It is expected that the standard plant will reflect through its design, construction, and operation an extremely low probability for accidents that could result in the release of significant quantities of radioactive fission products. The description shall be sufficient to permit understanding of the system designs and their relationship to the safety evaluations. Such items as the reactor core, reactor coolant system, instrumentation and control systems, electrical systems, containment system, other engineered safety features, auxiliary and emergency systems, power conversion systems, radioactive waste handling systems, and fuel handling systems shall be discussed* ***insofar as they are pertinent [emphasis added]****.*

Paragraph 52.47(a)(24) states that the design certification may include:

> *A representative conceptual design for those portions of the plant for which the application does not seek certification, to aid the NRC in its review of the FSAR and to permit assessment of the adequacy of the interface requirements in paragraph (a)(25) of this section;*

Paragraph 52.47(a)(25) requires that the DC application contain appropriate interface requirements, and states:

> *The interface requirements to be met by those portions of the plant for which the application does not seek certification. These requirements must be sufficiently detailed to allow completion of the FSAR;*

Paragraph (c) of 52.47, defines content requirements for design certification applications that have particular characteristics. Paragraph (c)(3) addresses modular reactors[k] and requires the following:

> *An application for certification of a modular nuclear power reactor design must describe and analyze the possible operating configurations of the reactor modules with common systems, interface requirements, and system interactions. The final safety analysis must also account for differences among the configurations, including any restrictions that will be necessary during the construction and startup of a given module to ensure the safe operation of any module already operating.*

In the statement of considerations for the final (1989) rule (Ref 3), the NRC stated that the Part 52 process "…provides for certification of advanced designs and permits certification of designs of less than full scope only in highly restricted circumstances."[l] Clearly, the NRC had intended that DC applications be a complete representation of the plant. The final rule's provisions on scope, see § 52.47, reflect a policy that certain designs, especially designs that are evolutions of light water designs now in operation, should not be certified unless they include all of a plant which can affect safe operation of the plant except its site-specific elements. The NRC provided examples of designs that are evolutions of currently operating light-water designs, including General Electric's ABWR, Westinghouse's SP/90, and Combustion Engineering's System 80+. NRC stated further that full-scope may also be required of certain advanced designs, namely, the passive light-water designs such as General Electric's Simplified Boiling Water Reactor (SBWR) and Westinghouse's AP600. NRC stated that considerations of safety, not market forces, constituted the basis for the final rule's requirement that these designs be full-scope designs. According to the staff, "…long experience with operating light water designs more than adequately demonstrates the adverse safety impact which portions of the balance of plant can have on the nuclear island. Given this experience, certification of these designs must be based on a consideration of the whole plant, or else the

k. Modular designs are defined in § 52.1. Modular plant designs are not portions of a single nuclear plant, rather they are separate nuclear power reactors with some shared or common systems.

l. Ref 3, at 54 FR 15373

certifications of those designs will lack that degree of finality which should be the mark of the certifications."[m]

However, the Commission stopped short of stating that no design of incomplete scope could ever be certified.

> *There is no reason to conclude that there could never be a design which protects the nuclear island against adverse effects caused by events in the balance of plant. The final rule therefore provides the opportunity for certification of designs of less than complete scope, if they belong to the class of advanced designs. See § 52.47(b)[1987 rule]. Examples of designs in this class include the passive light-water designs mentioned above and non-light-water designs such as General Electric's PRISM, Rockwell's SAFR, and General Atomic's MHTGR. But here too the rule sets a high standard: Certification of an advanced design of incomplete scope will be given only after a showing, using a full-scale prototype, that the balance of plant, cannot significantly affect the safe operation of the plant.*[n]

> *While analyses may be relied upon by the staff to demonstrate the acceptability of a particular safety feature which evolved from previous experience or to justify the acceptability of a scale model test, it is very unlikely that an advanced design would be certified solely on the basis of analyses. Prototype testing is likely to be required for certification of advanced non-light water designs because these revolutionary designs use innovative means to accomplish their safety functions, such as passive decay heat removal and reactivity control, which have not been licensed and operated in the United States.* [o]

Section 52.47(c)(2) [2007 rule] (Ref 4) requires applications for 'advanced' nuclear power plants to provide an essentially complete scope of design and meet the design qualification testing requirements in 10 CFR 50.43(e). Advanced designs differ significantly from evolutionary LWR designs or incorporate, to a greater extent than evolutionary designs do, simplified, inherent, passive, or other innovative means to accomplish their safety functions. Examples of advanced nuclear power plant designs listed in the rule include General Atomic's Modular High Temperature Gas-Cooled Reactor (MHTGR), General Electric's SBWR, and Westinghouse's AP600.

10 CFR 52, Subpart C, Combined Licenses

Under 10 CFR 52, Subpart C, *Combined Licenses*, the NRC specifies its requirements for technical information in the COL application final safety analyses report. Paragraph 52.79, *Contents of applications; technical information in final safety analysis report*, states:

> *(a) The application must contain a final safety analysis report that describes the facility, presents the design bases and the limits on its operation, and presents a safety analysis of the structures, systems, and components of the facility as a whole. The final safety analysis report shall include the following information, at a level of information sufficient to enable the Commission to reach a final conclusion on all safety matters that must be resolved by the Commission before issuance of a combined license:*

> *(2) A description and analysis of the structures, systems, and components of the facility with emphasis upon performance requirements, the bases, with technical justification therefore, upon which these requirements have been established, and the evaluations*

m. Ref 3, at 54 FR 15374

n A further discussion regarding prototype requirements for advanced reactors is provided in SOC for the final 2007 Part 52 rulemaking, 72 FR 49370.

o. 54 FR 15375

required to show that safety functions will be accomplished. It is expected that reactors will reflect through their design, construction, and operation an extremely low probability for accidents that could result in the release of significant quantities of radioactive fission products. The descriptions shall be sufficient to permit understanding of the system designs and their relationship to safety evaluations. Items such as the reactor core, reactor coolant system, instrumentation and control systems, electrical systems, containment system, other engineered safety features, auxiliary and emergency systems, power conversion systems, radioactive waste handling systems, and fuel handling systems shall be discussed insofar as they are pertinent. The following power reactor design characteristics and proposed operation will be taken into consideration by the Commission:

(i) Intended use of the reactor including the proposed maximum power level and the nature and inventory of contained radioactive materials;

(ii) The extent to which generally accepted engineering standards are applied to the design of the reactor;

(iii) The extent to which the reactor incorporates unique, unusual or enhanced safety features having a significant bearing on the probability or consequences of accidental release of radioactive materials;

(iv) The safety features that are to be engineered into the facility and those barriers that must be breached as a result of an accident before a release of radioactive material to the environment can occur. Special attention must be directed to plant design features intended to mitigate the radiological consequences of accidents. In performing this assessment, an applicant shall assume a fission product release from the core into the containment assuming that the facility is operated at the ultimate power level contemplated;

10 CFR 73, "Physical Protection of Plants and Materials"

10 CFR 73 defines, in part, 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 in which special nuclear material is used. Paragraph 73.1 requires, in part, that each licensee establish and maintain a physical protection system which will have capabilities for the protection of special nuclear material. The physical protection system shall be designed to protect against the design basis threats of theft or diversion of special nuclear material and radiological sabotage as stated in § 73.1(a).

10 CFR 73.46 requires, in part, that vital equipment must be located only within a vital area, and strategic special nuclear material must be stored or processed only in a material access area. Both vital areas and material access areas must be located within a protected area so that access to vital equipment and to strategic special nuclear material requires passage through at least three physical barriers. Vital area means any area which contains vital equipment. Vital equipment means any equipment, system, device, or material, the failure, destruction, or release of which could directly or indirectly endanger the public health and safety by exposure to radiation. Equipment or systems which would be required to function to protect public health and safety following such failure, destruction, or release are also considered to be vital.

10 CFR 73.55 defines requirements for physical protection of licensed activities in nuclear power reactors against radiological sabotage. The licensee is required to

"…establish and maintain an onsite physical protection system and security organization which will have as its objective to provide high assurance that activities involving special nuclear material are not inimical to the common defense and security and do not

> *constitute an unreasonable risk to the public health and safety. The physical protection system shall be designed to protect against the design basis threat of radiological sabotage as stated in § 73.1(a). "*

To achieve this general performance objective, the onsite physical protection system and security organization must include, but not necessarily be limited to, the capabilities to meet the specific requirements such as physical barriers, access restrictions, detection aids, and communications requirements.

These NRC security regulations help define the boundary of the nuclear facility in that any equipment within the security boundary would be governed by these regulations and, thus, would be required to be within the nuclear facility.

2.1.2 NRC Policy Statements

In SECY-88-202, "Standardization of Advanced Reactor Designs,"[p] the staff presented a set of criteria that they developed for use in the review of DOE's plans for standardization of three advanced reactor concepts. Two of the issues addressed in this paper were (1) scope and level of detail of design to be standardized and (2) plant options (number of reactor modules) to be standardized. The staff's proposed criteria for resolution of these issues were developed to be consistent with the intent of the Commission's policies on standardization and advanced reactors. The criteria were consistent with the staff's proposed rulemaking on Standard Design Certifications (10 CFR 52).

In the SECY paper, the staff listed four reasons that were given by the reactor designers for limiting the certified portion of the designs:

1. They contend that all of the plant's safety systems will be contained within the certified envelope (with no systems interactions between safety and non-safety portions of the plant capable of affecting performance of the plant's safety functions). This, it is proposed, eliminates the need for NRC to approve anything other than interface requirements for the remainder of the design.
2. They are concerned that if the non-safety portion of the design were certified, NRC would get involved in design and construction verification to a greater extent than is necessary.
3. They argue that not certifying the entire plant will allow greater flexibility to incorporate design improvements or improvements in technology without having to go through the process of amending the Design Certification.
4. They state that in order to allow utilities the flexibility of procuring the balance of plant in a competitive fashion with design differences to suit their needs, Design Certification of the entire plant is not desired.

The staff notes in that paper:

> *…the major contributors to non-standardized plants today are the differences from plant to plant external to the Nuclear Steam Supply System (NSSS). Problems external to the NSSS have been the initiator of many plant shutdowns, the focus of many Generic Safety Issues and have impacted plant safety. However, transients initiated in the non-safety related portions of the advanced designs should have less likelihood of leading to severe accidents. This is because the passive reactor shutdown and decay heat removal systems have the potential for high reliability since they are less vulnerable to failure modes involving active equipment, electric power, or human error. Therefore, even though failures or transients in the balance of plant could challenge safety systems, the overall risk from these challenges should be less than for LWRS. However, since the design and*

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p. Note that SECY-86-368, *NRC Activities Related to the Commission's Policy on the Regulation of Advanced Nuclear Power Plants,* was a predecessor document to SECY-88-202.

operation of the remainder of the plant is key to ensuring that the interface criteria with safety systems are met, that assumptions regarding accident initiators are maintained, and that operating experience gained on one plant is readily transferable to other plants, submittal of the entire plant for Design Certification is still preferred. This would eliminate the possibility of each plant varying substantially from the others, would make the preparation of a PRA and safety analysis more straight-forward and would minimize the time and staff resources required to review individual license applications to assess compliance with interface criteria. In addition, approval of a complete plant design at the Design Certification stage will afford a greater opportunity for wide public participation, as well as reducing the time and resources expended in repeatedly litigating the acceptability of a design at individual hearings.

In short, the benefits to the Commission from standardization are maximized when the entire plant is certified. For these reasons, the staff preference is to standardize and certify the entire plant. However, from the standpoint of performing a technical review, the staff could consider Design Certification of less than the complete plant provided that the certified portion of the plant contains all of the safety systems and the following criteria are met for the non-certified portion:

1. *The interface requirements established for the non-certified portions of the design are sufficiently detailed to allow completion of a final safety analysis and a PRA for the plant.*
2. *Compliance with the interface requirements established for the noncertified portions of the design is verifiable through inspection, testing (separately or in the plant), previous experience or analysis. Compliance with interface requirements dealing with reliability of components or systems shall be verifiable through previous experience or testing.*
3. *A representative design for the non-certified portions of the plant is submitted along with the application for Design Certification as an illustration of how the interface requirements can be met and as an aid in the review of the PRA and safety analysis.*

The above criteria would require certification of all the safety related portions of the plant and sufficient information on the other portions to determine overall safety. The staff would also require that the level of design detail submitted for the certified portion be final design information, equivalent to that provided in order to obtain an FDA. These criteria would ensure that the plant will be built and operated consistent with its safety analysis and PRA. Since the advanced designs are proposing balance of plant systems that are not safety related, the design flexibility desired by the designers would be retained for a large portion of the plant. The acceptability of the three DOE sponsored advanced reactor concepts with regard to scope and level of detail will be addressed in the respective SERS.

A review of the SERs referenced in SECY-88-0202 did not identify any relevant discussion regarding the topic of this paper.

In SECY-10-0034, "Potential Policy, Licensing, And Key Technical Issues for Small Modular Nuclear Reactor Designs," (Ref 6) the staff identified a number of potential policy and licensing issues that may require resolution during the review of design and license applications for some of these designs. In general, these issues result from the key differences between the new designs and current-generation LWRs (such as size, moderator, coolant, fuel design, and projected operational parameters), but they also result from industry-proposed review approaches and modifications to current policies and practices.

One of the issues discussed, item 4.4, "Industrial Facilities Using Nuclear-Generated Process Heat," identified potential policy and licensing issues for those facilities used to provide process heat for industrial applications. In this paper the staff stated:

> *The close coupling of the nuclear and process facilities raises concerns involving interface requirements and regulatory jurisdiction issues. Effects of the reactor on the commercial product of the industrial facility during normal operation must also be considered. For example, tritium could migrate to a hydrogen production facility and become a byproduct component of the hydrogen product. Resolution of these issues will require interfacing with other government agencies and may require Commission input to determine whether the design and ultimate use of the product is acceptable.*
>
> *This issue is applicable to license applications for new, first-of-a-kind SMR designs, including the NGNP. However, the staff believes that resolution for this issue need not occur until after a license application is submitted because it concerns site-specific issues associated with the staff's review of an operating license. Once a license application is received, the NRC staff will review how the nuclear facility is connected to the industrial facility, consider the interrelationship between the staffs of both facility, consider white papers or topical reports concerning this issue that it receives from DOE and potential SMR applicants, discuss design-specific proposals to address this matter, and review similar activities with nuclear and non-nuclear facilities. Should it be necessary, the staff will propose changes to existing regulatory guidance or new guidance concerning the effect of the industrial facility on the nuclear facility in a timeframe consistent with the licensing schedule.*

2.1.3 NRC Guidance

NUREG-0800, "Standard Review Plan (SRP) for the Review of Safety Analysis Reports for Nuclear Power Plants" (Ref 7)

NUREG-0800 provides guidance to the NRC staff in performing safety reviews of various types of license applications including DC applications and COL applications under 10 CFR Part 52. Implementation of the criteria and guidelines contained in the SRP by staff members in their review of applications provides assurance that a given design will comply with NRC regulations and provide adequate protection of the public health and safety.

As described in NUREG-0800, designs of SSCs that are to be addressed in Part 52 DC or COL applications (to the extent that the SSC is applicable to the specific design being reviewed) include:

- Reactor
- Reactor coolant system and connected system including steam generators
- Engineered safety features
- Instrumentation and controls
- Electric power including offsite and onsite power systems
- Auxiliary systems
- Steam and Power Conversion System
- Radioactive Waste Management

Because the nuclear – industrial facility boundary will most likely involve a process heat transfer line, it is relevant to review NRC guidance for DC/COL applications in the area of power conversion systems. In NUREG-0800, SRP, Section 10.3, "Main Steam Supply System," the staff describes the review of the main steam supply system (MSSS) as it extends from the containment up to the turbine stop valve. The specific areas of review are specified as follows:

1. *The review should verify that portions of the MSSS that are essential for safe shutdown of the reactor or for preventing or mitigating the consequences of accidents are evaluated to determine the following:*
    a. *A single malfunction or failure of an active component would not preclude safety-related portions of the system from functioning as required during normal operations, adverse environmental occurrences, and accident conditions, including loss of offsite power.*
    b. *Appropriate quality group and seismic design classifications are met for safety-related portions of the system.*
    c. *The system is capable of performing multiple functions, such as transporting steam to the power conversion system, providing heat sink capacity or pressure relief capability, or supplying steam to drive safety system pumps (e.g., turbine-driven AFW pumps), as may be specified for a particular design.*
    d. *The MSSS design includes the capability to operate the atmospheric dump valves remotely from the control room following a safe-shutdown earthquake (SSE) coincident with the loss of offsite power so that a cold shutdown can be achieved by depending only on safety-grade components.*
2. *The MSSS review should include measures that limit blowdown of the system if a steam line were to break.*
3. *The review includes the design of the MSSS with respect to the following:*
    a. *Functional capability of the system to transport steam from the nuclear steam supply system as required during all operating conditions.*
    b. *Capability to detect and control system leakage and to isolate portions of the system in case of excessive leakage or component malfunctions.*
    c. *Capability to preclude accidental releases to the environment.*
    d. *Provisions for functional testing of safety-related portions of the system.*

NUREG-0800, Section 10.3, “Acceptance Criteria #3” [Technical Rational], states:

> *For multiple-unit sites, units may cross-connect the MSSSs for startup, maintenance, or other related purposes. For such shared systems, the licensee must show that each MSSS can perform all of its required safety functions for its respective unit. Meeting GDC 5 will ensure that shared MSSSs at multiple-unit sites will execute their respective safety functions regardless of malfunctions in the other units.*

NUREG-0800, Sections 10.4.1, “Main Condensers, Acceptance Criteria #1,” states:

> *Acceptability of the design of the MC [main condensers] and support systems, as described in the applicant's safety analysis report (SAR), is based on meeting the requirements of General Design Criterion 60 (GDC 60) and on the similarity of the design to that of plants previously reviewed and found acceptable. The design of the MC and support systems is acceptable if the integrated design of the system meets the requirements of GDC 60 as related to failures in the design of the system which do not result in excessive releases of radioactivity to the environment.*

NUREG-0800, Section 10.4.5, “Circulating Water System, Acceptance Criteria #1” [Technical Requirements] states:

> *GDC 4 requires that structures, systems, and components important to safety shall be designed to accommodate the effects and be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. Although the circulating water system is not safety related, GDC 4 establishes CWS design limits that will minimize the potential for creating adverse environmental conditions (e.g., flooding of systems and components important to safety). Meeting the requirements of this criterion provides a level of assurance that systems and components important to safety will perform their intended safety functions.*

NUREG-0800, Section 10.4.6, “Condensate Cleanup Systems, Acceptance Criteria #2” [Technical Requirements] states:

> *For indirect cycle (pressurized-water reactor (PWR)) plants, SRP Section 5.4.2.1 provides the criteria for acceptable secondary water chemistry. SRP Section 5.4.2.1 refers to the guidelines provided in the latest version in the EPRI report series, “PWR Secondary Water Chemistry Guidelines.”*

NUREG-0800, Section 10.4.7, “Condensate and Feedwater Systems, Acceptance Criteria #4,” regarding heat removal capability, states:

> *The requirements of GDC 44, as related to the capability to transfer heat from structures, systems and components important to safety to an ultimate heat sink are met by demonstrating that the CFS [condensate and Feedwater system] is capable of providing heat removal under both normal operating and accident conditions. Sufficient redundancy of components is demonstrated so that under accident conditions the safety function can be performed assuming a single active component failure (which may be coincident with the loss of offsite power for certain events.) The system demonstrates capability to isolate components, subsystems, or piping if required so that the system safety function will be maintained.*

Regulatory Guide 1.206, “Combined License Applications for Nuclear Power Plants (LWR Edition)”(Ref 8)

This RG contains a similar guidance as NUREG-0800, for describing SSCs in COL applications.

RG 1.206, Section C.III.1.8, states that the NRC staff expects COL applicants who reference a certified design to provide complete designs for the entire facility including appropriate site-specific design information to replace the conceptual design portions of the Design Certification Document (DCD) for the referenced certified design. Refer to Figure 3, extracted from RG 1.206, regarding the typical breakdown of design information between DC and COL applications.

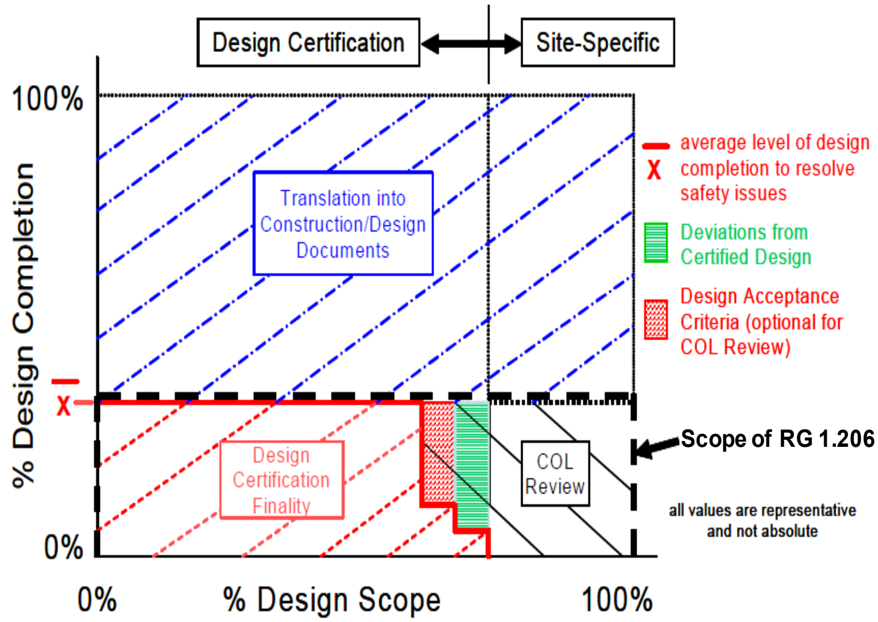

Figure 3. Combined License Application Referencing a Certified Design.

RG 1.206, Section C.I.10, "Steam and Power Conversion System," provides guidance regarding how the energy conversion portion of the plant should be described in a COL application.

> *This section should describe the secondary plant (steam and power conversion system), emphasizing those aspects of the design and operation that affect or could potentially affect the reactor and its safety features or contribute toward the control of radioactivity. The information provided should show the capability of the system to function without compromising (directly or indirectly) the safety of the plant, under both normal operating and transient situations. In addition, beginning with Section C.I.10.2 and for the other sections that follow, include a discussion of how the system design meets the applicable regulatory requirements and is consistent with the applicable regulatory guidance.*

RG 1.206, Section C.I.9, "Auxiliary Systems," states:

> *For systems that have little or no role in protecting the public against exposure to radiation, the description should provide enough information to allow the NRC staff to understand the design and operation and their effect on reactor safety, with emphasis on those aspects of design and operation that might affect the reactor and its safety features or contribute to the control of radioactivity.*

RG 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants - LWR Edition," Appendix A – "Interfaces For Standard Designs" (Ref 9)

As described in 10 CFR 52.47, the DC is to describe an essentially complete plant with the option that a representative conceptual design for those portions of the plant for which the application does not seek certification may be provided along with appropriate interface requirements. The conceptual design is intended to aid the NRC in its review of the DC FSAR and to permit assessment of the adequacy of the interface requirements. RG 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants - LWR Edition," Appendix A – "Interfaces For Standard Designs" provides guidance regarding acceptable approaches for describing standard plant interfaces.

> *Safety-related interfaces must be identified and defined for standard designs submitted under Option 1 (Reference Systems) of the Commission's standardization policy to establish the requirements that must be met and assumptions that must be verified by other unspecified portions of a nuclear plant design to ensure that systems, components, and structures within the standard design will perform their safety functions. Safety-related interfaces also include information that may be useful in the design and staff review of the unspecified portions of the plant design. The safety functions of a standard design are those essential functions that ensure (1) the integrity of the reactor coolant pressure boundary; (2) that the specified acceptable fuel design limits are not exceeded as a result of anticipated transients; (3) the capability to shut down the reactor and maintain it in a safe shutdown condition; and (4) the capability to prevent or mitigate the consequences of an accident that could result in radiation exposures in excess of applicable guidelines. Interfaces are used, therefore, to provide a basis for ensuring that the matching portions of a nuclear plant design, as described in a PSAR for a CP application that references the standard design or in another Standard Safety Analysis Report (SSAR) for a matching portion of the plant, are compatible with the standard design regarding the safety-related aspects of the plant design.*
>
> *This appendix describes safety-related interfaces, for light-water reactors only, that should be presented at the preliminary design stage of review by the reactor vendor in a Nuclear Steam Supply System SSAR (NSSS-SSAR) and by the architect-engineer in a Balance-of-Plant SSAR (BOP-SSAR). The interfaces for a BOP-SSAR, are also directly applicable to an SSAR describing an entire nuclear plant (NSSS plus BOP, but excluding utility- and site-specific items). This appendix also describes an acceptable format for presenting interfaces in an SSAR.*
>
> *Criteria for determining the acceptability of interfaces, as necessary for safety, are not included in this appendix. While not identified specifically as interface acceptance criteria, the criteria are part of other guidance already made available by the NRC, including that contained in the regulations, regulatory guides, and codes and standards.*

RG 1.70, Appendix A, II. “Sources of Interfaces” identified interfaces for standard designs as being derived from the following sources:

1. *Requirements for safe operation of the standard design that must be satisfied by matching portions of the plant design or by the utility (e.g. , cooling water and electric power requirements for the NSSS that must be provided by the BOP, an inservice inspection program for the NSSS and BOP that must be provided by the utility).*
2. *Assumptions made for the standard design that must be more precisely defined during the design coordination effort between the reactor vendor and the architect-engineer or between the architect-engineer and the utility (e.g., mass and energy release rates during a LOCA specified by the reactor vendor that must be coordinated with the containment design provided by the architect-engineer).*
3. *Site-related design assumptions upon which the standard design is based.*
4. *Criteria pertinent to the standard design described in the SSAR under review that may be useful for the design and staff review of matching systems, components, and structures (i.e., within the standard design, safety criteria for the items including codes and standards, General Design Criteria, and regulatory guides).*

2.2 NRC Historical Precedents

2.2.1 Midland Nuclear Plant

The application for a construction permit (CP) of the Midland Nuclear Plant, a dual pressurized water reactor (PWR) with each reactor core proposed at 2,452 MWt, was filed with the Atomic Energy Agency (predecessor agency to the NRC) on January 13, 1969. The CP application included the Preliminary Safety Analyses Report (PSAR) and 32 amendments (Ref 10). Following staff review and a public hearing before the Atomic Safety and Licensing Board (ASLB), CPs were issued on December 15, 1972. The application for an operating license was filed in 1977. Construction of the plant was halted and the plant was not completed as a nuclear power plant. However, the Midland plant presents the single identified historical precedent for a commercial nuclear power plant providing steam offsite to an industrial facility, a situation not unlike what is proposed for the HTGR.

A feature of the Midland Plant design was the provision to furnish process steam as well as electricity to an industrial facility located adjacent to the nuclear p1ant site. The steam in normal plant operation was to be furnished at various pressures and quantities [from 50 to 675 psia]. Two headers for each pressure were to transport the 191 psia and the 50 psia steam to the site boundary. A single additional header was to transport the 675 psia steam to the site boundary. The radioactivity content of the steam was required to comply with the limits set forth in 10 CFR, Part 20.

The Midland process steam control system was designed to control high- and low-pressure process steam to the industrial plant and to control transfers between process steam operating modes. There were three modes of operation. In Mode 1, Unit 1 supplied steam for both high-pressure evaporators and low-pressure evaporators. Extraction steam from the turbine provided the heating steam to low-pressure evaporators. Mode 2 was similar to Mode 1 except the heating steam to low-pressure evaporators was provided by means of pressure reducing valves from the main steam header. In Mode-3, Unit 2 supplied heating steam for both high-pressure evaporators and low-pressure evaporators. The control system was designed to provide smooth transfer from one mode of operation to the other.

Approximately 75 percent of the steam heat energy supplied by the nuclear boiler system was to be used to generate electrical energy. Steam containing the remaining heat energy was to be transported to the site boundary for process use by the industrial plant. Most of the steam was to be condensed and returned to the nuclear boiler system as heated feedwater. The steam not condensed was to be replaced by treated makeup from Dow.

Based on its review, the staff concluded that the power conversion system, including the provision to supply steam to the industrial facility, was in conformance with the regulatory criteria and design bases, could perform its designed functions, and was, therefore, acceptable.[q] The scope of this review is similar to that discussed in this paper for the energy conversion system.

2.3 Regulatory Foundation for Establishing Top-Level Regulatory Criteria

Top-level regulatory criteria for the transfer system can be established by reviewing example interface requirements in RG 1.206, Section 10, which provides the NRC guidance regarding FSAR content for the power conversion system and SRP Sections 10.2 through 10.4, which also address the power conversion system. The safety functions of the nuclear facility that must be preserved through the interface requirements would ensure the:

1. Integrity of the functional containment including the kernel and coatings of the coated fuel particles, the fuel matrix and fuel element graphite, primary helium heat transport circuit and reactor building

q. NUREG-0793, "Safety Evaluation Report related to the operation of Midland Plant, Units 1 and 2 Docket Nos. 50-329 and 50-330," , dated May 1982 (Ref 11).

2. Capability of the fuel to stay within design limits as a result of anticipated transients
3. Capability to shut down the reactor and maintain it in a safe shutdown condition
4. Capability to prevent or mitigate the consequences of an accident that could result in radiation exposures in excess of applicable guidelines.

2.4 Regulatory Foundation Summary

In general, NRC regulations and guidance specify that DC and COL applications together contain a complete description of the nuclear energy plant, including safety and non-safety portions of the plant. With respect to the non-safety portions of the plant, the staff expects that these SSCs be evaluated to ensure impacts to the safety basis are acceptable. The regulations and guidance documents do not describe situations such as the Midland arrangement with respect to scope of NRC regulatory jurisdiction. However, the Midland experience did provide an example where the NRC approved a configuration where process steam would be used in a facility not under their oversight. It appears reasonable to conclude that facilities that use process steam/heat that are located offsite could be considered outside of NRC regulatory jurisdiction given that the proper set of interface requirements are satisfied.

NRC regulations and guidance require that the plant descriptions in DC and COL applications shall be sufficient to permit understanding of the system designs and their relationship to safety evaluations. All items pertinent to fulfilling the safety analyses shall be described. It is anticipated that this will include items such as the reactor core, reactor coolant system, instrumentation and control systems, electrical systems, containment system, other engineered safety features, auxiliary and emergency systems, power conversion systems, radioactive waste handling systems, and fuel handling systems.

NRC guidance for DC applications allows for some systems to not be covered within the scope of the certification. The guidance for these systems specifies that conceptual design information and interface requirements be specified in the DC application. Site specific COL applications would then address these areas with site specific design.

Regulations for modular reactor plants require that an application for certification must describe and analyze the possible operating configurations of the reactor modules with common systems, interface requirements, and system interactions. The final safety analysis must also account for differences among the configurations, including any restrictions that will be necessary during the construction and startup of a given module to ensure the safe operation of any module already operating.

3. DEFINING THE NGNP NUCLEAR-INDUSTRIAL FACILITY AND DESIGN CERTIFICATION BOUNDARIES

3.1 Defining the Approach

Because the HTGR modular design can support different commercial applications, the site specific designs, that include the Energy Conversion Systems and configurations using multiple modules, can vary widely between applications. Given these varied applications, it is necessary to define two sets of regulatory boundaries to properly structure the possible HTGR licensing approaches and streamline the licensing process. First, there should be a clear understanding between the HTGR applicant and the NRC regarding those systems within the nuclear facility and under NRC regulatory jurisdiction (within the scope of the DC and COL applications) and those that fall outside NRC scope (industrial facility). Second, there should be a clear understanding regarding what plant scope is to be addressed in an HTGR DC and what scope can be addressed in a site specific COL application.

This paper first defines the boundary between the nuclear facility (those systems and programs under the regulatory jurisdiction of the NRC) and the industrial facility (those systems and programs that fall outside the regulatory jurisdiction of the NRC). Once that boundary is defined, the paper will define the boundary between the plant systems that should be addressed in a 10 CFR Part 52, DC application, and those that could fall outside of the DC scope and within the COL application scope.

3.2 The Nuclear Facility-Industrial Facility Boundary

In the past, NRC licensing of commercial nuclear power plants under 10 CFR Part 50 has generally involved the licensing of a complete plant including the nuclear steam supply system, support systems, and balance of plant systems (energy conversion systems). Typically, all of these systems were under NRC regulatory jurisdiction. These systems were within the typical site boundary, and most areas of the plant were within the security perimeter fence. As discussed in Section 2.2.1 above, there was one example in 1972 where the NRC authorized the construction of a commercial nuclear power plant, the Midland Nuclear Power Plant, which was to send steam generated from the nuclear plant to an offsite customer for use in an industrial application and receive condensate back from that customer. The customer's facility, including its energy demand systems, was located away from the nuclear plant site, was not part of the NRC licensing process and not under NRC's regulatory jurisdiction. While the Part 52 licensing process is different than the Part 50 process used at that time, the Midland example provides some insights into how such an arrangement could be licensed today.

Under 10 CFR Part 52, the NRC expects a license application for a nuclear power plant to include a complete design for the entire facility. The requirements for a COL application in 10 CFR 52, require that the final safety analysis report provide descriptions sufficient to permit understanding of the system designs and their relationship to safety evaluations. Items such as the reactor core, reactor coolant system, instrumentation and control systems, electrical systems, containment system, other engineered safety features, auxiliary and emergency systems, power conversion systems, radioactive waste handling systems, and fuel handling systems are required be discussed insofar as they are pertinent. The key term in this requirement is "insofar as they are pertinent." The following paragraphs provide a basis for defining what is pertinent with respect to an HTGR configuration that sends process steam or heat to an offsite user.

The SRP (and RG 1.206 for COL applications) specifies that complete descriptions of the structures, systems, and components discussed in Part 52 be provided to the NRC in the final DC or COL application. For site specific design information, the DC would provide conceptual design information, with the COL application providing the final design. In either case, before a license is to be issued under Part 52, a complete description of the plant is to be submitted to the NRC for review and approval. The areas covered by these documents are described in Section 2 of this paper.

The challenge for the HTGR applicant will be to describe enough of the plant and its interfaces so as to exclude the offsite energy demand systems of the customer while demonstrating that the sufficient protections are in place within the nuclear facility to provide reasonable assurance of safety regarding what transients might be initiated by customer operated systems.

While an obvious starting point for the boundary between the nuclear facility and industrial facility could be the physical nuclear plant site boundary or protected area boundary (the security fence), it is also necessary to define such a boundary on a system level, since certain systems will traverse any physical boundary. To accomplish this, the HTGR DC or COL application will show that its safety analyses bound any customer initiated transients involving these traversing systems. The HTGR safety analyses will describe bounding assumptions regarding the customer initiated transients and require that appropriate interface requirements be met by any process connections to the customer plant. There is precedence in the Part 52 DC process for the use of interface requirements for this type of approach. For example, in the Part 52 DC application process, when part of the plant is site specific and outside the scope of the DC, the DC provides interface requirements that must be met by any COL applicant and its site specific design. These interface requirements can take the form of process limits or equipment design requirements. For example, the DC may require that the COL application specify a site-specific ultimate heat sink that provides cooling of emergency service water such that the maximum supply water temperature is 95°F under the peak heat load conditions or that the site specific electrical system design ensures that the probability of losing power during the loss of power generated by the nuclear unit, the loss of power from the transmission network, or the loss of the largest load is minimized (Ref US-APWR, DCD, Tier 1, Section 3) (Ref 12). Other interface requirements may include criteria for site-specific fire-water supplies. These interface requirements used in LWR DC and COL licensing provide insight as to how HTGR licensing might provide adequate separation between the nuclear and industrial facilities.

To describe the boundary between the nuclear and industrial facilities it is necessary to discuss two aspects of the facility:

1. The area that must be protected under the NRC security requirements in 10 CFR 73.
2. The appropriate nuclear facility design requirements and associated interface requirements that need to be satisfied so that any transient or incident involving the industrial facility cannot create a condition that is outside the envelope of the HTGR's safety analyses.

3.2.1 Security Related Considerations

As discussed in Section 2, 10 CFR 73 defines, in part, requirements for the establishment and maintenance of a physical protection system that will have the capability to protect special nuclear material at fixed sites in which special nuclear material is used. Both vital areas and material access areas must be located within a protected area. Because of these security requirements, any nuclear facility boundary for the HTGR would need to encompass all of the areas of the plant that would need to be included within the plant's protected area (e.g., vital areas) as defined in their security plan.[r]

3.2.2 Nuclear Plant Design and Interface Considerations

A primary consideration for the nuclear-industrial boundary definition is that all SSCs that perform safety-related or risk significant functions for the HTGR would be within the nuclear facility boundary. The boundary is less clear with respect to other SSCs that are not safety-related or risk significant as these SSCs could present challenges to the plant or create transients requiring safety system mitigation. An approach to define this boundary for areas outside of the safety-related/risk significant SSCs is described below.

The standard HTGR plant will include a primary-to-secondary heat transfer device, such as a steam generator or an intermediate heat exchanger, that transfers heat from the helium primary system to a secondary medium (water in the case of the steam generator and helium in case of an indirect process heat supply system). This secondary medium could then transfer steam/process heat to an energy conversion system such as (1) an onsite electrical generation device and/or (2) a transfer system made up of pipes, valves, pumps, instrumentation, etc, that provides the secondary steam or gas to an offsite customer and returns the fluid or gas to the HTGR primary system heat exchanger. This transfer system would start at the secondary side outlet of the primary system heat exchanger, traverse across the HTGR site (nuclear facility), leave the HTGR site and enter the customer's facility (industrial facility). A similar transfer line would provide return flow back to the HTGR heat exchanger. The logical interface boundary between the two facilities would be at some point in the transfer system before the feeding part of the system departs from the HTGR site and after the return line enters the HTGR site. The energy transfer function of this pipe is not unlike a transmission cable leaving the site that transfers electric power offsite. The interface requirements and the nuclear facility side protection devices must be defined sufficiently so that the safety analysis can be bounded for all possible transients that could be initiated at the industrial facility.

Based on the requirements in Part 52, guidance in the SRP and RG 1.206, and industry precedents, an energy conversion system located within the HTGR protected area, such as a turbine generator that produces electric power, is integral to the operation of the nuclear side of the plant and controlled from

r. 10 CFR 73.2 defines *protected area* as an area encompassed by physical barriers and to which access is controlled.

the HTGR control room would be considered within the nuclear facility and not within the industrial facility. This conclusion is based on 10 CFR 52.47 and 52.79 requirements for DC and COL applications to describe systems "insofar as they are pertinent," and the integral relationship this onsite electric power system would have with the nuclear facility including (but not limited to) electric plant control from the HTGR control room facility, impact on electric power supplies to the HTGR plant, the potential for turbine generator missiles, proximity with respect to security issues, water quality of steam generator feed, cooling tower plume impacts, and flooding issues with the condenser cooling system. However, it is justifiable to exclude from the nuclear facility and Part 52 licensing scope an energy demand system (such as a process heat system for a petro-chemical process or an offsite turbine generator) located outside of the protected area that is independent from the HTGR site such that the system is not controlled from the HTGR facilities nor is the HTGR dependent on, or adversely affected by, any system outputs, provided that appropriate interface requirements are established for the transfer system. Regardless of whether the energy conversion/demand system is within the nuclear facility or not, analysis would be required with respect to potential missiles, security issues, flooding issues, process stream feedback or other impacts to HTGR SSCs that perform a safety function. An offsite energy demand system would require a process heat transfer system that would serve as the interface between the HTGR site and the customer site. An analysis would be required to be performed of the potential impacts that the transfer system might impose on the HTGR and both preventative and mitigative measures would be necessary based on the safety analyses.

To fully understand the scope of this analysis, a review of SRP guidance and RG 1.206 in the area of energy conversion systems can provide insight. These guidance documents describe the regulatory requirements and acceptance criteria that such systems need to meet. If one considers the aforementioned energy transfer system as akin to a main steam supply system in a pressurized LWR, it would be expected that this system would have monitoring and, if necessary, isolation capability similar to MSIVs[s]. If the downstream portion of the process heat transfer system ran to the customer property, then appropriate interface requirements would need to be established for the section of pipe leading up to the point where the nuclear facility isolation or other protection devices would exist. Similarly, the condensate return line from the industrial facility to the nuclear facility would also need to be evaluated for impacts such as line breaks, water quality for use in the steam generator, and heat removal needs.

Example interface requirements can be noted in RG 1.206, Section 10, which provides the NRC guidance regarding FSAR content for the power conversion system and SRP Sections 10.2 through 10.4, which also address the power conversion system. In reviewing these examples, a set of high level design requirements and interface requirements can be developed for the transfer system. The combination of nuclear facility transfer system design and interface requirements that would be imposed on the site-specific portion of the transfer system design (industrial facility) would need to demonstrate that all applicable regulatory requirements for the energy conversion system would be met. A review of the aforementioned regulatory guidance documents has identified the following list of functional requirements that would be imposed on the combination of nuclear facility transfer system design and interface requirements that would need to be met to satisfy applicable regulatory requirements:

1. Failures or transients within the industrial facility portion of the transfer system would not preclude safety-related portions of the nuclear facility from functioning as required during normal operations, anticipated operational occurrences, and accident conditions.
2. Nuclear facility plant system transients caused by industrial facility systems or the electrical transmission grid would be limited (in frequency and severity) and analyzed in the plant's safety analyses, similar to the way transmission grid disturbances are evaluated in existing LWRs.

s. The HTGR safety analysis may determine that such isolation capability is not required in which case this design feature would not be a boundary consideration.

3. No portion of the transfer system within the scope of the industrial facility would be required to perform any safety or safe shutdown function, or be relied upon as a supporting system to a safety-related system.
4. The transfer system would have monitoring capability to detect disturbances and, if required by the HTGR safety analysis, isolate the industrial facility system during transients and accidents.
5. Releases of radioactive material from the transfer system would meet required limits. Monitoring and/or sampling would be performed, as necessary, to ensure limits are met.

Having met the above functional requirements for each interface with the industrial facility, an appropriate nuclear facility boundary can be determined. Any components within the nuclear facility boundary would need to physically exist within the protected area boundary to satisfy security requirements.

3.3 Design Certification Boundary

Having defined the nuclear facility as those systems that fall within the regulatory jurisdiction of the NRC, the next step is to determine the scope of HTGR systems that should fall within the DC and those that could be addressed in a COL application. This discussion will focus on the boundary between the DC and the site specific portion of the plant within the nuclear facility.

Because of the modularity aspect of HTGRs, future HTGR DC applications may only request certification for a portion of what is typically part of a LWR nuclear facility and what has been submitted in recent LWR DC applications. While areas such as the control room, radwaste facility, and reactor service building may be included within the nuclear facility boundary, they may be excluded from a HTGR design certification along with the typical secondary side design by defining interface requirements for these systems and structures. The basis for this approach is provided below.

The standard plant systems would include those expected to be described in a design certification (DC) document. 10 CFR 52.47 describes the type of information required to be included in a DC. For a modular nuclear reactor design, the DC must describe and analyze the possible operating configurations of the reactor modules with common systems, interface requirements, and system interactions. The DC final safety analysis must also account for differences among the configurations, including any restrictions that will be necessary during the construction and startup of a given module to ensure the safe operation of any module already operating.[t] The plant systems described in the DC would be reviewed and approved by the NRC. The DC would contain interface requirements for those portions of the plant that are site specific (see Item 2 below) and outside of the DC.

Part 52 provides an acceptable approach for developing a design certification application for a standard HTGR module as part of a single or multi-module reactor plant that uses different site specific portions of the nuclear facility and different Energy Conversion Systems (e.g., turbine generators for electric power production and/or process steam delivery system equipment) for the modules. The certified portion of the plant would include the standard parts of the nuclear facility excluding any site specific design information. The DC application would provide conceptual design information for a typical Energy Conversion System(s) and necessary interface requirements. Interface requirements would be needed to address:

1. Requirements for safe operation of the standard design that must be satisfied by matching portions of the site-specific design
2. Site-related design assumptions upon which the standard design is based

t. 10 CFR 50.47(c)(3).

3. Criteria pertinent to the standard design described in the DC application that may be useful for the design and NRC review of matching systems, structures and components (within the standard design, safety criteria for the items including codes and standards, Principal Design Criteria, and regulatory guides)
4. Requirements to preserve the following specific HTGR safety functions:
    a. Prevent or mitigate the consequences of Design Basis Events (DBEs) to comply with 10 CFR §50.34 offsite dose limits
    b. Prevent the frequency of Beyond Design Basis Events (BDBEs) with consequences greater than the 10 CFR §50.34 dose limits from increasing into the DBE region
    c. Mitigate the consequences of Anticipated Operational Occurrences (AOOs) to comply with the offsite dose limits of 10 CFR Part 20
    d. Prevent the frequency of DBEs with consequences greater than the 10 CFR Part 20 offsite dose limits from increasing into the AOO region.

As discussed in Section 2, per 10 CFR 52.47(a)(24) a representative conceptual design for those portions of the plant for which the application does not seek certification is necessary to aid the NRC in its review of the FSAR and to permit assessment of the adequacy of the interface requirements. The interface requirements must be sufficiently detailed to allow completion of the FSAR.

The certified portion of the reactor plant design and the safety analysis would bound all worst case operating and accident scenarios for potential site-specific Energy Conversion Systems. The design certification application would also include a conceptual design description of the equipment and interface requirements for the potential operating configurations. This conceptual design information would not be included in the final certified design. Each COL application that referenced this design certification would need to describe the site-specific design and operating information, and show that the site specific systems including the Energy Conversion System satisfy the applicable DCD interface requirements. NRC would then review and document approval of this COL application information in a safety evaluation report. Subsequent COL applications, or S-COLAs, referencing the same design certification and using the same site specific systems such as the Energy Conversion System could replicate the information provided in the initial Reference COL application, or R-COLA and not be subject to additional NRC review of this information under the NRC's "one issue, one review, one position" design centered review approach.[u]

The certification boundary for the HTGR would include the following SSCs, as a minimum:

- Reactor pressure vessel, including core support structure, fuel, reactivity control system, and graphite matrix
- Cross-vessel(s)
- Primary-to-secondary heat transfer pressure vessel(s)
- Piping connecting the primary helium circuit to support systems, (e.g., shutdown cooling system, primary helium service and purification system)
- Support systems: shutdown cooling system, primary helium service and purification system
- NHSS protective system
- Secondary piping penetrating the primary-to-secondary heat transfer pressure vessel(s) up to the isolation valves, if applicable

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u. Refer to Regulatory Information Summary 2006-06, *New Reactor Standardization Needed To Support The Design-Centered Licensing Review Approach* (Ref 13)

- If required by the safety analysis[v], isolation valves in secondary steam and/or high temperature fluid supply from and return to the industrial facility and Energy Conversion Systems
- Reactor building
- New and spent fuel handling systems within the reactor building.

The DC application would address, as necessary to provide the degree of flexibility desired by the applicant regarding the deployment of the HTGR, the interfaces, transients and accident conditions for the full range of HTGR-NHSS and Energy Conversion System configurations, operating conditions and process demands, and integrated risk including total accident source terms. The DC would also address operation in a multi-module installation of varying ratings and configurations (e.g., up to [12][w] modules), operation while one or more other modules are being constructed and tested, and operation while one or more other modules are in refueling, shutdown for maintenance, or being decommissioned.

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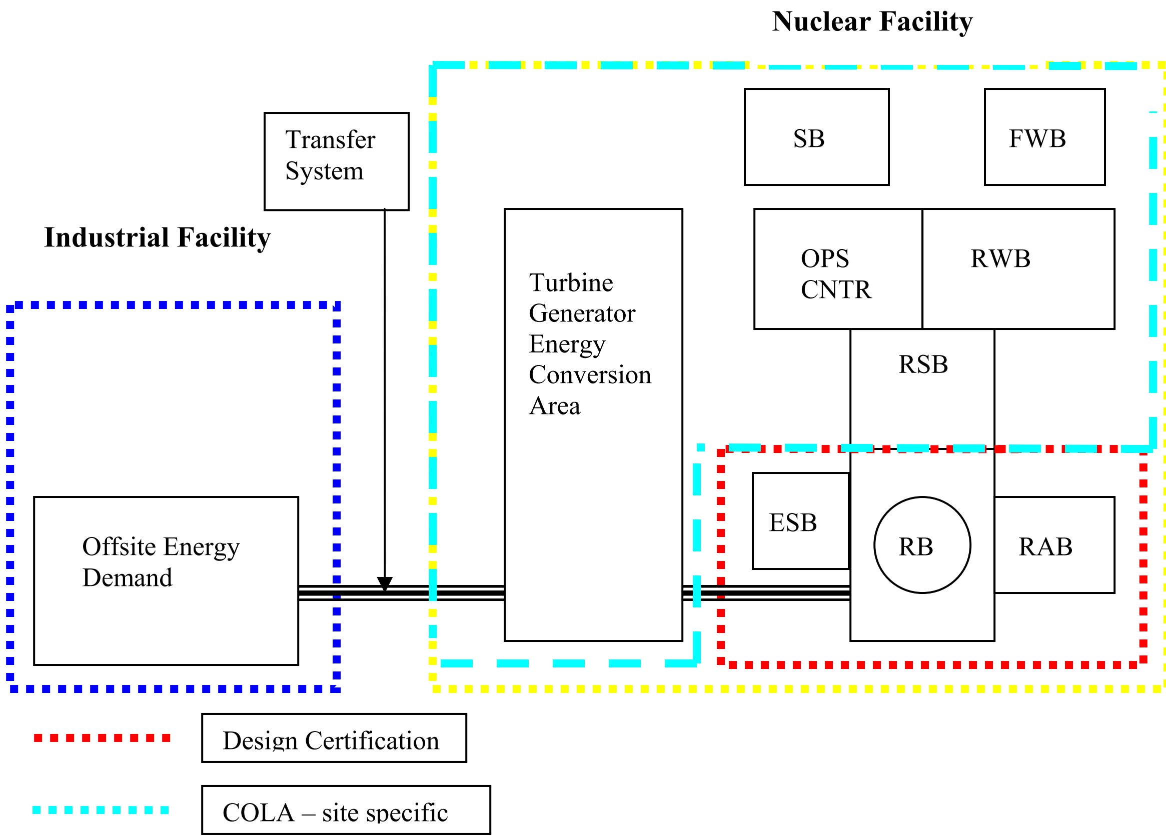
v. As discussed in *Next Generation Nuclear Plant (NGNP) Prismatic HTGR Conceptual Design Project Evaluation of Alternate HTGR Technology Applications*, NGNP-R00017, Rev 0, prepared by General Atomics, for a design that uses an intermediate heat exchanger the need for isolation valves on the secondary side of the intermediate heat exchanger remains an open issue. (Ref 14)

w. Items in brackets [ ] are representative; final values will be defined as the design of the HTGR – NHSS progresses.

Figure 4 illustrates typical demarcation lines for a single module HTGR between the nuclear facility and industrial facility, and the DC and COL application discussed in this paper.



RB – Reactor Building including reactor vessel, primary circuit, cross vessel, secondary circuit pressure vessel, piping connecting the primary helium circuit to support systems, (e.g., shutdown cooling system, primary helium service and purification system)
RAB – Reactor Auxiliary Building
ESB – Electrical Service Building
RSB – Reactor Support Building
OPS CNTR – Operations Center and Control Room
RWB – Radwaste Building
SB – Security Building
FWB – Fire Water Building and Fire Pump House

Figure 4 – Notional Regulatory Demarcation Boundaries

It is noted that General Atomics (GA), in their Conceptual Design Report submitted to the U.S. Department of Energy, proposed a 350-MWt SC-MHR, high temperature, gas-cooled, graphite-moderated reactor utilizing a prismatic graphite block fuel form that would provide process heat/steam to an offsite industrial facility. The plant arrangement for the demonstration plant consists of two onsite areas – nuclear island (NI)[x] and the onsite Energy Conversion Area (ECA)[y]. The NI contains the Reactor

[x] The term *Nuclear Island* used in the GA report is not synonymous with the term *nuclear facility* used in this report to define the systems within the NRC oversight boundary.

Building and other SSCs comprising the standard reactor module (SRM) and the adjacent balance-of-NI structures housing SSCs related to plant control, fuel handling and storage, and various reactor service and auxiliary systems. The ECA constitutes the balance of plant, including the turbine generator(s) for electricity production and the process steam delivery system equipment. While the GA CDR does not specially address regulatory boundaries, GA proposed to seek a DC for the SRM portion of the DP. The scope of this SRM, and an associated DC application, provides an example for discussion and includes the following:

- SSCs within the Reactor Building
- SSCs within the Reactor Auxiliary Building
- SSCs within the Electrical Services Building
- NI cooling water system
- Spent Fuel Cooling Water System
- Shutdown Cooling Water System

Other SSCs within the proposed GA NI such as the control room, reactor service building and radwaste building would not be within the scope of the certified design. The ECA would also not be included within the scope of the SRM. A DC for such a SRM would then need to provide conceptual design information and interface requirements for the portion of the NI not addressed as part of the DC, and the ECA systems and structures.

The process heat lines that traverse offsite would be part of the nuclear facility scope up to the point of the nuclear – industrial facility boundary at which they would enter the industrial facility part of the facility. This line would need to satisfy the boundary interface requirements discussed in Section 3.2 of this paper.[z]

3.4 Scope Defined in Site-specific COL Application

The scope of the plant that would be site-specific (outside the DC scope) would fall into two subcategories:

1. Plant systems that are not part of the DC but would be expected to be described in a COL application. The COL application would address interface requirements identified in the DC for those systems not within the DC.
2. Plant systems that would not be expected to be described in detail in the DC or the COL application, except to describe how applicable DCD/COL application interface requirements are met by these systems. These systems would be part of the industrial facility. The detailed description of these plant systems and programs would not be reviewed and approved by the NRC. Depending on the site specific design, the COLA will contain specific interface requirements for those portions of the industrial plant that interface with the COLA systems but are not described in detail in the COLA because they fall within the scope of the industrial facility and outside the regulatory jurisdiction of the NRC.

The COL application referencing a DC would need to provide site specific design information for all areas addressed as conceptual design in the applicable DC including the Energy Conversion System. The COL application would also need to provide information demonstrating that the site specific design satisfied the interface requirements in the DC. For a COL application that does not reference a DC, the

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[y] While the ECA with the turbine generator was considered physically separate it was still within the HTGR site and therefore still considered within the *nuclear Island* boundary from a regulatory oversight perspective.

[z] NGNP-R00016, Revision 0, Next Generation Nuclear Plant (NGNP), Prismatic HTGR Conceptual Design Project Conceptual Design Report – Steam Cycle Modular Helium Reactor (SC-MHR) Demonstration Plant (Ref 1)

applicant will need to submit design information on the entire plant within the nuclear facility and should not include any conceptual design information.

The first COL application for a site specific plant arrangement could serve as the R-COLA, and S-COLAs referencing the same design certification and using the same site specific systems such as the Energy Conversion System could replicate the information provided in the R-COLA and not be subject to additional NRC review of this information under the NRC's "one issue, one review, one position" design centered review approach.[aa]

aa. Refer to Regulatory Information Summary 2006-06, *New Reactor Standardization Needed To Support The Design-Centered Licensing Review Approach* (Ref 13)

3.5 Scope Outside of COL Application

No specific system descriptive information would be needed for the COL application for the scope of the plant outside the nuclear facility, since this part of the plant would be outside the scope of the NRC review. Instead, the COL application would demonstrate how interface requirements specified in either the COL application or DC would be met by the specific industrial facility interface.

Figure 5 illustrates the overall nuclear-industrial facility boundary approach being presented above.

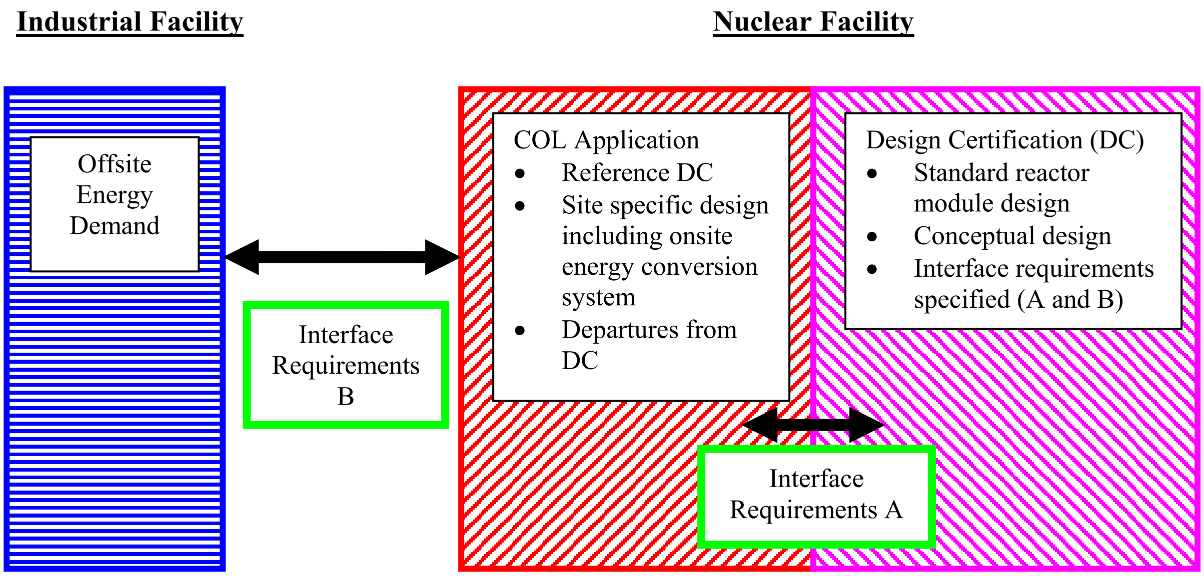

Figure 5. Illustration of approach to nuclear-industrial facility and DC/COL boundaries

3.6 Protection of Plant from Transients and Hazards Generated from Facilities Outside of NRC Regulatory Jurisdiction

As discussed earlier, the DC safety analyses would bound any transients initiated within the industrial facility. For example, the DC might assume a certain size explosion hazard. The DC would need to specify appropriate analyses demonstrating that the offsite explosion hazard was bounded by the DC analyses. The COLA would provide analysis demonstrating that the DC interface requirements were met.

However, specific system descriptions of the industrial facility portion of the plant would not be required in the COLA beyond that needed to demonstrate that the interface requirements and types hazards were properly analyzed (e.g., providing a list of hazardous chemicals, their quantities and distance from the site buildings).

4. OUTCOME OBJECTIVES

The NGNP Project is seeking general concurrences and/or comments from the NRC regarding an approach to defining a boundary between the HTGR nuclear facility, under the regulatory jurisdiction of the NRC (within the scope of the DC and COL applications), and the interfaces to an end user facility that would fall outside the scope of the NRC (industrial facility), as well as defining the plant scope to be addressed in an HTGR DC. The NGNP Project is seeking agreement on the following specific aspects of its approach to define these boundaries:

1. Since the NRC clearly has regulatory jurisdiction over plant facilities that are required to be protected under physical security regulations, SSCs needed to be within the plant's security boundary would be part of the nuclear facility.
2. All SSCs that perform safety-related or risk-significant functions for the HTGR would be within the nuclear facility boundary.
3. An energy conversion system that is located within the HTGR protected area boundary, is integral to the HTGR facility, and is controlled by the HTGR control room would be considered within the nuclear facility. An energy conversion system can be excluded from the nuclear facility scope if it is located outside the protected area boundary and is separated from the HTGR facility by a transfer system with interface criteria that act to ensure that the HTGR facility is not dependent on, or adversely affected by, events that occur within the separate industrial facility.
4. Regardless of whether the energy conversion system is within the nuclear facility or within the separate industrial facility, analysis would be required with respect to potential missiles, security issues, flooding issues, or other impacts to HTGR SSCs that perform a safety function.
5. Beyond the criteria defined in Items 1 through 4 above, the boundary between the HTGR nuclear facility and the industrial facility with respect to regulatory jurisdiction can be defined by properly describing the nuclear facility system design, the transfer system design, and interfaces using an appropriate set of conceptual design information and interface requirements. The following represent an appropriate set of high-level design and interface requirements for this boundary[bb]:
    a. Failures or transients within the industrial facility portion of the transfer system would not preclude safety-related portions of the nuclear facility from functioning as required during normal operations, anticipated operational occurrences, and accident conditions.
    b. Nuclear facility plant system transients caused by industrial facility systems or the electrical transmission grid would be limited (in frequency and severity) and analyzed in the plant's safety analyses, similar to the way transmission grid disturbances are evaluated in existing LWRs.
    c. No portion of the transfer system within the scope of the industrial facility would be required to perform any safety or safe shutdown function, or be relied upon as a supporting system to a safety-related system.
    d. The transfer system would have monitoring capability to detect and, if required by the HTGR safety analysis, isolate the industrial facility system during transients and accidents.
    e. Releases of radioactive materials from the transfer system would meet required limits. Monitoring and/or sampling would be performed, as necessary, to ensure limits are met.
6. No specific system descriptive information would be needed for the DC/COL application for the scope of the plant outside the nuclear facility, since this part of the plant would be outside the scope of the NRC review. Instead, the COL application would demonstrate how interface requirements

bb. Any interface with the industrial facility would involve a transfer system that would provide steam or process heat to the customer and return condensate or makeup fluid to the nuclear facility.

specified in either the COL application or DC would be met by the specific industrial facility interface

7. The HTGR nuclear facility can be further divided into systems that should be addressed within a 10 CFR Part 52 DC application and those that may be appropriately described in a site-specific Part 52 COL application. The DC application would provide, as necessary address the degree of flexibility desired by the applicant regarding the deployment of the HTGR, a description and analyses of the possible operating configurations of the reactor modules with common systems, interface requirements, and system interactions, accounting for differences among the configurations, including any restrictions that will be necessary during the construction and startup of a given module to ensure the safe operation of any module already operating. At a minimum, the SSCs within the scope of an HTGR DC should include:
   a. The reactor pressure vessel, including core support structure, fuel, reactivity control system, and graphite matrix
   b. The cross-vessel(s)
   c. The primary-to-secondary heat transfer pressure vessel(s)
   d. Piping connecting the primary helium circuit to support systems, (e.g., shutdown cooling system, primary helium service and purification system)
   e. Support systems: shutdown cooling system, primary helium service and purification system
   f. NHSS protective system
   g. Secondary piping penetrating the primary-to-secondary heat transfer pressure vessel(s) up to the isolation valves, if applicable
   h. If required by the HTGR safety analysis, the isolation valves in secondary steam and/or high temperature fluid supply to and return from the industrial facility and/or Energy Conversion Systems
   i. Reactor building
   j. New and spent fuel handling systems within the reactor building.

8. Conceptual design information and interface requirements would be provided in the HTGR DC application, as appropriate, for SSCs not within the scope of the DC. The interface requirements would address[cc]:
   a. Requirements for safe operation of the standard design that must be satisfied by matching portions of the site specific design
   b. Site-related design assumptions upon which the standard design is based
   c. Criteria pertinent to the standard design described in the DC application that may be useful for the design and review of matching systems, components, and structures (within the standard design, safety criteria for the items including codes and standards, Principal Design Criteria, and regulatory guides)
   d. Requirements to preserve the following specific HTGR safety functions[dd]:
      (1) Prevent or mitigate the consequences of DBEs to comply with 10 CFR §50.34 offsite dose limits

cc. These interface requirements for site specific SSCs are beyond those specified for the nuclear – industrial facility boundary.

dd. Functions from NGNP white paper, INL/EXT-10-19509, “Structures, Systems, and Components Safety Classification,” dated September 2010.

(2) Prevent the frequency of BDBEs with consequences greater than the 10 CFR §50.34 dose limits from increasing into the DBE region

(3) Mitigate the consequences of AOOs to comply with the offsite dose limits of 10 CFR Part 20

(4) Prevent the frequency of DBEs with consequences greater than the 10 CFR Part 20 offsite dose limits from increasing into the AOO region.

9. A site specific COL application that references a DC would provide site specific design information for all areas addressed as conceptual design in the applicable DC including the Energy Conversion System if this system is within the nuclear facility boundary. Additionally, the COL application would need to provide information demonstrating that the site specific design satisfied the interface requirements in the DC. This includes verification that the nuclear-industrial facility boundary interface requirements were satisfied for the site-specific design.

10. For a COL application that does not reference a DC, the applicant will need to submit design information on the entire facility within the nuclear facility and should not include any conceptual design information for the facility. This COL application would need to describe the nuclear-industrial facility boundary interface requirements (see Item 4 above) and show that they are satisfied by the site-specific design.

5. REFERENCES

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